NEXTera ENERGY DUANE ARNOLD

October 18, 2018

NG-18-0121
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Duane Arnold Energy Center
Docket No. 50-331
Renewed Op. License No. DPR-49

Subject: Response to Second Round - Request for Additional Information (RAI) – Duane Arnold Energy Center (DAEC) – LAR TSCR-166, Adoption of EAL Scheme Pursuant to NEI 99-01 – EPID L-2017-LLA-420

References:

1. NextEra Energy Duane Arnold, LLC letter NG-17-0235, License Amendment Request (TSCR-166), Adoption of EAL Scheme Pursuant to NEI 99-01, Revision 6, dated December 15, 2017 (ML17363A069)
2. NextEra Energy Duane Arnold, LLC letter NG-18-0090, Response to Request for Additional Information Regarding License Amendment Request (TSCR-166), Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," July 26, 2018 (ML18212A232)
3. NRC E-Mail: Final Second Round - Request for Additional Information (RAI) - Duane Arnold Energy Center (DAEC) - LAR TSCR-166, Adoption of EAL Scheme Pursuant to NEI 99-01 - EPID L-2017-LLA-0420, October 9, 2018

In Reference 1, NextEra Energy Duane Arnold, LLC (hereafter NextEra Energy Duane Arnold) submitted a License Amendment Request for the Duane Arnold Energy Center (DAEC) pursuant to 10 CFR 50.90. In Reference 2, NextEra Energy Duane Arnold submitted additional information regarding that application in response to a NRC staff request for additional information. Subsequent to that submittal, in Reference 3, NRC staff have identified an additional request for additional information.

Enclosure 1 of this letter provides the NextEra Energy Duane Arnold response to the NRC staff's second round request for additional information.

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NRR

The following information is provided as attachments to the Enclosure to aid NRC review and approval and replaces the Attachments in their entirety from References 1 and 2:

- Attachment 1 - Updated Redline Markup of NEI 99-01 Revision 6
- Attachment 2 - Updated Clean Copy of the Proposed DAEC EAL Scheme
- Attachment 3 - Updated Deviations and Differences Matrix
- Attachment 4 - Updated Supporting Technical Information
- Attachment 5 - Updated DAEC EAL Scheme Wallboards

This additional information does not impact the 10 CFR 50.92 evaluation of "No Significant Hazards Consideration" previously provided in the referenced application.

This letter makes no new commitments and does not change any existing commitments.

If you have any questions regarding this matter, please contact Michael Davis, Licensing Manager at (319) 851-7032.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 18, 2018

Dean Curtland
Site Director
NextEra Energy Duane Arnold, LLC

Enclosure

cc: Regional Administrator, USNRC, Region III,
Project Manager, USNRC, Duane Arnold Energy Center
Resident Inspector, USNRC, Duane Arnold Energy Center
A. Leek (State of Iowa)

ENCLOSURE 1

DUANE ARNOLD ENERGY CENTER

Response to Second Round - Request for Additional Information (RAI)
Duane Arnold Energy Center (DAEC)
LAR TSCR-166, Adoption of EAL Scheme Pursuant to NEI 99-01
EPID L-2017-LLA-420

RAI-DAEC-1

The proposed value for the Potential Loss of Fuel Clad Barrier threshold is based on the guidance provided by NEI 99-01, Revision 6, and is not based on site-specific conditions. Section 3.3, "NSSS Design Differences," of NEI 99-01, Revision 6, provides that, "Developers will need to consider the relevant aspects of their plant's design and operating characteristics when converting the generic guidance of this document into a site-specific classification scheme."

Please verify that the Loss of the Fuel Clad Barrier threshold value for the Drywell and Torus radiation monitors, based on a loss of the reactor coolant system resulting in approximately 2% to 5% clad damage, considers the site-specific aspects of the DAEC design and operating characteristics. If the fuel clad barrier threshold value is not based on DAEC site-specific values, revise the proposed fuel clad barrier threshold value to align with site-specific values.

DAEC Response

After further clarifying discussion with the NRC staff during an October 3, 2018 telephone call, NextEra Energy Duane Arnold acknowledges that the previously proposed calculated values were based solely on a reactor coolant system activity of 300 uci/gm Dose Equivalent Iodine (DEI) as provided by the applicable Developer Notes of NEI 99-01, Revision 6; and as such did not accurately reflect a site-specific range of fuel clad damage of approximately 2% -5%. NRC staff have clarified that the intent of this EAL is for the threshold value to provide a radiation monitor threshold value indicative of fuel clad damage of approximately 2% -5%.

In the attachments to this letter, NextEra Energy Duane Arnold provides revised threshold values for the Drywell and Torus radiation monitors indicative of Loss of the Fuel Clad Barrier that are based on approximately 5% clad damage.

NextEra Energy Duane Arnold intends to formally revise validation document V-23 (engineering calculation NEE-323-CALC-001) contained in Attachment 4 to reflect this new understanding of the intent of the EAL threshold prior to implementation of the revised EALs. The table on the following page provides the updated 5% clad release threshold values based on a scaling of the present 100% and 20% clad damage values provided in the current calculation.

Development of Fission Product Barrier EAL Threshold Values from NEE-323-CALC-001

NOTE: Fuel Clad barrier LOSS 4.A(B) threshold values below are scaled from the 100% gap release instead of calculated based on 300uci/gm DEI as assumed in NEI 99-01 Revision 6 developer guidance. This variation from the NRC endorsed guidance is due to the calculated value not reflecting the intended 2-5% gap release threshold due to differences in plant design. The calculation will be formally revised to reflect this change in methodology.

	Drywell dose rate R/hr	Torus dose rate R/hr	Drywell dose rate R/hr	Torus dose rate R/hr
1691 MWth 100% Gap release	22700	2140	Values below are rounded for ease of use, as well as to provide a step-wise progression	Values below are rounded for ease of use, as well as to provide a step-wise progression
Scaling factor to account for power uprate to 1912 MWth	1.13	1.13		
Updated 100% Gap release	25667	2420		
After application of 0.2 scaling factor for 20% Gap release *CTMNT barrier LOSS 4.A(B)*	5133	484	**5000**	**500**
After application of 0.05 scaling factor for 5% Gap release *Fuel Clad barrier LOSS 4.A(B)*	1283	121	**1250**	**125**

ATTACHMENT 1

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATING TO

LICENSE AMENDMENT REQUEST TSCR-166

UPDATED REDLINE MARKUP OF NEI 99-01 REVISION 6

311 pages follow

~~NEI 99-01 [Revision 6]~~

~~Development of Emergency Action Levels for Non-Passive Reactors~~

~~November 2012~~

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~~NEI 99-01 [Revision 6]~~

~~Nuclear Energy Institute~~
Duane Arnold Energy Center (DAEC) Emergency Action Levels Technical Bases Document

TBD, 2018 ~~November 2012~~

~~ACKNOWLEDGMENTS~~

~~This document was prepared by the Nuclear Energy Institute (NEI) Emergency Action Level (EAL) Task Force.~~

~~**NEI Chairperson:** David Young~~

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~~EXECUTIVE SUMMARY~~

~~Federal regulations require that a nuclear power plant operator develop a scheme for the classification of emergency events and conditions. This scheme is a fundamental component of an emergency plan in that it provides the defined thresholds that will allow site personnel to rapidly implement a range of pre-planned emergency response measures. An emergency classification scheme also facilitates timely decision-making by an Offsite Response Organization (ORO) concerning the implementation of precautionary or protective actions for the public.~~

~~The purpose of Nuclear Energy Institute (NEI) 99-01 is to provide guidance to nuclear power plant operators for the development of a site-specific emergency classification scheme. The methodology described in this document is consistent with Federal regulations, and related US Nuclear Regulatory Commission (NRC) requirements and guidance. In particular, this methodology has been endorsed by the NRC as an acceptable approach to meeting the requirements of 10 CFR § 50.47(b)(4), related sections of 10 CFR § 50, Appendix E, and the associated planning standard evaluation elements of NUREG-0654/ FEMA-REP-1, Rev. 1,~~ ~~*Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*~~~~, November 1980.~~

~~NEI 99-01 contains a set of generic Initiating Conditions (ICs), Emergency Action Levels (EALs) and fission product barrier status thresholds. It also includes supporting technical basis information, developer notes and recommended classification instructions for users. Users should implement ICs, EALs and thresholds that are as close as possible to the generic material presented in this document with allowance for changes necessary to address site-specific considerations such as plant design, location, terminology, etc.~~

~~Properly implemented, the guidance in NEI 99-01 will yield a site-specific emergency classification scheme with clearly defined and readily observable EALs and thresholds. Other benefits include the development of a sound basis document, the adoption of industry-standard instructions for emergency classification (e.g., transient events, classification of multiple events, upgrading, downgrading, etc.), and incorporation of features to improve human performance. An emergency classification using this scheme will be appropriate to the risk posed to plant workers and the public, and should be the same as that made by another NEI 99-01 user plant in response to a similar event.~~

~~The individuals responsible for developing an emergency classification scheme are strongly encouraged to review all applicable NRC requirements and guidance prior to beginning their efforts. Questions concerning this document may be directed to the NEI Emergency Preparedness staff, NEI EAL task force members or submitted to the Emergency Preparedness Frequently Asked Questions process.~~

~~Finally, unique State and local requirements associated with an emergency classification scheme are not reflected in this guidance. Incorporation of these requirements may be performed on a case-by-case basis in conjunction with the appropriate ORO agency. Any such changes will require a review under the applicable sections of 10 CFR 50.~~

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~~DEVELOMENT OF~~ DUANE ARNOLD EMERGENCY ACTION LEVELS ~~FOR NON-PASSIVE REACTORS~~ TECHNICAL BASIS DOCUMENT

1 BASIS FOR EMERGENCY ACTION LEVELS~~REGULATORY BACKGROUND~~

1.1 OPERATING REACTORS

Title 10, Code of Federal Regulations (CFR), Energy, contains the U.S. Nuclear Regulatory Commission (NRC) regulations that apply to nuclear power facilities. Several of these regulations govern various aspects of an emergency classification scheme. A review of the relevant sections listed below will aid the reader in understanding the key terminology provided in Section 3.0 of this document.

- 10 CFR § 50.47(a)(1)(i)
- 10 CFR § 50.47(b)(4)
- 10 CFR § 50.54(q)
- 10 CFR § 50.72(a)
- 10 CFR § 50, Appendix E, IV.B, Assessment Actions
- 10 CFR § 50, Appendix E, IV.C, Activation of Emergency Organization

The above regulations are supplemented by various regulatory guidance documents. Three documents of particular relevance to NEI 99-01 are:

NUREG-0654/FEMA-REP-1, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, October 1980. [Refer to Appendix 1, *Emergency Action Level Guidelines for Nuclear Power Plants*]
NUREG-1022, *Event Reporting Guidelines 10 CFR § 50.72 and § 50.73*
Regulatory Guide 1.101, *Emergency Response Planning and Preparedness for Nuclear Power Reactors*

~~The above list is not all-inclusive and it is strongly recommended that scheme developers consult with licensing/regulatory compliance personnel to identify and understand all applicable requirements and guidance. Questions may also be directed to the NEI Emergency Preparedness staff.~~

~~1.2 PERMANENTLY DEFUELED STATION~~

~~NEI 99-01 provides guidance for an emergency classification scheme applicable to a permanently defueled station. This is a station that generated spent fuel under a 10 CFR § 50 license, has permanently ceased operations and will store the spent fuel onsite for an extended period of time. The emergency classification levels applicable to this type of station are consistent with the requirements of 10 CFR § 50 and the guidance in NUREG 0654/FEMA-REP-1.~~

~~In order to relax the emergency plan requirements applicable to an operating station, the owner of a permanently defueled station must demonstrate that no credible event can result in a significant radiological release beyond the site boundary. It is expected that this verification will confirm that the source term and motive force available in the permanently defueled condition are insufficient to warrant classifications of a Site Area Emergency or General Emergency. Therefore, the generic Initiating Conditions (ICs) and Emergency Action Levels (EALs) applicable to a permanently defueled station may result in either a Notification of Unusual Event (NOUE) or an Alert classification.~~

~~The generic ICs and EALs are presented in Appendix C, *Permanently Defueled Station ICs/EALs*.~~

~~1.3~~1.2 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

Selected guidance in NEI 99-01 is applicable to licensees electing to use their 10 CFR 50 emergency plan to fulfill the requirements of 10 CFR 72.32 for a stand-alone ISFSI. The emergency classification levels applicable to an ISFSI are consistent with the requirements of 10 CFR ~~§~~ 50 and the guidance in NUREG 0654/FEMA-REP-1. The initiating conditions germane to a 10 CFR ~~§~~ 72.32 emergency plan (as described in NUREG-1567) are subsumed within the classification scheme for a 10 CFR ~~§~~ 50.47 emergency plan.

The generic ICs and EALs for an ISFSI are presented in Section 8, ISFSI ICs/EALs. IC E-HU1 covers the spectrum of credible natural and man-made events included within the scope of an ISFSI design. This IC is not applicable to installations or facilities that may process and/or repackage spent fuel (e.g., a Monitored Retrievable Storage Facility or an ISFSI at a spent fuel processing facility). In addition, appropriate aspects of IC HU1 and IC HA1 should also be included to address a HOSTILE ACTION directed against an ISFSI.

The analysis of potential onsite and offsite consequences of accidental releases associated with the operation of an ISFSI is contained in NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees.* NUREG-1140 concluded that the postulated worst-case accident involving an ISFSI has insignificant consequences to public health and safety. This evaluation shows that the maximum offsite dose to a member of the public due to an accidental release of radioactive materials would not exceed 1 rem Effective Dose Equivalent.

Regarding the above information, the expectations for an offsite response to an Alert classified under a 10 CFR ~~§~~72.32 emergency plan are generally consistent with those for a Notification of Unusual Event in a 10 CFR ~~§~~50.47 emergency plan (e.g., to provide assistance if requested). Also, the licensee's Emergency Response Organization (ERO) required for 10 CFR ~~§~~72.32 emergency plan is different than that prescribed for a 10 CFR ~~§~~50.47 emergency plan (e.g., no emergency technical support function).

~~1.4~~1.3 **NRC ORDER EA-12-051**

The Fukushima Daiichi accident of March 11, 2012, was the result of a tsunami that exceeded the plant's design basis and flooded the site's emergency electrical power supplies and distribution systems. This caused an extended loss of power that severely compromised the key safety functions of core cooling and containment integrity, and ultimately led to core damage in three reactors. While the loss of power also impaired the spent fuel pool cooling function, sufficient water inventory was maintained in the pools to preclude fuel damage from the loss of cooling.

Following a review of the Fukushima Daiichi accident, the NRC concluded that several measures were necessary to ensure adequate protection of public health and safety under the provisions of the backfit rule, 10 CFR 50.109(a)(4)(ii). Among them was to provide each spent fuel pool with reliable level instrumentation to significantly enhance the ability of key decision-makers to allocate resources effectively following a beyond design basis event. To this end, the NRC issued Order EA-12-051, *Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation*, on March 12, 2012, to all US nuclear plants with an operating license, construction permit, or combined construction and operating license.

NRC Order EA-12-051 states, in part, "All licensees ... shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred." To this end, all licensees must provide:

- A primary and back-up level instrument that will monitor water level from the normal level to the top of the used fuel rack in the pool;
- A display in an area accessible following a severe event; and
- Independent electrical power to each instrument channel and provide an alternate remote power connection capability.

NEI 12-02, *Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,"* provides guidance for complying with NRC Order EA-12-051.

NEI 99-01, Revision 6, includes three EALs that reflect the availability of the enhanced spent fuel pool level instrumentation associated with NRC Order EA-12-051. These EALs are included within ~~existing~~ ICs ~~AA2~~RA2, ~~and new ICs AS2~~ RS2, and ~~AG2~~RG2. ~~Associated EAL notes, bases and developer notes are also provided.~~

~~It is recommended that these EALs be implemented when the enhanced spent fuel pool level instrumentation is available for use.~~

~~The regulatory process that licensees follow to make changes to their emergency plan, including non-scheme changes to EALs, is 10 CFR 50.54(q). In accordance with this regulation, licensees are responsible for evaluating a proposed change and determining whether or not it results in a reduction in the effectiveness of the plan. As a result of the licensee's determination, the licensee will either make the change or submit it to the NRC for prior review and approval in accordance with 10 CFR 50.90.~~

~~1.5 APPLICABILITY TO ADVANCED AND SMALL MODULAR REACTOR DESIGNS~~

~~The guidance in this document primarily addresses commercial nuclear power reactors in the United States, operating or permanently defueled, as of 2012 (so called 1st and 2nd generation plant designs); however, it may be adapted to advanced non-passive designs (often referred to as 3rd generation plant designs) as well. Developers of an emergency classification scheme for an advanced non-passive reactor plant may need to propose deviations from the generic guidance to account for the differences in design parameters and criteria, and operating characteristics and capabilities, between 2nd and 3rd generation plants.~~

~~There are significant design and operating differences between large commercial nuclear power plants (of any generation) and Small Modular Reactors (SMRs) (e.g., differences in source term). For this reason, this document is not applicable to SMRs.~~

2 KEY TERMINOLOGY USED IN ~~NEI 99-01~~DAEC EAL SCHEME

There are several key terms that appear throughout the ~~NEI 99-01~~EAL methodology. These terms are introduced in this section to support understanding of subsequent material. As an aid to the reader, the following table is provided as an overview to illustrate the relationship of the terms to each other.

Emergency Classification Level			
Unusual Event	Alert	SAE	GE
↓	↓	↓	↓
Initiating Condition	Initiating Condition	Initiating Condition	Initiating Condition
↓	↓	↓	↓
Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis
(1) - When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition. This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL~~. —~~.			

2.1 EMERGENCY CLASSIFICATION LEVEL (ECL)

One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are:

Notification of Unusual Event (NOUE)
Alert
Site Area Emergency (SAE)
General Emergency (GE)

2.1.1 Notification of Unusual Event (NOUE)[1]

Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of ~~safety systems~~SAFETY SYSTEMS occurs.

~~[1] This term is sometimes shortened to Unusual Event (UE) or other similar site-specific terminology. The terms Notification of Unusual Event, NOUE and Unusual Event are used interchangeably throughout this document~~

Purpose: The purpose of this classification is to assure that the first step in future response has been carried out, to bring the operations staff to a state of readiness, and to provide systematic handling of unusual event information and decision-making.

2.1.2 Alert

Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.

Purpose: The purpose of this classification is to assure that emergency personnel are readily available to respond if the situation becomes more serious or to perform confirmatory radiation monitoring if required, and provide offsite authorities current information on plant status and parameters.

2.1.3 Site Area Emergency

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.

Purpose: The purpose of the Site Area Emergency declaration is to assure that emergency response centers are staffed, to assure that monitoring teams are dispatched, to assure that personnel required for evacuation of near-site areas are at duty stations if the situation becomes more serious, to provide consultation with offsite authorities, and to provide updates to the public through government authorities.

2.1.4 General Emergency (GE)

Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Purpose: The purpose of the General Emergency declaration is to initiate predetermined protective actions for the public, to provide continuous assessment of information from the licensee and offsite organizational measurements, to initiate additional measures as indicated by actual or potential releases, to provide consultation with offsite authorities, and to provide updates for the public through government authorities.

2.2 INITIATING CONDITION (IC)

An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

Discussion: An IC describes an event or condition, the severity or consequences of which meets the definition of an emergency classification level. An IC can be expressed as a continuous, measurable parameter (e.g., RCS leakage), an event (e.g., an earthquake) or the status of one or more fission product barriers (e.g., loss of the RCS barrier).

Appendix 1 of NUREG-0654 does not contain example Emergency Action Levels (EALs) for each ECL, but rather Initiating Conditions (i.e., plant conditions that indicate that a radiological emergency, or events that could lead to a radiological emergency, has occurred). NUREG-0654 states that the Initiating Conditions form the basis for establishment by a licensee of the specific plant instrumentation readings (as applicable) which, if exceeded, would initiate the emergency classification. Thus, it is the specific instrument readings that would be the EALs.

~~Considerations for the assignment of a particular Initiating Condition to an emergency classification level are discussed in Section 3.~~

2.3 EMERGENCY ACTION LEVEL (EAL)

A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Discussion: EAL statements may utilize a variety of criteria including instrument readings and status indications; observable events; results of calculations and analyses; entry into particular procedures; and the occurrence of natural phenomena.

2.4 FISSION PRODUCT BARRIER THRESHOLD

A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

Discussion: Fission product barrier thresholds represent threats to the defense in depth design concept that precludes the release of radioactive fission products to the environment. This concept relies on multiple physical barriers, any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

Fuel Clad
Reactor Coolant System (RCS)
Containment

Upon determination that one or more fission product barrier thresholds have been exceeded, the combination of barrier loss and/or potential loss thresholds is compared to the fission product barrier IC/EAL criteria to determine the appropriate ECL.

In some accident sequences, the ICs and EALs presented in the Abnormal Radiation Levels/ Radiological Effluent (AR) Recognition Category will be exceeded at the same time, or shortly after, the loss of one or more fission product barriers. This redundancy is intentional as the former ICs address radioactivity releases that result in certain offsite doses from whatever cause, including events that might not be fully encompassed by fission product barriers (e.g., spent fuel pool accidents, design containment leakage following a LOCA, etc.).

3 DESIGN OF THE ~~NEI 99-01~~ DAEC EMERGENCY CLASSIFICATION SCHEME

3.1 ASSIGNMENT OF EMERGENCY CLASSIFICATION LEVELS (ECLS)

An effective emergency classification scheme must incorporate a realistic and accurate assessment of risk, both to plant workers and the public. There are obvious health and safety risks in underestimating the potential or actual threat from an event or condition; however, there are also risks in overestimating the threat as well (e.g., harm that may occur during an evacuation). The ~~NEI 99-01~~DAEC emergency classification scheme attempts to strike an appropriate balance between reasonably anticipated event or condition consequences, potential accident trajectories, and risk avoidance or minimization.

There are a range of "non-emergency events" reported to the US Nuclear Regulatory Commission (NRC) staff in accordance with the requirements of 10 CFR ~~§~~50.72. Guidance concerning these reporting requirements, and example events, are provided in NUREG-1022. Certain events reportable under the provisions of 10 CFR ~~§~~50.72 may also require the declaration of an emergency.

In order to align each Initiating Conditions (IC) with the appropriate ECL, it was necessary to determine the attributes of each ECL. The goal of this process is to answer the question, "What events or conditions should be placed under each ECL?" The following sources provided information and context for the development of ECL attributes.

- Assessments of the effects and consequences of different types of events and conditions
- ~~Typical~~ DAEC abnormal and emergency operating procedure setpoints and transition criteria
- ~~Typical~~ DAEC Technical Specification limits and controls
- ~~Radiological Effluent Technical Specifications (RETS)/~~Offsite Dose ~~Calculation~~ Assessment Manual (ODAM) radiological release limits
- Review of selected Updated Final Safety Analysis Report (UFSAR) accident analyses
- Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs)
- NUREG 0654, Appendix 1, *Emergency Action Level Guidelines for Nuclear Power Plants*
- Industry Operating Experience
- Input from ~~industry~~ DAEC subject matter experts ~~and NRC staff members~~

The following ECL attributes ~~were~~ are ~~created~~ used ~~by the Revision 6 Preparation Team~~ to aid in the development of ICs and Emergency Action Levels (EALs). ~~The team decided to include the attributes in this revision since they~~The attributes may be useful in briefing and training settings (e.g., helping an Emergency Director understand why a particular condition is classified as an Alert). ~~It should be stressed that developers not attempt to redefine these attributes or apply them in any fashion that would change the generic guidance contained in this document[1].~~

~~[1] The use of ECL attributes is at the discretion of a licensee and is not a requirement of the NRC. If a licensee chooses in incorporate the ECL attributes into their scheme basis document, it must be very clear that the NRC staff has not endorsed their acceptability or application for any purpose. In particular, the staff does not consider the~~

~~attribute statements to supersede the established ECL definitions. As a result, the use of the attributes as a basis for justifying EAL changes is unacceptable.~~

~~The attributes of each ECL are presented below.~~

3.1.1 Notification of Unusual Event (NOUE)

A Notification of Unusual Event, as defined in section 2.1.1, includes but is not limited to an event or condition that involves:

(A) A precursor to a more significant event or condition.

(B) A minor loss of control of radioactive materials or the ability to control radiation levels within the plant.

(C) A consequence otherwise significant enough to warrant notification to local, State and Federal authorities.

3.1.2 Alert

An Alert, as defined in section 2.1.2, includes but is not limited to an event or condition that involves:

(A) A loss or potential loss of either the fuel clad or Reactor Coolant System (RCS) fission product barrier.

(B) An event or condition that significantly reduces the margin to a loss or potential loss of the fuel clad or RCS fission product barrier.

(C) A significant loss of control of radioactive materials resulting in an inability to control radiation levels within the plant, or a release of radioactive materials to the environment that could result in doses greater than 1% of an EPA PAG at or beyond the site boundary.

(D) A HOSTILE ACTION occurring within the OWNER CONTROLLED AREA, including those directed at an Independent Spent Fuel Storage Installation (ISFSI).

3.1.3 Site Area Emergency (SAE)

A Site Area Emergency, as defined in section 2.1.3, includes but is not limited to an event or condition that involves:

(A) A loss or potential loss of any two fission product barriers - fuel clad, RCS and/or containment.

(B) A precursor event or condition that may lead to the loss or potential loss of multiple fission product barriers within a relatively short period of time. Precursor events and conditions of this type include those that challenge the monitoring and/or control of multiple ~~safety systems~~SAFETY SYSTEMS.

(C) A release of radioactive materials to the environment that could result in doses greater than 10% of an EPA PAG at or beyond the site boundary.

(D) A HOSTILE ACTION occurring within the plant PROTECTED AREA.

3.1.4 General Emergency (GE)

A General Emergency, as defined in section 2.1.4, includes but is not limited to an event or condition that involves:

(A) Loss of any two fission product barriers AND loss or potential loss of the third barrier - fuel clad, RCS and/or containment.

(B) A precursor event or condition that, unmitigated, may lead to a loss of all three fission product barriers. Precursor events and conditions of this type include those that lead directly to core damage and loss of containment integrity.

(C) A release of radioactive materials to the environment that could result in doses greater than an EPA PAG at or beyond the site boundary.

(D) A HOSTILE ACTION resulting in the loss of key safety functions (reactivity control, core cooling/RPV water level or RCS heat removal) or damage to spent fuel.

3.1.5

Risk-Informed Insights

Emergency preparedness is a defense-in-depth measure that is independent of the assessed risk from any particular accident sequence; however, the development of an effective emergency classification scheme can benefit from a review of risk-based assessment results. To that end, the development and assignment of certain ICs and EALs also considered insights from several site-specific probabilistic safety assessments ~~(PSA - also known as probabilistic risk assessment, PRA)~~. Some generic insights from this review included:

1. Accident sequences involving a prolonged loss of all AC power are significant contributors to core damage frequency at many ~~Pressurized Water Reactors (PWRs) and~~ Boiling Water Reactors (BWRs). For this reason, a loss of all AC power for greater than 15 minutes, with the plant at or above Hot Shutdown, was assigned an ECL of Site Area Emergency. Precursor events to a loss of all AC power were also included as an Unusual Event and an Alert.

   A station blackout coping analyses performed in response to 10 CFR ~~§~~ 50.63 and Regulatory Guide 1.155, *Station Blackout,* may be used to determine a time-based criterion to demarcate between a Site Area Emergency and a General Emergency. The time dimension is critical to a properly anticipatory emergency declaration since the goal is to maximize the time available for State and local officials to develop and implement offsite protective actions.

2. For severe core damage events, uncertainties exist in phenomena important to accident progressions leading to containment failure. Because of these uncertainties, predicting the status of containment integrity may be difficult under severe accident conditions. This is why maintaining containment integrity alone following sequences leading to severe core damage is an insufficient basis for not escalating to a General Emergency.

3. PSAs indicated that leading contributors to latent fatalities were sequences involving a containment bypass, a large Loss of Coolant Accident (LOCA) with early containment failure, a Station Blackout lasting longer than the ~~site-specific~~DAEC coping period, and a reactor coolant pump seal failure. The generic EAL methodology needs to be sufficiently rigorous to address these sequences in a timely fashion.

3.2 TYPES OF INITIATING CONDITIONS AND EMERGENCY ACTION LEVELS

The NEI 99-01 methodology makes use of symptom-based, barrier-based and event-based ICs and EALs. Each type is discussed below.

Symptom-based ICs and EALs are parameters or conditions that are measurable over some range using plant instrumentation (e.g., core temperature, reactor coolant level, radiological effluent, etc.). When one or more of these parameters or conditions are off-normal, reactor operators will implement procedures to identify the probable cause(s) and take corrective action.

Fission product barrier-based ICs and EALs are the subset of symptom-based EALs that refer specifically to the level of challenge to the principal barriers against the release of radioactive material from the reactor core to the environment. These barriers are the fuel cladding, the reactor coolant system pressure boundary, and the containment. The barrier-based ICs and EALs consider the level of challenge to each individual barrier - potentially lost and lost - and the total number of barriers under challenge.

Event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. These include the failure of an automatic reactor scram~~/trip~~ to shut down the reactor, natural phenomena (e.g., an earthquake), or man-made hazards such as a toxic gas release.

~~3.3 NSSS DESIGN DIFFERENCES~~

~~The NEI 99-01 emergency classification scheme accounts for the design differences between PWRs and BWRs by specifying EALs unique to each type of Nuclear Steam Supply System (NSSS). There are also significant design differences among PWR NSSSs; therefore, guidance is provided to aid in the development of EALs appropriate to different PWR NSSS types. Where necessary, development guidance also addresses unique considerations for advanced non-passive reactor designs such as the Advanced Boiling Water Reactor (ABWR), the Advanced Pressurized Water Reactor (APWR) and the Evolutionary Power Reactor (EPR).~~

~~Developers will need to consider the relevant aspects of their plant's design and operating characteristics when converting the generic guidance of this document into a site-specific classification scheme. The goal is to maintain as much fidelity as possible to the intent of generic ICs and EALs within the constraints imposed by the plant design and operating characteristics. To this end, developers of a scheme for an advanced non-passive reactor may need to add, modify or delete some information contained in this document; these changes will be reviewed for acceptability by the NRC as part of the scheme approval process.~~

~~The guidance in NEI 99-01 is not applicable to advanced passive light water reactor designs. An Emergency Classification Scheme for this type of plant should be developed in accordance with NEI 07-01,~~ ~~*Methodology for Development of Emergency Action Levels, Advanced Passive Light Water Reactors.*~~

~~3.4~~3.3 DAEC-SPECIFIC ORGANIZATION AND PRESENTATION OF GENERIC INFORMATION

The scheme's generic information is organized by Recognition Category in the following order.

~~A~~R - Abnormal Radiation Levels / Radiological Effluent – Section 6
C - Cold Shutdown / Refueling System Malfunction – Section 7
E - Independent Spent Fuel Storage Installation (ISFSI) – Section 8
F - Fission Product Barrier – Section 9
H - Hazards and Other Conditions Affecting Plant Safety – Section 10
S - System Malfunction – Section 11
~~PD - Permanently Defueled Station – Appendix C~~

Each Recognition Category section contains a matrix showing the ICs and their associated emergency classification levels.

The following information and guidance is provided for each IC:

ECL – the assigned emergency classification level for the IC.

Initiating Condition – provides a summary description of the emergency event or condition.

Operating Mode Applicability – Lists the modes during which the IC and associated EAL(s) are applicable (i.e., are to be used to classify events or conditions).

~~Example~~ Emergency Action Level(s) – Provides examples of reports and indications that are considered to meet the intent of the IC. ~~Developers should address each example EAL. If the generic approach to the development of an example EAL cannot be used (e.g., an assumed instrumentation range is not available at the plant), the developer should attempt to specify an alternate means for identifying entry into the IC.~~

For Recognition Category F, the fission product barrier thresholds are presented in tables ~~applicable to BWRs and PWRs,~~ and arranged by fission product barrier and the degree of barrier challenge (i.e., potential loss or loss). This presentation method shows the synergism among the thresholds, and supports accurate assessments.

Basis – Provides background information that explains the intent and application of the IC and EALs. In some cases, the basis also includes relevant source information and references.

~~**Developer Notes** – Information that supports the development of the site-specific ICs and EALs. This may include clarifications, references, examples, instructions for calculations, etc. Developer notes should not be included in the site's emergency classification scheme basis document. Developers may elect to include information resulting from a developer note action in a basis section.~~

~~**ECL Assignment Attributes** – Located within the Developer Notes section, specifies the attribute used for assigning the IC to a given ECL.~~

~~3.5~~3.4 IC AND EAL MODE APPLICABILITY

The ~~NEI 99-01~~ DAEC emergency classification scheme was developed recognizing that the applicability of ICs and EALs will vary with plant mode. For example, some symptom-based ICs and EALs can be assessed only during the power operations, startup, or hot standby/shutdown modes of operation when all fission product barriers are in place, and plant instrumentation and ~~safety systems~~SAFETY SYSTEMS are fully operational. In the cold shutdown and refueling modes, different symptom-based ICs and EALs will come into play to reflect the opening of systems for routine maintenance, the unavailability of some ~~safety system~~SAFETY SYSTEM components and the use of alternate instrumentation.

The following table shows which Recognition Categories are applicable in each plant mode. The ICs and EALs for a given Recognition Category are applicable in the indicated modes.

MODE APPLICABILITY MATRIX

	Recognition Category					
Mode	**~~A~~R**	**C**	**E**	**F**	**H**	**S**
Power Operations	X		X	X	X	X
Startup	X		X	X	X	X
~~Hot Standby~~						
Hot Shutdown	X		X	X	X	X
Cold Shutdown	X	X	X		X	
Refueling	X	X	X		X	
Defueled	X	X	X		X	
~~Permanently Defueled~~						

~~Typical BWR~~ DAEC Operating Modes

Power Operations (1):	Mode Switch in Run
Startup (2):	Mode Switch in Startup/Hot Standby or Refuel (with all vessel head closure bolts fully tensioned)
Hot Shutdown (3):	Mode Switch in Shutdown, Average Reactor Coolant Temperature > ~~200~~ 212 °F (with all vessel head closure bolts fully tensioned)
Cold Shutdown (4):	Mode Switch in Shutdown, Average Reactor Coolant Temperature ≤ ~~200~~ 212 °F (with all vessel head closure bolts fully tensioned)
Refueling (5):	Mode Switch in Shutdown or Refuel~~, and~~ (with one or more vessel head closure bolts less than fully tensioned)~~.~~

~~Typical PWR OPERATING MODES~~

~~Power Operations (1):~~	~~Reactor Power > 5%, Keff ≥ 0.99~~
~~Startup (2):~~	~~Reactor Power ≤ 5%, Keff ≥ 0.99~~
~~Hot Standby (3):~~	~~RCS ≥ 350 °F, Keff < 0.99~~
~~Hot Shutdown (4):~~	~~200 °F < RCS < 350 °F, Keff < 0.99~~
~~Cold Shutdown (5):~~	~~RCS < 200 °F, Keff < 0.99~~
~~Refueling (6):~~	~~One or more vessel head closure bolts less than fully tensioned~~

~~Developers will need to incorporate the mode criteria from unit-specific Technical Specifications into their emergency classification scheme. In addition, the scheme must also include the following mode designation specific to NEI 99-01:~~

~~Defueled (None):~~	~~All fuel removed from the reactor vessel (i.e., full core offload during refueling or extended outage).~~

4 ~~SITE-SPECIFIC SCHEME DEVELOPMENT GUIDANCE~~DEVELOPMENT OF THE DAEC EMERGENCY CLASSIFICATION SCHEME

~~This section provides detailed guidance for developing a site-specific emergency classification scheme. Conceptually, the approach discussed here mirrors the approach used to prepare emergency operating procedures – generic material prepared by reactor vendor owners groups is converted by each nuclear power plant into site-specific emergency operating procedures. Likewise, the emergency classification scheme developer will use the generic guidance in NEI 99-01 to prepare a site-specific emergency classification scheme and the associated basis document.~~

~~It is important that the NEI 99-01 emergency classification scheme be implemented as an integrated package. Selected use of portions of this guidance is strongly discouraged as it will lead to an inconsistent or incomplete emergency classification scheme that will likely not receive the necessary regulatory approval.~~

4.1 GENERAL ~~IMPLEMENTATION GUIDANCE~~DEVELOPMENT PROCESS

~~The guidance in NEI 99-01 is not intended to be applied to plants "as-is"; however, developers should attempt to keep their site-specific schemes as close to the generic guidance as possible. The goal is to meet the intent of the generic Initiating Conditions (ICs) and Emergency Action Levels (EALs) within the context of site-specific characteristics – locale, plant design, operating features, terminology, etc. Meeting this goal will result in a shorter and less cumbersome NRC review and approval process, closer alignment with the schemes of other nuclear power plant sites and better positioning to adopt future industry-wide scheme enhancements.~~

~~When properly developed, the~~The DAEC ICs and EALs ~~should~~were developed to be unambiguous and readily assessable.

~~As discussed in Section 3, the generic guidance includes ICs and example EALs. It is the intent of this guidance that both be included in site-specific documents as each serves a specific purpose.~~ The IC is the fundamental event or condition requiring a declaration. The EAL(s) is the pre-determined threshold that defines when the IC is met. ~~If some feature of the plant location or design is not compatible with a generic IC or EAL, efforts should be made to identify an alternate IC or EAL.~~

~~If an IC or EAL includes an explicit reference to a mode dependent technical specification limit that is not applicable to the plant, then that IC and/or EAL need not be included in the site-specific scheme. In these cases, developers must provide adequate documentation to justify why the IC and/or EAL were not incorporated (i.e., sufficient detail to allow a third party to understand the decision not to incorporate the generic guidance).~~

Useful acronyms and abbreviations associated with the ~~NEI 99-01~~DAEC emergency classification scheme are presented in Appendix A, Acronyms and Abbreviations. ~~Site-specific entries may be added if necessary.~~

Many words or terms used in the ~~NEI 99-01~~DAEC emergency classification scheme have scheme-specific definitions. These words and terms are identified by being set in all capital letters (i.e., ALL CAPS). The definitions are presented in Appendix B, Definitions.

~~Below are examples of acceptable modifications to the generic guidance. These may be incorporated depending upon site developer and user preferences.~~

~~The ICs within a Recognition Category may be placed in reverse order for presentation purposes (e.g., start with a General Emergency at the left/top of a user aid, followed by Site Area Emergency, Alert and NOUE).~~
~~The Initiating Condition numbering may be changed.~~
~~The first letter of a Recognition Category designation may be changed, as follows, provided the change is carried through for all of the associated IC identifiers.~~

- ~~R may be used in lieu of A~~
- ~~M may be used in lieu of S~~

~~For example, the Abnormal Radiation Levels / Radiological Effluent category designator "A" (for Abnormal) may be changed to "R" (for Radiation). This means that the associated ICs would be changed to RU1, RU2, RA1, etc.~~

~~The ICs and EALs from Recognition Categories S and C may be incorporated into a common presentation method (e.g., one table) provided that all related notes and mode applicability requirements are maintained.~~
~~The ICs and EALs for Emergency Director judgment and security related events may be placed under separate Recognition Categories.~~
~~The terms EAL and threshold may be used interchangeably.~~

~~The material in the Developer Notes section is included to assist developers with crafting correct IC and EAL statements. This material is not required to be in the final emergency classification scheme basis document.~~

4.2 CRITICAL CHARACTERISTICS

~~As discussed above, developers are encouraged to keep their site-specific schemes as close to the generic guidance as possible.~~ When crafting the scheme, ~~developers should satisfy themselves~~DAEC ensured that certain critical characteristics have been met. These critical characteristics are listed below.

- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are consistent with industry guidance; while the actual wording may be different, the classification intent is maintained. With respect to Recognition Category F, ~~a site-specific scheme must~~DAEC includes a ~~some type of~~ user-aid to facilitate timely and accurate classification of fission product barrier losses and/or potential losses. The user-aid logic ~~must be~~is consistent with the classification logic presented in Section 9.

- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are technically complete and accurate (i.e., they contain the information necessary to make a correct classification).
- EAL statements use objective criteria and observable values.
- ICs, EALs, Operating Mode Applicability and Note statements and formatting consider human factors and are user-friendly.
- The scheme facilitates upgrading and downgrading of the emergency classification where necessary.
- The scheme facilitates classification of multiple concurrent events or conditions.

4.3 INSTRUMENTATION USED FOR EALS

~~Instrumentation referenced in EAL statements should include that described in the emergency plan section which addresses 10 CFR 50.47(b)(8) and (9) and/or Chapter 7 of the FSAR. Instrumentation used for EALs need not be safety-related, addressed by a Technical Specification or ODCM/RETS control requirement, nor powered from an emergency power source; however, EAL developers should strive to~~ DAEC incorporated instrumentation that is reliable and routinely maintained in accordance with site programs and procedures. Alarms referenced in EAL statements ~~should be~~are those that are the most operationally significant for the described event or condition.

~~Scheme developers should ensure that specified values used as~~ EAL setpoints are within the calibrated range of the referenced instrumentation, and consider any automatic instrumentation functions that may impact accurate EAL assessment. In addition, EAL setpoint values ~~should~~ do not use terms such as "off-scale low" or "off-scale high" since that type of reading may not be readily differentiated from an instrument failure. ~~Findings and violations related to EAL instrumentation issues may be located on the NRC website.~~

~~4.4 PRESENTATION OF SCHEME INFORMATION TO USERS~~

~~The US Nuclear Regulatory Commission (NRC) expects licensees to establish and maintain the capability to assess, classify and declare an emergency condition promptly within 15 minutes after the availability of indications to plant operators that an emergency action level has been, or may be, exceeded. When writing an emergency classification procedure and creating related user aids, the developer must determine the presentation method(s) that best supports the end users by facilitating accurate and timely emergency classification. To this end, developers should consider the following points.~~

- ~~The first users of an emergency classification procedure are the operators in the Control Room. During the allowable classification time period, they may have responsibility to perform other critical tasks, and will likely have minimal assistance in making a classification assessment.~~
- ~~As an emergency situation evolves, members of the Control Room staff are likely to be the first personnel to notice a change in plant conditions. They can assess the changed conditions and, when warranted, recommend a different emergency classification level to the Technical Support Center (TSC) and/or Emergency Operations Facility (EOF).~~
- ~~Emergency Directors in the TSC and/or EOF will have more opportunity to focus on making an emergency classification, and will probably have advisors from Operations available to help them.~~

~~Emergency classification scheme information for end users should be presented in a manner with which licensed operators are most comfortable. Developers will need to work closely with representatives from the Operations and Operations Training Departments to develop readily usable and easily understood classification tools (e.g., a procedure and related user aids). If necessary, an alternate method for presenting~~

~~emergency classification scheme information may be developed for use by Emergency Directors and/or Offsite Response Organization personnel.~~

~~A wallboard is an acceptable presentation method provided that it contains all the information necessary to make a correct emergency classification. This information includes the ICs, Operating Mode Applicability criteria, EALs and Notes. Notes may be kept with each applicable EAL or moved to a common area and referenced; a reference to a Note is acceptable as long as the information is adequately captured on the wallboard and pointed to by each applicable EAL[1]. Basis information need not be included on a wallboard but it should be readily available to emergency classification decision-makers.~~

~~In some cases, it may be advantageous to develop two wallboards – one for use during power operations, startup and hot conditions, and another for cold shutdown and refueling conditions.~~

~~Alternative presentation methods for the Recognition Category F ICs and fission product barrier thresholds are acceptable and include flow charts, block diagrams, and checklist-type tables. Developers must ensure that the site-specific method addresses all possible threshold combinations and classification outcomes shown in the BWR or PWR EAL fission product barrier tables. The NRC staff considers the presentation method of the Recognition Category F information to be an important user aid and may request a change to a particular proposed method if, among other reasons, the change is necessary to promote consistency across the industry.~~

~~4.5 INTEGRATION OF ICs/EALs WITH PLANT PROCEDURES~~

~~A rigorous integration of IC and EAL references into plant operating procedures is not recommended. This approach would greatly increase the administrative controls and workload for maintaining procedures. On the other hand, performance challenges may occur if recognition of meeting an IC or EAL is based solely on the memory of a licensed operator or an Emergency Director, especially during periods of high stress.~~

~~Developers should consider placing appropriate visual cues (e.g., a step, note, caution, etc.) in plant procedures alerting the reader/user to consult the site emergency classification procedure. Visual cues could be placed in emergency operating procedures, abnormal operating procedures, alarm response procedures, and normal operating procedures that apply to cold shutdown and refueling modes. As an example, a step, note or caution could be placed at the beginning of an RCS leak abnormal operating procedure that reminds the reader that an emergency classification assessment should be performed.~~

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[1] ~~Where appropriate, the Notes shown in the generic guidance typically include the event/condition ECL and the duration time specified in the EAL. If developers prefer to have several ICs reference a common NOTE on a wallboard display, it is acceptable to remove the ECL and time criterion and use a generic statement. For example, a common NOTE could read "The Emergency Director should declare the emergency promptly upon determining that the applicable EAL time has been exceeded, or will likely be exceeded."~~

~~4.6 BASIS DOCUMENT~~

~~A basis document is an integral part of an emergency classification scheme. The material in this document supports proper emergency classification decision-making by providing informing background and development information in a readily accessible format. It can be referred to in training situations and when making an actual emergency classification, if necessary. The document is also useful for establishing configuration management controls for EP-related equipment and explaining an emergency classification to offsite authorities. The content of the basis document should include, at a minimum, the following:~~

- ~~A site-specific Mode Applicability Matrix and description of operating modes, similar to that presented in section 3.5.~~
- ~~A discussion of the emergency classification and declaration process reflecting the material presented in Section 5. This material may be edited as needed to align with site-specific emergency plan and implementing procedure requirements.~~
- ~~Each Initiating Condition along with the associated EALs or fission product barrier thresholds, Operating Mode Applicability, Notes and Basis information.~~
- ~~A listing of acronyms and defined terms, similar to that presented in Appendices A and B, respectively. This material may be edited as needed to align with site-specific characteristics.~~
- ~~Any site-specific background or technical appendices that the developers believe would be useful in explaining or using elements of the emergency classification scheme.~~

~~A Basis section should not contain information that could modify the meaning or intent of the associated IC or EAL. Such information should be incorporated within the IC or EAL statements, or as an EAL Note. Information in the Basis should only clarify and inform decision-making for an emergency classification.~~

~~Basis information should be readily available to be referenced, if necessary, by the Emergency Director. For example, a copy of the basis document could be maintained in the appropriate emergency response facilities.~~

~~Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency Director refers to it during an event), the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q).~~

4.~~7~~4.4 EAL/THRESHOLD REFERENCES TO AOP AND EOP SETPOINTS/CRITERIA

~~As reflected in the generic guidance,~~Some of the criteria/values used in several EALs and fission product barrier thresholds ~~may be~~ are drawn from ~~a plant's~~DAEC AOPs and EOPs. This approach is intended to maintain good alignment between operational diagnoses and emergency classification assessments. ~~Developers should verify that a~~Appropriate administrative controls are in place to ensure that a subsequent change to an AOP or EOP is screened to determine if an evaluation pursuant to 10 CFR 50.54(q) is required.

~~4.8 DEVELOPER AND USER FEEDBACK~~

~~Questions or comments concerning the material in this document may be directed to the NEI Emergency Preparedness staff, NEI EAL task force members or submitted to the Emergency Preparedness Frequently Asked Questions process.~~

5 GUIDANCE ON ~~MAKING EMERGENCY CLASSIFICATIONS~~USING THE DAEC EALS

5.1 GENERAL CONSIDERATIONS

When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL.

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, Interim Staff Guidance, *Emergency Planning for Nuclear Power Plants.*

All emergency classification assessments should be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, validation could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel. The validation of indications should be completed in a manner that supports timely emergency declaration.

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. If an ongoing radiological release is detected and the release start time is unknown, it should be assumed that the release duration specified in the IC/EAL has been exceeded, absent data to the contrary.

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that 1) the activity proceeds as planned and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10 ~~§~~ CFR 50.72.

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, RCS leak rate calculation, etc.); the EAL and/or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. ~~The NEI 99-01~~This scheme provides the Emergency Director with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Emergency Director will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated into the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

5.2

CLASSIFICATION METHODOLOGY

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL(s) must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, then the IC is considered met and the associated ECL is declared in accordance with plant procedures.

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01.

5.3 CLASSIFICATION OF MULTIPLE EVENTS AND CONDITIONS

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. ~~For example:~~

> ~~If an Alert EAL and a Site Area Emergency EAL are met, whether at one unit or at two different units, a Site Area Emergency should be declared.~~

~~T~~Additionally, there is no "additive" effect from multiple EALs meeting the same ECL. For example:

> If two Alert EALs are met~~, whether at one unit or at two different units,~~ an Alert should be declared.

Related guidance concerning classification of rapidly escalating events or conditions is provided in Regulatory Issue Summary (RIS) 2007-02, *Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events*.

5.4 CONSIDERATION OF MODE CHANGES DURING CLASSIFICATION

The mode in effect at the time that an event or condition occurred, and prior to any plant or operator response, is the mode that determines whether or not an IC is applicable. If an event or condition occurs, and results in a mode change before the emergency is declared, the emergency classification level is still based on the mode that existed at the time that the event or condition was initiated (and not when it was declared). Once a different mode is reached, any new event or condition, not related to the original event or condition, requiring emergency classification should be evaluated against the ICs and EALs applicable to the operating mode at the time of the new event or condition.

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Hot Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

5.5 CLASSIFICATION OF IMMINENT CONDITIONS

Although EALs provide specific thresholds, the Emergency Director must remain alert to events or conditions that could lead to meeting or exceeding an EAL within a relatively short period of time (i.e., a change in the ECL is IMMINENT). If, in the judgment of the Emergency Director, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all emergency classification levels, this approach is particularly important at the higher emergency classification levels since it provides additional time for implementation of protective measures.

5.6 EMERGENCY CLASSIFICATION LEVEL UPGRADING AND DOWNGRADING

An ECL may be downgraded when the event or condition that meets the highest IC and EAL no longer exists, and other site-specific downgrading requirements are met. If downgrading the ECL is deemed appropriate, the new ECL would then be based on a lower applicable IC(s) and EAL(s). The ECL may also simply be terminated.

The following approach to downgrading or terminating an ECL is recommended.

ECL	Action When Condition No Longer Exists
Unusual Event	Terminate the emergency in accordance with plant procedures.
Alert	Downgrade or terminate the emergency in accordance with plant procedures.
Site Area Emergency with no long-term plant damage	Downgrade or terminate the emergency in accordance with plant procedures.
Site Area Emergency with long-term plant damage	Terminate the emergency and enter recovery in accordance with plant procedures.
General Emergency	Terminate the emergency and enter recovery in accordance with plant procedures.

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02.

5.7

CLASSIFICATION OF SHORT-LIVED EVENTS

As discussed in Section 3.2, event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. By their nature, some of these events may be short-lived and, thus, over before the emergency classification assessment can be completed. If an event occurs that meets or exceeds an EAL, the associated ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include a failure of the reactor protection system to automatically scram~~/trip~~ the reactor followed by a successful manual scram~~/trip~~ or an earthquake.

5.8 CLASSIFICATION OF TRANSIENT CONDITIONS

Many of the ICs and/or EALs contained in this document employ time-based criteria. These criteria will require that the IC/EAL conditions be present for a defined period of time before an emergency declaration is warranted. In cases where no time-based criterion is specified, it is recognized that some transient conditions may cause an EAL to be met for a brief period of time (e.g., a few seconds to a few minutes). The following guidance should be applied to the classification of these conditions.

EAL momentarily met during expected plant response - In instances where an EAL is briefly met during an expected (normal) plant response, an emergency declaration is not warranted provided that associated systems and components are operating as expected, and operator actions are performed in accordance with procedures.

EAL momentarily met but the condition is corrected prior to an emergency declaration – If an operator takes prompt manual action to address a condition, and the action is successful in correcting the condition prior to the emergency declaration, then the applicable EAL is not considered met and the associated emergency declaration is not required. For illustrative purposes, consider the following example.

> An ATWS occurs and the auxiliary feedwater system fails to automatically start. Steam generator levels rapidly decrease and the plant enters an inadequate RCS heat removal condition (a potential loss of both the fuel clad and RCS barriers). If an operator manually starts the auxiliary feedwater system in accordance with an EOP step and clears the inadequate RCS heat removal condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event; emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations where an operator is able to take a successful corrective action prior to the Emergency Director completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

5.9 AFTER-THE-FACT DISCOVERY OF AN EMERGENCY EVENT OR CONDITION

In some cases, an EAL may be met but the emergency classification was not made at the time of the event or condition. This situation can occur when personnel discover that an event or condition existed which met an EAL, but no emergency was declared, and the event or condition no longer exists at the time of discovery. This may be due to the event or condition not being recognized at the time or an error that was made in the emergency classification process.

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR ~~§~~50.72 within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

5.10 RETRACTION OF AN EMERGENCY DECLARATION

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022.

6 ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

~~Table AR-1: Recognition Category "AR" Initiating Condition Matrix~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
~~AU1RU1~~ ~~Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document)ODCM limits for 60 minutes or longer.~~ ~~*Op. Modes: All*~~	~~AA1RA1~~ ~~Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.~~ ~~*Op. Modes: All*~~	~~AS1RS1~~ ~~Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.~~ ~~*Op. Modes: All*~~	~~AG1RG1~~ ~~Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.~~ ~~*Op. Modes: All*~~
~~AU2RU2~~ ~~UNPLANNED loss of water level above irradiated fuel.~~ ~~*Op. Modes: All*~~	~~AA2RA2~~ ~~Significant lowering of water level above, or damage to, irradiated fuel.~~ ~~*Op. Modes: All*~~	~~AS2RS2~~ ~~Spent fuel pool level at (site-specific Level 3 description)40 ft 8 in (Level 3).~~ ~~*Op. Modes: All*~~	~~AG2RG2~~ ~~Spent fuel pool level cannot be restored to at least (site-specific Level 3 description)40 ft 8 in (Level 3) for 60 minutes or longer.~~ ~~*Op. Modes: All*~~
	~~AA3RA3~~ ~~Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.~~ ~~*Op. Modes: All*~~		

~~AU1~~RU1

ECL: Notification of Unusual Event

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the ~~(site-specific effluent release controlling document)~~ODAM limits for 60 minutes or longer.

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example Emergency Action Levels:** (1 or 2 or 3)~~

Notes:

- The Emergency Director should declare the ~~Unusual Event~~event promptly upon determining that the applicable time~~60 minutes~~ has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit~~60 minutes~~.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

~~1~~RU1.1 Reading on **ANY** Table R-1 effluent radiation monitor greater than column "NOUE" for ~~2 times the (site-specific effluent release controlling document) limits for~~ 60 minutes or longer:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
Gaseous	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
Liquid	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
Liquid	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

RU1.2~~2~~ ~~(site-specific monitor list and threshold values corresponding to 2 times the controlling document limits)~~

Reading on ~~ANY~~ANY effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

RU1.3 Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ~~(site-specific effluent release controlling document)~~ODAM limits for 60 minutes or longer.

Definitions:

None

Basis:

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

~~Nuclear power plants~~DAEC incorporates design features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, and to control and monitor intentional releases. The occurrence of an extended, uncontrolled radioactive release to the environment is indicative of degradation in these features and/or controls.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

EAL RU1.1 - This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.

EAL RU1.2 - This EAL addresses radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. This EAL will typically be associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

EAL RU1.3 - This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyseis or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).

Escalation of the emergency classification level would be via IC ~~AA1~~RA1.

~~**Developer Notes:**~~

~~The "site-specific effluent release controlling document" is the Radiological Effluent Technical Specifications (RETS) or, for plants that have implemented Generic Letter 89-01[1], the~~

~~[1] *Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program*~~

~~Offsite Dose Calculation Manual (ODCM). These documents implement regulations related to effluent controls (e.g., 10 CFR Part 20 and 10 CFR Part 50, Appendix I). As appropriate, the RETS or ODCM methodology should be used for establishing the monitor thresholds for this IC.~~

~~Listed monitors should include the effluent monitors described in the RETS or ODCM.~~

~~Developers may also consider including installed monitors associated with other potential effluent pathways that are not described in the RETS or ODCM[1,2]. If included, EAL values for these monitors should be determined using the most applicable dose/release limits presented in the RETS or ODCM. It is recognized that a calculated EAL value may be below what the monitor can read; in that case, the monitor does not need to be included in the list. Also, some monitors may not be governed by Technical Specifications or other license-related related requirements; therefore, it is important that the associated EAL and basis section clearly identify any limitations on the use or availability of these monitors.~~

~~Some sites may find it advantageous to address gaseous and liquid releases with separate EALs.~~

~~Radiation monitor readings should reflect values that correspond to a radiological release exceeding 2 times a release control limit. The controlling document typically describes methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values. In cases where a methodology is not adequately defined, developers should determine values consistent with effluent control regulations (e.g., 10 CFR Part 20 and 10 CFR Part 50 Appendix I) and related guidance.~~

~~For EAL #2 – Values in this EAL should be 2 times the setpoint established by the radioactivity discharge permit to warn of a release that is not in compliance with the specified limits. Indexing the value in this manner ensures consistency between the EAL and the setpoint established by a specific discharge permit.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the~~

~~[1] This includes consideration of the effluent monitors described in the site emergency plan section(s) which address the requirements of 10 CFR 50.47(b)(8) and (9).~~

~~[2] Developers should keep in mind the requirements of 10 CFR 50.54(q) and the guidance provided by INPO related to emergency response equipment when considering the addition of other effluent monitors.~~

~~scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.1.B~~

~~AU2~~RU2

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

~~1~~RU2.1 a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:

- Report to control room (visual observation)
- Fuel pool level indication (LI-3413) ~~LESS THAN~~less than 36 feet and lowering
- WR GEMAC Floodup indication (LI-4541) coming on scale~~(site-specific level indications).~~

AND

b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors.

- ~~(site-specific list of area radiation monitors)~~ Spent Fuel Pool Area, RI-9178
- North Refuel Floor, RI-9163
- New Fuel Vault Area, RI-9153
- South Refuel Floor, RI-9164
- NW Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REFUELING PATHWAY: The reactor refueling cavity, spent fuel pool and fuel transfer canal.

Basis:

This IC addresses a decrease in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level decrease will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations ~~(if available)~~. A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an UNPLANNED loss of water level.

During preparation for reactor cavity flood up prior to entry into refuel mode, reactor vessel level instrument LI-4541 (WR GEMAC, FLOODUP) on control room panel 1C04 is placed in service by I&C personnel connecting a compensating air signal after the reference leg is disconnected from the reactor head. Normal refuel water level is above the top of the span of this flood up level indicator. A valid indication (e.g., not due to loss of compensating air signal or other instrument channel failure) of reactor cavity level coming on span for this instrument is used at DAEC as an indicator of uncontrolled reactor cavity level decrease.

DAEC Technical Specifications require a minimum of 36 feet of water in the spent fuel pool when moving irradiated fuel into the secondary containment. During refueling, the gates between the reactor cavity and the refueling cavity are removed and the spent fuel pool level indicator LI- 3413 is used to monitor refueling water level. Procedures require that a normal refueling water level be maintained at 37 feet 5 inches. A low level alarm actuates when spent fuel pool level drops below 37 feet 1 inch. Symptoms of inventory loss at DAEC include visual observation of decreasing water levels in reactor cavity or spent fuel storage pool, Reactor Building (RB) fuel storage pool radiation monitor or refueling area radiation monitor alarms, observation of a decreasing trend on the spent fuel pool water level indicator, and actuation of the spent fuel pool low water level alarm. To eliminate minor level perturbations from concern, DAEC uses LI-3413 indicated water level below 36 feet and lowering.

Increased radiation levels can be detected by the local area radiation monitors surrounding the spent fuel pool and refueling cavity areas. Applicable area radiation monitors are those listed in AOP 981.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC ~~AA2~~RA2.

~~**Developer Notes:**~~

~~The "site-specific level indications" are those indications that may be used to monitor water level in the various portions of the REFUELING PATHWAY. Specify the mode applicability of a particular indication if it is not available in all modes.~~

~~The "site-specific list of area radiation monitors" should contain those area radiation monitors that would be expected to have increased readings following a decrease in water level in the site-specific REFUELING PATHWAY. In cases where a radiation monitor(s) is not available or would not provide a useful indication, consideration should be given to including alternate indications such as UNPLANNED changes in tank and/or sump levels.~~
~~Development of the EALs should consider the availability and limitations of mode-dependent, or other controlled but temporary, radiation monitors. Specify the mode applicability of a particular monitor if it is not available in all modes.~~
~~ECL Assignment Attributes: 3.1.1.A and 3.1.1.B~~

~~AA1~~RA1

ECL: Alert

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example Emergency Action Levels:**~~
~~(1 or 2 or 3 or 4)~~

Notes:

- The Emergency Director should declare the ~~Alert~~ event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit~~15 minutes~~.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RA1.1 Reading on **ANY** ~~of the following~~Table R-1 effluent radiation monitor~~s~~ greater than ~~the reading shown~~column “Alert” for 15 minutes or longer:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
Gaseous	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
Liquid	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
Liquid	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

RA1.2 ~~(site-specific monitor list and threshold values)~~
Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE
________or 50 mrem thyroid CDE at or beyond ~~(site-specific dose receptor point)~~SITE BOUNDARY. [Preferred]

RA1.3 Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond ~~(site-specific dose receptor point)~~the SITE BOUNDARY for one hour of exposure.

R~~a~~A1.4 Field survey results indicate **EITHER** of the following at or beyond ~~(site-specific dose receptor point)~~the SITE BOUNDARY:

- Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

This IC is modified by a note that EAL RA1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC ~~AS1~~RS1.

~~**Developer Notes:**~~

~~While this IC may not be met absent challenges to one or more fission product barriers, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status or the fission product matrix alone. For many of the DBAs analyzed in the Updated Final Safety Analysis Report, the discriminator will not be the number of fission product barriers challenged, but rather the amount of radioactivity released to the environment.~~

~~The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE…".~~

~~The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power~~

~~plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.~~

~~The "site-specific monitor list and threshold values" should be determined with consideration of the following:~~

- ~~Selection of the appropriate installed gaseous and liquid effluent monitors.~~
- ~~The effluent monitor readings should correspond to a dose of 10 mrem TEDE or 50 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.~~
- ~~Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for ICs AS1 and AG1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for ICs AS1 and AG1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.~~

~~The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.~~

~~Indications from a real-time dose projection system are not included in the generic EALs.~~

~~Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~- Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.2.C~~

~~AA2~~RA2

ECL: Alert

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel.

Operating Mode Applicability: All

Emergency Action Levels:

~~Example Emergency Action Levels: (1 or 2 or 3)~~

RA2.1 Uncovery of irradiated fuel in the REFUELING PATHWAY.

RA2.2 Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by ~~ANY of the following radiation monitors:~~Hi Rad alarm for **ANY** of the following ARMs:

- Spent Fuel Pool Area, RI-9178
- North Refuel Floor, RI-9163
- New Fuel Vault Area, RI-9153
- South Refuel Floor, RI-9164

OR

Reading greater than 5 R/hr on ~~ANY~~**ANY** of the following radiation monitors (in Mode 5 only):

- NW Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

RA2.3 ~~(site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms)~~

Lowering of spent fuel pool level to ~~(site-specific Level 2 value). [*See Developer Notes*]~~25.17 feet.

Definitions:

REFUELING PATHWAY – The reactor refueling cavity, spent fuel pool and fuel transfer canal.

Basis:

This IC addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool~~(*see Developer Notes*)~~.

These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

Expected radiation monitor alarm(s) during preplanned transfer of highly radioactive material through the affected areas are not considered valid alarms for the purpose of comparison to these EALs.

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1.

Escalation of the emergency would be based on either Recognition Category ~~A~~ R or C ICs.

EAL RA2.1

This EAL escalates from ~~AU2~~ RU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used ~~(e.g., a boil-off curve)~~. Classification of an event using this EAL should be based on the totality of available indications, reports, and observations.

While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

EAL RA2.2

This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. ~~A rise in readings~~An alarm on these radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).

Threshold values for the Drywell monitors are only applicable in Mode 5 since the calculated radiation levels from damage to irradiated fuel would be masked by the typical background levels on these monitors during plant operation, and mechanical damage to a fuel assembly in the vessel can only happen with the reactor head removed.

EAL RA2.3

Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.

Escalation of the emergency classification level would be via ICs ~~AS1~~ RS1 or ~~AS2~~ RS2~~(*see AS2 Developer Notes*)~~.

~~**Developer Notes:**~~

~~For EAL #1~~

~~Depending upon the availability and range of instrumentation, this EAL may include specific readings indicative of fuel uncovery; consider water and radiation level readings. Specify the mode applicability of a particular indication if it is not available in all modes.~~

~~For EAL #2~~

~~The "site specific listing of radiation monitors, and the associated readings, setpoints and/or alarms" should contain those radiation monitors that could be used to identify damage to an irradiated fuel assembly (e.g., confirmatory of a release of fission product gases from irradiated fuel).~~

~~For EALs #1 and #2~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display~~

~~range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build in an appropriate level of corroboration between monitor readings into the classification assessment.~~

~~Development of the EALs should also consider the availability and limitations of mode-dependent, or other controlled but temporary, radiation monitors. Specify the mode applicability of a particular monitor if it is not available in all modes.~~

~~For EAL #3~~

~~In accordance with the discussion in Section 1.4, NRC Order EA-12-051, it is recommended that this EAL be implemented when the enhanced spent fuel pool level instrumentation is available for use. The "site-specific Level 2 value" is usually the spent fuel pool level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck. This site-specific level is determined in accordance with NRC Order EA-12-051 and NEI 12-02, and applicable owner's group guidance.~~

~~Developers should modify the EAL and/or Basis section to reflect any site-specific constraints or limitations associated with the design or operation of instrumentation used to determine the Level 2 value.~~

~~ECL Assignment Attributes: 3.1.2.B and 3.1.2.C~~

~~AA3~~RA3

ECL: Alert

Initiating Condition: Radiation levels that impede access to ~~equipment~~ areas necessary for normal plant operations~~, cooldown or shutdown~~.

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example Emergency Action Levels:** (1 or 2)~~

~~**Note:** If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.~~

RA3.1 Dose rate greater than 15 mR/hr in **ANY** of the following areas:

- Control Room ~~ARM~~ (RM-9162)
- Central Alarm Station (by survey)
  ~~(other site-specific areas/rooms)~~

~~2 An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas:~~

~~(site-specific list of plant rooms or areas with entry-related mode applicability identified)~~

Definitions:

None

Basis:

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Director should consider the cause of the increased radiation levels and determine if another IC may be applicable.

~~For EAL 2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).~~

~~An emergency declaration is not warranted if any of the following conditions apply.~~

~~The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.~~

~~The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).~~

~~The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).~~

~~The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.~~

Escalation of the emergency classification level would be via Recognition Category ~~A~~R, C or F ICs.

~~**Developer Notes:**~~

~~EAL #1~~

~~The value of 15mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times.~~

~~The "other site-specific areas/rooms" should include any areas or rooms requiring continuous occupancy to maintain normal plant operation, or to perform a normal cooldown and shutdown.~~

~~EAL #2~~

~~The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed. (e.g., an action to address an off normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.~~

~~The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).~~

~~If the equipment in the listed room or area was already inoperable, or out of service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.~~

~~Rooms and areas listed in EAL #1 do not need to be included in EAL #2, including the Control Room.~~

~~ECL Assignment Attributes: 3.1.2.C~~

~~AS1~~RS1

ECL: Site Area Emergency

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example Emergency Action Levels:**~~
~~(1 or 2 or 3)~~

Notes:

- The Emergency Director should declare the ~~Site Area Emergency~~event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded ~~15 minutes~~the specified time limit.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RS1.1 Reading on **ANY** ~~of the following~~Table R-1 effluent radiation monitor~~s~~ greater than column "SAE" ~~the reading shown~~ for 15 minutes or longer:

~~Effluent Monitor Classification Thresholds~~		
	~~Monitor~~	~~SAE~~
~~Gaseous~~	~~Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)~~	~~1.0E-01 uCi/cc~~
	~~Turbine Building ventilation rad monitor (Kaman 1/2)~~	~~1.0E-01 uCi/cc~~
	~~Offgas Stack rad monitor (Kaman 9/10)~~	~~4.5E+02 uCi/cc~~
	~~LLRPSF rad monitor (Kaman 12)~~	~~1.0E-01 uCi/cc~~

Table R-1 - Effluent Monitor Classification Thresholds					
	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
	LLRPSF rad monitor (Kaman 12)	--	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	--	--	1.7E+04 cps	1.5E+03 cps
	RHRSW & ESW rad monitor (RM-1997)	--	--	1.2E+04 cps	8.4E+02 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	--	--	1.8E+04 cps	1.0E+03 cps

RS1. ~~(site-specific monitor list and threshold values)~~

2 Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond ~~(site-specific dose receptor point)~~the SITE BOUNDARY. [Preferred]

RS1.3 Field survey results indicate **EITHER** of the following at or beyond ~~(site-specific dose receptor point)~~the SITE BOUNDARY:

- Closed window dose rates greater than 100 mR/hr expected to continue for 60 -minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

This IC is modified by a note that EAL RS1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions. However, if Kaman monitor readings are sustained for 15 minutes or longer and the required MIDAS dose assessments cannot be completed within this period, then the declaration can be made using Kaman readings PROVIDED the readings are not from an isolated flow path.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. If Kaman readings are not valid, field survey results may be utilized to assess this IC using EAL RS1.3.

Escalation of the emergency classification level would be via IC ~~AG1~~RG1.

~~**Developer Notes:**~~

~~While this IC may not be met absent challenges to multiple fission product barriers, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status or the fission product matrix alone. For many of the DBAs analyzed in the Updated Final Safety Analysis Report, the discriminator will not be the number of fission product barriers challenged, but rather the amount of radioactivity released to the environment.~~

~~The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE…".~~

~~The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power~~

~~plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.~~

~~The "site-specific monitor list and threshold values" should be determined with consideration of the following:~~

- ~~Selection of the appropriate installed gaseous effluent monitors.~~
- ~~The effluent monitor readings should correspond to a dose of 100 mrem TEDE or 500 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.~~
- ~~Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for ICs AA1 and AG1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for ICs AA1 and AG1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.~~

~~The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.3.C~~

~~AS2~~RS2
~~*[See Developer Notes]*~~

ECL: Site Area Emergency

Initiating Condition: Spent fuel pool level at ~~(site-specific Level 3 description)~~16.36 feet.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

RS2.1 Lowering of spent fuel pool level to 16.36 feet.~~(site-specific Level 3 value).~~

Definitions:

None

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

It is recognized that this IC would likely not be met until well after another Site Area Emergency IC was met; however, it is included to provide classification diversity.

Escalation of the emergency classification level would be via IC ~~AG1~~ RG1 or ~~AG2~~RG2.

~~**Developer Notes:**~~

~~In accordance with the discussion in Section 1.4, NRC Order EA-12-051, it is recommended that this IC and EAL be implemented when the enhanced spent fuel pool level instrumentation is available for use. The "site-specific Level 3 value" is usually that spent fuel pool level where fuel remains covered and actions to implement make-up water addition should no longer be deferred. This site-specific level is determined in accordance with NRC Order EA-12-051 and NEI 12-02, and applicable owner's group guidance.~~

~~Developers should modify the EAL and/or Basis section to reflect any site-specific constraints or limitations associated with the design or operation of instrumentation used to determine the Level 3 value.~~

~~ECL Assignment Attributes: 3.1.3.B~~

~~A~~RG1

ECL: General Emergency

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example Emergency Action Levels:**~~
~~(1 or 2 or 3)~~

Notes:

- The Emergency Director should declare the ~~General Emergency~~event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded ~~15 minutes~~the specified time limit.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RG1.1 Reading on **ANY** ~~of the following~~Table R-1 effluent radiation monitor~~s~~ greater than ~~the reading shown~~column "GE" for 15 minutes or longer:

~~Effluent Monitor Classification Thresholds~~		
	~~Monitor~~	~~GE~~
~~Gaseous~~	~~Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)~~	~~1.0E+00 uCi/cc~~
	~~Turbine Building ventilation rad monitor (Kaman 1/2)~~	~~1.0E+00 uCi/cc~~
	~~Offgas Stack rad monitor (Kaman 9/10)~~	~~4.5E+03 uCi/cc~~

Table R-1 - Effluent Monitor Classification Thresholds					
-	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

~~(site-specific monitor list and threshold values)~~

RG1.2 Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond ~~(site-specific dose receptor point)~~the SITE BOUNDARY. [Preferred]

RG1.3 Field survey results indicate **EITHER** of the following at or beyond ~~(site-specific dose receptor point)~~the SITE BOUNDARY:

- Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

This IC is modified by a note that EAL RG1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions. However, if Kaman monitor readings are sustained for 15 minutes or longer and the required MIDAS dose assessments cannot be completed within this period, then the declaration can be made using Kaman readings PROVIDED the readings are not from an isolated flow path.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. If Kaman readings are not valid, field survey results may be utilized to assess this IC using EAL RG1.3.

~~**Developer Notes:**~~

~~The effluent ICs/EALs are included to provide a basis for classifying events that cannot be readily classified on the basis of plant conditions alone. The inclusion of both types of ICs/EALs more fully addresses the spectrum of possible events and accidents.~~

~~While this IC may not be met absent challenges to multiple fission product barriers, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant status or the fission product matrix alone. For many of the DBAs analyzed in the Updated Final Safety Analysis Report, the discriminator will not be the number of fission product barriers challenged, but rather the amount of radioactivity released to the environment.~~

~~The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE…".~~

~~The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.~~

~~The "site-specific monitor list and threshold values" should be determined with consideration of the following:~~

- ~~Selection of the appropriate installed gaseous effluent monitors.~~
- ~~The effluent monitor readings should correspond to a dose of 1,000 mrem TEDE or 5,000 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.~~
- ~~Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for ICs AA1 and AS1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for ICs AA1 and AS1. Acceptable sources of this information include, but are not limited to, the RETS/ODCM and values used in the site's emergency dose assessment methodology.~~
- ~~Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.~~

~~The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.4.C~~

~~AG2~~RG2

~~[See Developer Notes]~~

ECL: General Emergency

Initiating Condition: Spent fuel pool level cannot be restored to at least 16.36 feet~~.~~ ~~(site-specific Level 3 description)~~ for 60 minutes or longer.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

Note: The Emergency Director should declare the ~~General Emergency~~ event promptly upon determining that the applicable time ~~60 minutes~~ has been exceeded, or will likely be exceeded.

RG2.1 Spent fuel pool level cannot be restored to at least 16.36 feet~~.~~ ~~(site-specific Level 3 value)~~ for 60 minutes or longer.

Definitions:

None

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another General Emergency IC was met; however, it is included to provide classification diversity.

~~Developer Notes:~~

~~In accordance with the discussion in Section 1.4, NRC Order EA-12-051, it is recommended that this IC and EAL be implemented when the enhanced spent fuel pool level instrumentation is available for use. The "site-specific Level 3 value" is usually that spent fuel pool level where fuel remains covered and actions to implement make-up water addition should no longer be deferred. This site-specific level is determined in accordance with NRC Order EA-12-051 and NEI 12-02, and applicable owner's group guidance.~~

~~Developers should modify the EAL and/or Basis section to reflect any site-specific constraints or limitations associated with the design or operation of instrumentation used to determine the Level 3 value.~~

~~ECL Assignment Attributes: 3.1.4.C~~

7 COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

~~Table C-1: Recognition Category "C" Initiating Condition Matrix~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
~~**CU1** UNPLANNED loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory for 15 minutes or longer.~~ ~~*Op. Modes: Cold Shutdown, Refueling*~~*5, 6*	~~**CA1** Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~	~~**CS1** Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory affecting core decay heat removal capability.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~	~~**CG1** Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory affecting fuel clad integrity with containment challenged.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~
~~**CU2** Loss of all but one AC power source to emergency buses for 15 minutes or longer.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling, Defueled*~~	~~**CA2** Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling, Defueled*~~		
~~**CU3** UNPLANNED increase in RCS temperature.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~	~~**CA3** Inability to maintain the plant in cold shutdown.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~		
~~**CU4** Loss of Vital DC power for 15 minutes or longer.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold Shutdown, Refueling*~~	—	—	
~~**CU5** Loss of all onsite or offsite communications capabilities.~~ ~~*Op. Modes:*~~ *5, 6*~~*Cold*~~ Shutdown, ~~*Refueling, Defueled*~~	—	—	

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
	~~**CA6** Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.~~ ~~*Op. Modes: 5, 6Cold Shutdown, Refueling*~~		

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

CU1

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ inventory for 15 minutes or longer.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~4, 5

Emergency Action Levels:

~~**Example Emergency Action Levels:** (1 or 2)~~

Note: The Emergency Director should declare the ~~Unusual Event~~event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

CU1.1 UNPLANNED loss of reactor coolant results ~~[~~in ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level less than a required lower limit for 15 minutes or longer.

CU1.2 a. ~~(Reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level cannot be monitored.

AND

b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool~~increase in (site-specific sump and/or tank) Suppression Pool,or Drywell and Reactor Building floor and equipment drain sump levels~~.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

EAL CU1.1 recognizes that the minimum required ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable-_operating procedure but may be specified in another controlling document.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

~~———~~EAL CU1.2 addresses a condition where all means to determine ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level have been lost. ~~In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the (reactor vessel/RCS [*PWR*] or RPV [*BWR*]).~~

If all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RSC inventory loss was occurring by observing sump and Suppression Pool level changes. The drywell floor and equipment drain sumps, reactor building equipment and floor drain sumps receive all liquid waste from floor and equipment drains inside the primary containment and reactor building. A rise in Suppression Pool water level may be indicative of valve misalignment or leakage in systems that discharge to the Torus. Sump and Suppression Pool level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

~~**Developer Notes:**~~

~~EAL #1 – It is recognized that the minimum allowable reactor vessel/RCS/RPV level may have many values over the course of a refueling outage. Developers should solicit input from licensed operators concerning the optimum wording for this EAL statement. In particular, determine if the generic wording is adequate to ensure accurate and timely classification, or if specific setpoints can be included without making the EAL statement unwieldy or potentially inconsistent with actions that may be taken during an outage. If specific setpoints are included, these should be drawn from applicable operating procedures or other controlling documents.~~

~~EAL #2.b – Enter any "site-specific sump and/or tank" levels that could be expected to increase if there were a loss of inventory (i.e., the lost inventory would enter the listed sump or tank).~~

~~ECL Assignment Attributes: 3.1.1.A~~

CU2

ECL: Notification of Unusual Event

Initiating Condition: Loss of all but one AC power source to ~~emergency~~ essential buses for 15 minutes or longer.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~4, 5, Defueled

~~Example~~ Emergency Action Levels:

Note: The Emergency Director should declare the ~~Unusual Event~~event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU2.1 a. AC power capability to ~~(site-specific emergency buses)~~1A3 and 1A4 buses is reduced to a single power source for 15 minutes or longer.

AND

b. Any additional single power source failure will result in loss of ~~all~~ **ALL** AC power to SAFETY SYSTEMS.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety-related.~~

Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- ~~A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency~~ essential ~~buses being back-fed from the unit main generator.~~
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of

~~emergency~~ essential buses being ~~back~~-fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

~~Developer Notes:~~

~~For a power source that has multiple generators, the EAL and/or Basis section should reflect the minimum number of operating generators necessary for that source to provide required power to an AC emergency bus. For example, if a backup power source is comprised of two generators (i.e., two 50%-capacity generators sized to feed 1 AC emergency bus), the EAL and Basis section must specify that both generators for that source are operating.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~Developers should modify the bulleted examples provided in the basis section, above, as needed to reflect their site-specific plant designs and capabilities.~~

~~The EALs and Basis should reflect that each independent offsite power circuit constitutes a single power source. For example, three independent 345kV offsite power circuits (i.e., incoming power lines) comprise three separate power sources. Independence may be determined from a review of the site-specific UFSAR,~~

~~**SBO analysis or related loss of electrical power studies.**~~

~~**The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is recognized in AOPs and EOPS, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2.**~~

~~**At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.**~~

~~**ECL Assignment Attributes: 3.1.1.A**~~

CU3

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED increase in RCS temperature.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~4, 5

Emergency Action Levels:

~~**Example Emergency Action Levels: (1 or 2)**~~

Note: The Emergency Director should declare the ~~Unusual Event~~event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

CU3.1 UNPLANNED increase in RCS temperature to greater than ~~(site-specific Technical Specification cold shutdown temperature limit)~~212°F.

CU3.2 Loss of **ALL** RCS temperature and ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level indication for 15 minutes or longer.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

~~CONTAINMENT CLOSURE: The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.~~

Basis:

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or the inability to determine RCS temperature and level, represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

EAL CU3.1 involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

EAL CU3.2 reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

~~**Developer Notes:**~~

~~For EAL #1, enter the "site-specific Technical Specification cold shutdown temperature limit" where indicated.~~

~~ECL Assignment Attributes: 3.1.1.A~~

CU4

ECL: Notification of Unusual Event

Initiating Condition: Loss of Vital DC power for 15 minutes or longer.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~4, 5

~~Example Emergency~~Emergency Action Levels:

Note: The Emergency Director should declare the ~~Unusual E~~event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

CU4.1 Indicated voltage is less than ~~(site-specific bus voltage value)~~105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses~~required Vital DC buses~~ for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety--related.~~

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the Vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category ~~A~~R.

~~**Developer Notes:**~~

~~The "site-specific bus voltage value" should be based on the minimum bus voltage necessary for adequate operation of SAFETY SYSTEM equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed.~~

~~The typical value for an entire battery set is approximately 105 VDC. For a 60 cell string of batteries, the cell voltage is approximately 1.75 Volts per cell. For a 58 string battery set, the minimum voltage is approximately 1.81 Volts per cell.~~

~~ECL Assignment Attributes: 3.1.1.A~~

CU5

ECL: Notification of Unusual Event

Initiating Condition: Loss of all onsite or offsite communications capabilities.

Operating Mode Applicability: ~~Cold Shutdown, Refueling5, 6~~4, 5, Defueled

Emergency Action Levels:

~~**Example EmergencyEmergency Action Levels: (1 or 2 or 3)**~~

CU5.1 Loss of **ALL** of the following onsite communication methods:

- ~~(site-specific list of communications methods)~~Plant Operations Radio System
- In-Plant Phone System
- Plant Paging System (Gaitronics)

CU5.2 Loss of **ALL** of the following ~~ORO~~offsite response organization communications methods:

- DAEC All-Call phone
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system
- FTS Phone system

~~(site-specific list of communications methods)~~

CU5.3 Loss of **ALL** of the following NRC communications methods:

- FTS Phone system
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system
- ~~(site-specific list of communications methods)~~

Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to ~~ORO~~offsite response organizations and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

EAL CU5.1 addresses a total loss of the communications methods used in support of routine plant operations..

EAL CU5.2 addresses a total loss of the communications methods used to notify all ~~ORO~~offsite response organizations of an emergency declaration. The offsite response organizations referred to here are the State of Iowa, Linn County, and Benton County~~The OROs referred to here are (see Developer Notes)~~.

EAL CU5.3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

~~**Developer Notes:**~~

~~EAL #1 – The "site-specific list of communications methods" should include all communications methods used for routine plant communications (e.g., commercial or site telephones, page party systems, radios, etc.). This listing should include installed plant equipment and components, and not items owned and maintained by individuals.~~

~~EAL #2 – The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to OROs as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. Example methods are ring-down/dedicated telephone lines, commercial telephone lines, radios, satellite telephones and internet-based communications technology.~~

~~In the Basis section, insert the site-specific listing of the OROs requiring notification of an emergency declaration from the Control Room in accordance with the site Emergency Plan, and typically within 15 minutes.~~

~~EAL #3 – The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to the NRC as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. These methods are typically the dedicated Emergency Notification System (ENS) telephone line and commercial telephone lines.~~

~~ECL Assignment Attributes: 3.1.1.C~~

CA1

ECL: Alert

Initiating Condition: Loss of ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ inventory.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~4, 5

Emergency Action Levels:

~~**Example Emergency**~~**Emergency** ~~**Action Levels:** (1 or 2)~~

Note: The Emergency Director should declare the ~~Alert~~ event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

CA1.1 Loss of ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ inventory as indicated by level less than ~~(site-specific level)~~119.5 inches.

CA1.2 a. ~~(Reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level cannot be monitored for 15 minutes or longer

AND

b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool ~~UNPLANNED increase in (site-specific sump and/or tank) levels~~ due to a loss of ~~(reactor vessel/RCS [PWR] or~~ RPV ~~[BWR])~~ inventory.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For EAL CA1.1, a lowering of water level below ~~(site specific level)~~119.5 inches indicates that operator actions have not been successful in restoring and maintaining ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

Although related, EAL CA1.1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

For EAL CA1.2, the inability to monitor ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, ~~operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the (reactor vessel/RCS [*PWR*] or RPV [*BWR*]).~~ the operators would need to determine that ~~RSC~~RCS inventory loss was occurring by observing sump and Suppression Pool level changes. The drywell floor and equipment drain sumps, reactor building equipment and floor drain sumps receive all liquid waste from floor and equipment drains inside the primary containment and reactor building. A rise in Suppression Pool water level may be indicative of valve misalignment or leakage in systems that discharge to the Torus. Sump and Suppression Pool level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1

If the ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

~~**Developer Notes:**~~

~~For EAL #1 – the "site-specific level" should be based on either:~~

- ~~[*BWR*] Low-Low ECCS actuation setpoint/Level 2. This setpoint was chosen because it is a standard operationally significant setpoint at which some (typically high pressure ECCS) injection systems would automatically start and is a value significantly below the low RPV water level RPS actuation setpoint specified in IC CU1.~~
- ~~[*PWR*] The minimum allowable level that supports operation of normally used decay heat removal systems (e.g., Residual Heat Removal or Shutdown Cooling). If multiple levels exist, specify each along with the appropriate mode or configuration dependency criteria.~~

~~For EAL #2 – The type and range of RCS level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for a PWR. As appropriate to the plant design, alternate means of determining RCS level are installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. The instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown.~~

~~Enter any "site-specific sump and/or tank" levels that could be expected to increase if there were a loss of inventory (i.e., the lost inventory would enter the listed sump or tank).~~

~~ECL Assignment Attributes: 3.1.2.B~~

CA2

ECL: Alert

Initiating Condition: Loss of all offsite and all onsite AC power to ~~emergency~~ essential buses for 15 minutes or longer.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~4, 5, Defueled

Emergency Action Levels:

~~**Example Emergency**~~**Emergency** ~~**Action Levels:**~~

Note: The Emergency Director should declare the ~~Alert~~ event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

CA2.1 Loss of **ALL** offsite and **ALL** onsite AC Power to ~~(site-specific emergency buses)~~1A3 and 1A4 buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety - related.~~

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via IC CS1 or ~~AS1~~RS1.

~~**Developer Notes:**~~

~~For a power source that has multiple generators, the EAL and/or Basis section should reflect the minimum number of operating generators necessary for that source to provide adequate power to an AC emergency bus. For example, if a backup power source is comprised of two generators~~

~~(i.e., two 50% capacity generators sized to feed 1 AC emergency bus), the EAL and Basis section must specify that both generators for that source are operating.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.2.B~~

CA3

ECL: Alert

Initiating Condition: Inability to maintain the plant in cold shutdown.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~4, 5

Emergency Action Levels:

~~**Example Emergency Action Levels:**~~

~~(1 or 2)~~

Note: The Emergency Director should declare the ~~Alert~~ event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CA3.1 UNPLANNED increase in RCS temperature to greater than ~~(site-specific Technical Specification cold shutdown temperature limit)~~212°F for greater than the duration specified in ~~the following table~~Table C-2.

Table C-2~~:~~ RCS Heat-up Duration Thresholds		
RCS ~~Status~~Integrity	**CONTAINMENT CLOSURE Status**	**Heat-up Duration**
Intact ~~(but not at reduced inventory [*PWR*])~~	Not applicable	60 minutes*
Not intact ~~(or at reduced inventory [*PWR*])~~	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

CA3.2 UNPLANNED RCS pressure increase greater than ~~(site-specific pressure reading)~~10 psig due to a loss of RCS cooling. ~~(This EAL does not apply during water-solid plant conditions. [*PWR*])~~

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

~~CONTAINMENT CLOSURE: The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.~~

Basis:

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

RCS integrity is intact when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams).

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact~~, or RCS inventory is reduced (e.g., mid-loop operation in PWRs).~~. The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact~~ or is at reduced inventory [*PWR*]~~, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because

1) the evaporated reactor coolant may be released directly into the Containment atmosphere and subsequently to the environment, and
2) there is reduced reactor coolant inventory above the top of irradiated fuel.

EAL CA3.2 provides a pressure-based indication of RCS heat-up.

Escalation of the emergency classification level would be via IC CS1 or ~~AS1~~RS1.

~~**Developer Notes:**~~

~~For EAL #1 – Enter the "site-specific Technical Specification cold shutdown temperature limit" where indicated. The RCS should be considered intact or not intact in accordance with site-specific criteria.~~

~~For EAL #2 – The "site-specific pressure reading" should be the lowest change in pressure that can be accurately determined using installed instrumentation, but not less than 10 psig.~~

~~For PWRs, this IC and its associated EALs address the concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*. A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncovery can occur. NRC analyses show that there are sequences that can cause core uncovery in 15 to 20 minutes, and severe core damage within an hour after decay heat removal is lost. The allowed time frames are consistent with the guidance provided by Generic Letter 88-17 and believed to be conservative given that a low pressure Containment barrier to fission product release is established.~~

~~ECL Assignment Attributes: 3.1.2.B~~

CA6

ECL: Alert

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~4, 5

Emergency Action Levels:

~~**Example Emergency Action Levels:**~~ :

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- -If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

CA6.1 a. The occurrence of **ANY** of the Table C-3 hazardous events:~~The occurrence of **ANY** of the following hazardous events:~~

Table C-3 Hazardous Events
• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

- ~~Seismic event (earthquake)~~
- ~~Internal or external flooding event~~
- ~~High winds or tornado strike~~
- ~~FIRE~~
- ~~EXPLOSION~~
- ~~(site-specific hazards)River level above 757 feet~~
- ~~River Water Supply (RWS) pit low level alarm~~
- ~~Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director~~

AND

b. ~~**EITHER** of the following:~~

1. Event damage has caused indications of degraded performance in ~~at least~~ one train of a SAFETY SYSTEM needed for the current operating mode.

~~OR~~ AND

2. ~~2~~**EITHER** of the following: ~~-~~

- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode~~, or~~.

OR

- The event has ~~caused~~ resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM ~~component or structure~~ needed for the current operating mode.~~;~~

Definitions:

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction, or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety--related.~~

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria CA6.1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

An event affecting a single-train SAFETY SYSTEM (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under CA6 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train SAFETY SYSTEM, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

~~This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.~~

~~EAL 1.b.1 addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.~~

~~EAL 1.b.2 addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.~~

Escalation of the emergency classification level would be via IC ~~CS1 or AS1~~RS1.

~~**Developer Notes:**~~

~~For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).~~

~~Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.~~

~~ECL Assignment Attributes: 3.1.2.B~~

CS1

ECL: Site Area Emergency

Initiating Condition: Loss of ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ inventory affecting core decay heat removal capability.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~4, 5

Emergency Action Levels:

~~**Example Emergency Action Levels:**~~
~~(1 or 2 or 3)~~

Note: The Emergency Director should declare the ~~Site Area Emergency~~event promptly upon determining that the applicable time ~~30 minutes~~ has been exceeded, or will likely be exceeded.

CS1.1 a. CONTAINMENT CLOSURE not established.

AND

b. ~~(Reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level ~~LESS THAN~~less than ~~(site-specific level)~~+64 inches~~"~~

CS1.2 a. CONTAINMENT CLOSURE established.

AND

b. ~~(Reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level ~~LESS THAN~~less than ~~(site-specific level).~~+15~~"~~ inches

CS1.3 a. ~~(Reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by ~~ANY~~ **EITHER** of the following:

- ~~(Site-specific radiation monitor)~~ Drywell Monitor (9184A/B) reading greater than ~~(site-specific value)~~5.0 R/hr
- ~~Erratic source range monitor indication [*PWR*]~~
- UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool ~~UNPLANNED increase in (site-specific sump and/or tank) levels~~ of sufficient magnitude to indicate core uncovery
- ~~(Other site-specific indications)~~

Definitions:

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses a significant and prolonged loss of ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If ~~RCS/~~reactor vessel level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified ~~RCS/~~reactor vessel levels of EALs CS1.1.b and CS1.2.b reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.~~.~~

In the Cold Shutdown and Refueling Modes, LT/LI-4559, 4560, and 4561 (RX VESSEL NARROW RANGE LEVEL) instruments read up to 22" high due to hot calibrations. LI-4541 (WR GEMAC, FLOODUP) should be used in these Modes for comparison to EAL thresholds since it is calibrated cold and reads accurately. If normal means of RPV level indication are not available due to plant evolutions, redundant means of RPV level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

In EAL CS1.3.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~.

These EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*.

Escalation of the emergency classification level would be via IC CG1 or ~~AG1~~RG1.

~~**Developer Notes:**~~

~~Accident analyses suggest that fuel damage may occur within one hour of uncovery depending upon the amount of time since shutdown; refer to Generic Letter 88-17, SECY 91-283, NUREG-1449 and NUMARC 91-06.~~

~~The type and range of RCS level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for a PWR. As~~

~~appropriate to the plant design, alternate means of determining RCS level are installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. The instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown.~~

~~PWR~~

~~For EAL #1.b – the "site-specific level" is 6" below the bottom ID of the RCS loop. This is the level at 6" below the bottom ID of the reactor vessel penetration and not the low point of the loop. If the availability of on-scale level indication is such that this level value can be determined during some shutdown modes or conditions, but not others, then specify the mode-dependent and/or configuration states during which the level indication is applicable. If the design and operation of water level instrumentation is such that this level value cannot be determined at any time during Cold Shutdown or Refueling modes, then do not include EAL #1 (classification will be accomplished in accordance with EAL #3).~~

~~For EAL #2.b – The "site-specific level" should be approximately the top of active fuel. If the availability of on-scale level indication is such that this level value can be determined during some shutdown modes or conditions, but not others, then specify the mode-dependent and/or configuration states during which the level indication is applicable. If the design and operation of water level instrumentation is such that this level value cannot be determined at any time during Cold Shutdown or Refueling modes, then do not include EAL #2 (classification will be accomplished in accordance with EAL #3).~~

~~For EAL #3.b – first bullet - As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a "site-specific radiation monitor" that could be used to detect core uncovery and the associated "site-specific value" indicative of core uncovery. It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build in an appropriate level of corroboration between monitor readings into the classification assessment.~~

~~For EAL #3.b – second bullet - Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.~~

~~For EAL #3.b – third bullet – Enter any 'site-specific sump and/or tank" levels that could be expected to change if there were a loss of RCS/reactor vessel inventory of sufficient magnitude to indicate core uncovery. Specific level values may be included if desired.~~

~~For EAL #3.b – fourth bullet – Developers should determine if other reliable indicators exist to identify fuel uncovery (e.g., remote viewing using cameras). The goal is to identify any unique or~~

~~site-specific indications, not already used elsewhere, that will promote timely and accurate emergency classification.~~

~~BWR~~

~~For EAL #1.b – "site-specific level" is the Low-Low-Low ECCS actuation setpoint / Level 1. The BWR Low-Low-Low ECCS actuation setpoint / Level 1 was chosen because it is a standard operationally significant setpoint at which some (typically low pressure ECCS) injection systems would automatically start and attempt to restore RPV level. This is a RPV water level value that is observable below the Low-Low/Level 2 value specified in IC CA1, but significantly above the Top of Active Fuel (TOAF) threshold specified in EAL #2.~~

~~For EAL #2.b – The "site-specific level" should be for the top of active fuel.~~

~~For EAL #3.b – first bullet - As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a "site-specific radiation monitor" that could be used to detect core uncovery and the associated "site specific value" indicative of core uncovery. It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build in an appropriate level of corroboration between monitor readings into the classification assessment.~~

~~For BWRs that do not have installed radiation monitors capable of indicating core uncovery, alternate site-specific level indications of core uncovery should be used if available.~~

~~For EAL #3.b – second bullet - Because BWR source range monitor (SRM) nuclear instrumentation detectors are typically located below core mid-plane, this may not be a viable indicator of core uncovery for BWRs.~~

~~For EAL #3.b – third bullet – Enter any "site-specific sump and/or tank" levels that could be expected to change if there were a loss of RPV inventory of sufficient magnitude to indicate core uncovery. Specific level values may be included if desired.~~

~~For EAL #3.b – fourth bullet - Developers should determine if other reliable indicators exist to identify fuel uncovery (e.g., remote viewing using cameras). The goal is to identify any unique or site-specific indications, not already used elsewhere, that will promote timely and accurate emergency classification.~~

~~ECL Assignment Attributes: 3.1.3.B~~

CG1

ECL: General Emergency

Initiating Condition: Loss of ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ inventory affecting fuel clad integrity with containment challenged.

Operating Mode Applicability: ~~Cold Shutdown, Refueling~~4, 5

Emergency Action Levels:

~~**Example Emergency**~~**Emergency** ~~**Action Levels:** (1 or 2)~~

Note: The Emergency Director should declare the ~~General Emergency~~event promptly upon determining that the applicable time ~~30 minutes~~ has been exceeded, or will likely be exceeded.

CG1.1 a. ~~(Reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level ~~LESS THAN~~less than ~~(site-specific level)~~+15 "inches for 30 minutes or longer.

AND

b~~b~~. **ANY** indication from the Secondary Containment Challenge Table ~~(see below)~~C-1.

CG1.2
a. ~~(Reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **EIHER**~~**ANY**~~ of the following:

- Drywell Monitor (9184A/B) ~~(Site-specific radiation monitor)~~ reading ~~GREATER THAN~~greater than ~~(site-specific value)~~5.0 R/hr.
- ~~Erratic source range monitor indication [*PWR*]~~
- UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool ~~UNPLANNED increase in (site-specific sump and/or tank) levels~~ of sufficient magnitude to indicate core uncovery.
- ~~(Other site-specific indications)~~

AND

c. **ANY** indication from the Secondary Containment Challenge Table ~~(see below~~ C-1~~)~~.

Table C-1 Secondary Containment Challenge ~~Table~~
• CONTAINMENT CLOSURE not established* • Drywell Hydrogen or Torus Hydrogen ~~GREATER THAN~~greater than 6% **AND** Drywell Oxygen or Torus Oxygen ~~GREATER THAN~~greater than 5% ~~(Explosive mixture) exists inside containment~~ • UNPLANNED increase in containment pressure • Secondary containment radiation monitors above max safe operating limits (MSOL) of EOP 3, Table 6~~radiation monitor reading above (site-specific value) [*BWR*]~~

* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

Definitions:

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.~~CONTAINMENT CLOSURE: The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.~~

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the inability to restore and maintain reactor vessel level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If ~~RCS/~~reactor vessel level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

In EAL CG1.2.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

For EAL CG1.2.b, the calculated radiation level on the Drywell Monitors (9184A/B) is without the reactor head in place. Calculated in radiation levels with the reactor head in place are below the normal variation in background readings of these monitors.

The inability to monitor ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~ level may be caused by instrumentation and/or power failures~~,~~ or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(reactor vessel/RCS [*PWR*] or~~ RPV ~~[*BWR*])~~.

<u>For the Containment Challenge Table, Secondary Containment max safe operating (MSOL) limits from EOP 3 are defined as the highest parameter value at which neither: (1) equipment necessary for the safe shutdown of the plant will fail nor (2) personnel access necessary for the safe shutdown of the plant will be precluded.</u>

~~T~~<u>T</u>hese EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*.

~~**Developer Notes:**~~

~~Accident analyses suggest that fuel damage may occur within one hour of uncovery depending upon the amount of time since shutdown; refer to Generic Letter 88-17, SECY 91-283, NUREG-1449 and NUMARC 91-06.~~

~~The type and range of RCS level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for a PWR. As appropriate to the plant design, alternate means of determining RCS level are installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. The instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown.~~

~~For EAL #1.a – The "site-specific level" should be approximately the top of active fuel. If the availability of on-scale level indication is such that this level value can be determined during some shutdown modes or conditions, but not others, then specify the mode-dependent and/or configuration states during which the level indication is applicable. If the design and operation of water level instrumentation is such that this level value cannot be determined at any time during Cold Shutdown or Refueling modes, then do not include EAL #1 (classification will be accomplished in accordance with EAL #2).~~

~~For EAL #2.b – first bullet – As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a "site-specific radiation monitor" that could be used to detect core uncovery and the associated "site-specific value" indicative of core uncovery. It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than~~

~~approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build in an appropriate level of corroboration between monitor readings into the classification assessment.~~

~~For BWRs that do not have installed radiation monitors capable of indicating core uncovery, alternate site-specific level indications of core uncovery should be used if available.~~

~~For EAL #2.b – second bullet – Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations. Because BWR Source Range Monitor (SRM) nuclear instrumentation detectors are typically located below core mid-plane, this may not be a viable indicator of core uncovery for BWRs.~~

~~For EAL #2.b – third bullet – Enter any "site-specific sump and/or tank" levels that could be expected to change if there were a loss of inventory of sufficient magnitude to indicate core uncovery. Specific level values may be included if desired.~~

~~For EAL #2.b – fourth bullet – Developers should determine if other reliable indicators exist to identify fuel uncovery (e.g., remote viewing using cameras). The goal is to identify any unique or site-specific indications, not already used elsewhere, that will promote timely and accurate emergency classification.~~

~~For the Containment Challenge Table:~~

~~Site shutdown contingency plans typically provide for re-establishing CONTAINMENT CLOSURE following a loss of RCS heat removal or inventory control functions.~~

~~For "Explosive mixture", developers may enter the minimum containment atmospheric hydrogen concentration necessary to support a hydrogen burn (i.e., the lower deflagration limit). A concurrent containment oxygen concentration may be included if the plant has this indication available in the Control Room.~~

~~For BWRs, the use of secondary containment radiation monitors should provide indication of increased release that may be indicative of a challenge to secondary containment. The "site-specific value" should be based on the EOP maximum safe values because these values are easily recognizable and have a defined basis.~~

~~ECL Assignment Attributes: 3.1.4.B~~

8 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) ICS/EALS

~~**Table E-1: Recognition Category "E" Initiating Condition Matrix**~~

~~**UNUSUAL EVENT**~~

~~**E-HU1** Damage to a loaded cask CONFINEMENT BOUNDARY.~~
~~*Op. Modes: All*~~

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

E-HU1

ECL: Notification of Unusual Event

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

E-HU1.1 Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by ~~an on-contact~~a radiation reading greater than the values shown ~~below~~on Table E-1~~(2 times the site-specific cask specific technical specification allowable radiation level)~~ on the ~~surface of the~~ spent fuel cask.

Table E-1 Cask Dose Rates	
61BT DSC	
3 feet from HSM Surface	800 mrem/hr
Outside HSM Door – Centerline of DSC	200 mrem/hr
End Shield Wall Exterior	40 mrem/hr

Definition:

CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

Basis:

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category ~~A~~ R IC R~~A~~U1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

~~ISFSI MALFUNCTION~~

~~Developer Notes:~~

~~The results of the ISFSI Safety Analysis Report (SAR) [per NUREG 1536], or a SAR referenced in the cask Certificate of Compliance and the related NRC Safety Evaluation Report, identify the natural phenomena events and accident conditions that could potentially affect the CONFINEMENT BOUNDARY. This EAL addresses damage that could result from the range of identified natural or man-made events (e.g., a dropped or tipped over cask, EXPLOSION, FIRE, EARTHQUKE, etc.).~~

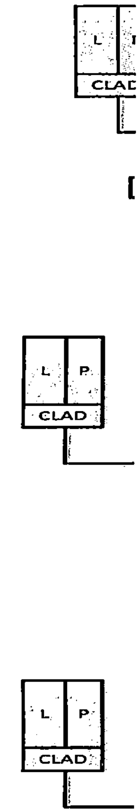
~~The allowable radiation level for a spent fuel cask can be found in the cask's technical specification located in the Certificate of Compliance.~~

~~ECL Assignment Attributes: 3.1.1.B~~

9 FISSION PRODUCT BARRIER ICS/EALS

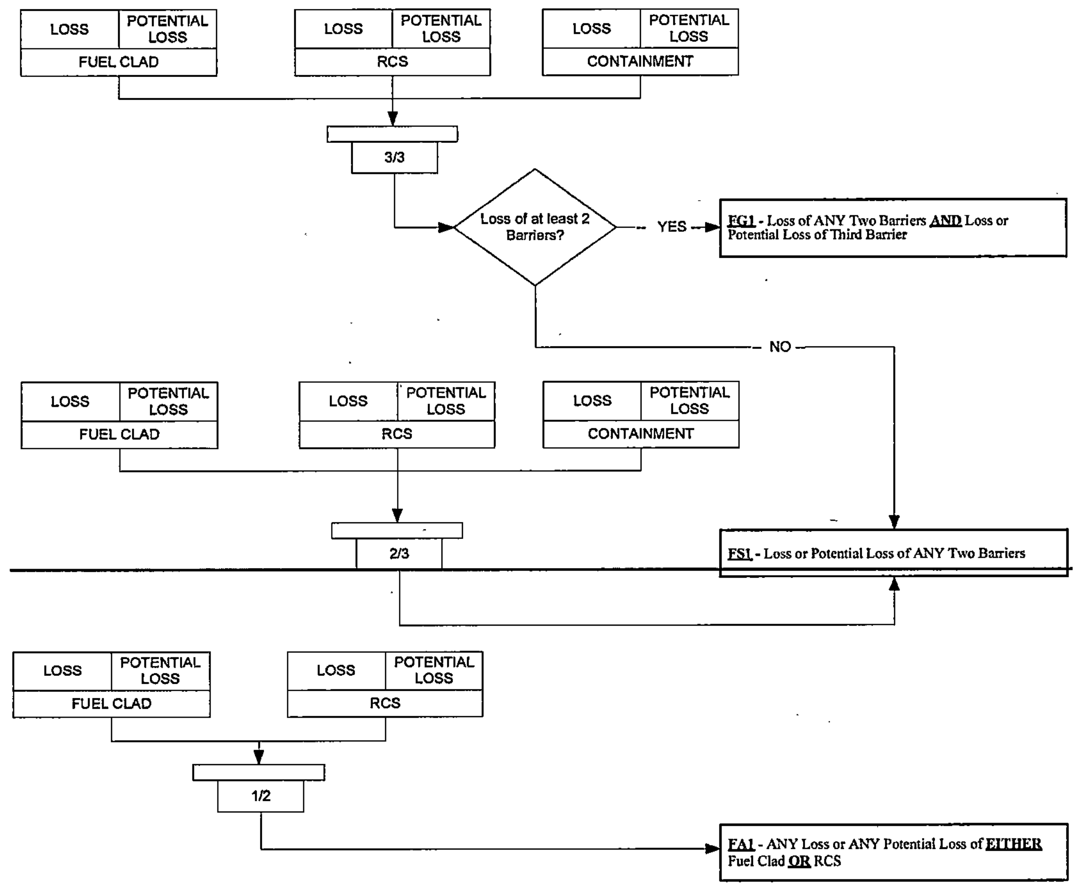
~~Table 9-F-1: Recognition Category "F" Initiating Condition Matrix~~

	~~**ALERT**~~
~~**FA1**~~	~~Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.~~ ~~*Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*~~*1, 2, 3, 4*
	~~**SITE AREA EMERGENCY**~~
~~**FS1**~~	~~Loss or Potential Loss of any two barriers.~~ ~~*Op. Modes:*~~ *1, 2, 3, 4*~~*Power Operation, Hot Standby, Startup, Hot Shutdown*~~
	~~**GENERAL EMERGENCY**~~
~~**FG1**~~	~~Loss of any two barriers and Loss or Potential Loss of the third barrier.~~ ~~*Op. Modes:*~~ *1, 2, 3, 4*~~*Power Operation, Hot Standby, Startup, Hot Shutdown*~~



~~**See Table 9-F-2 for BWR EALs**~~
~~**See Table 9-F-3 for PWR EALs**~~

~~**Developer Note:** The adjacent logic flow diagram is for use by developers and is not required for site-specific implementation; however, a site-specific scheme must include some type of user aid to facilitate timely and accurate classification of fission product barrier losses and/or potential losses. Such aids are typically comprised of logic flow diagrams, "scoring" criteria or checkbox-type matrices. The user-aid logic must be consistent with that of the adjacent diagram.~~


LOSS
POTENTIAL LOSS
FUEL CLAD
LOSS
POTENTIAL LOSS
RCS
LOSS
POTENTIAL LOSS
CONTAINMENT
3/3
Loss of at least 2 Barriers?
YES
FG1 - Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier
NO
LOSS
POTENTIAL LOSS
FUEL CLAD
LOSS
POTENTIAL LOSS
RCS
LOSS
POTENTIAL LOSS
CONTAINMENT
2/3
FS1 - Loss or Potential Loss of ANY Two Barriers
LOSS
POTENTIAL LOSS
FUEL CLAD
LOSS
POTENTIAL LOSS
RCS
1/2
FA1 - ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS

~~Developer Notes~~

1. ~~The logic used for these initiating conditions reflects the following considerations:~~

   - ~~The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.~~

   - ~~Unusual Event ICs associated with fission product barriers are addressed in Recognition Category S.~~

2. ~~For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC AG1 has been exceeded.~~

3. ~~The fission product barrier thresholds specified within a scheme are expected to reflect plant-specific design and operating characteristics. This may require that developers create different thresholds than those provided in the generic guidance.~~

4. ~~Alternative presentation methods for the Recognition Category F ICs and fission product barrier thresholds are acceptable and include flow charts, block diagrams, and checklist-type tables. Developers must ensure that the site-specific method addresses all possible threshold combinations and classification outcomes shown in the BWR or PWR EAL fission product barrier tables. The NRC staff considers the presentation method of the Recognition Category F information to be an important user aid and may request a change to a particular proposed method if, among other reasons, the change is necessary to promote consistency across the industry.~~

5. ~~As used in this Recognition Category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location – inside containment, a secondary-side system (i.e., PWR steam generator tube leakage), an interfacing system, or outside of containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.~~

6. ~~At the Site Area Emergency level, classification decision-makers should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.~~

7. ~~The ability to escalate to a higher emergency classification level in response to degrading conditions should be maintained. For example, a steady increase in RCS leakage would represent an increasing risk to public health and safety.~~

Table ~~9-F~~F-1~~-2~~: ~~BWR~~ DAEC EAL Fission Product Barrier Table

Thresholds for LOSS or POTENTIAL LOSS of Barriers

FA1 ALERT	FS1 SITE AREA EMERGENCY	FG1 GENERAL EMERGENCY
ANY Loss **OR ANY** Potential Loss of **EITHER** the Fuel Clad **OR** RCS barrier~~Any **ANY** Loss **or ANY**any Potential Loss of **either** the Fuel Clad **or OR** RCS barrier~~.	Loss ~~or~~ **OR** Potential Loss of ~~any~~ **ANY** two barriers.	Loss of **ANY** two barriers **AND** Loss **OR** Potential Loss of the third barrier~~Loss of any **ANY** two barriers and Loss **or OR** Potential Loss of the third barrier.~~
Operating Mode Applicability: 1, 2, 3	**Operating Mode Applicability:** 1, 2, 3	**Operating Mode Applicability:** 1, 2, 3

Fuel Clad Barrier		**RCS Barrier**		**Containment Barrier**	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. RCS Activity		**1. Primary Containment ~~Pressure~~Conditions**		**1. Primary Containment Conditions**	
A. Coolant activity greater than 300 µCi/gm dose equivalent I-131.~~A. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).~~	Not Applicable	A. Primary containment pressure greater than ~~(site-specific value)~~2 psig due to RCS leakage.	Not Applicable	A. UNPLANNED rapid drop in ~~primary containment~~Drywell pressure following ~~primary containment~~Drywell pressure rise **OR** B. ~~Primary containment~~Drywell pressure response not consistent with LOCA conditions. **OR** C. UNISOLABLE direct downstream pathway to the environment exists after primary	A. ~~Primary containment~~Torus pressure greater than ~~(site-specific value)~~53 psig **OR** B. Drywell or Torus H2 cannot be determined to be ~~LESS THAN~~less than 6% and Drywell ~~or~~**OR** Torus O2 cannot be determined to be less than 5% ~~**(site-specific explosive mixture) exists inside primary containment**~~ **OR**

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
				containment isolation signal **OR** D. Intentional primary containment venting per EOPs	C. HC~~T~~L (Graph 4 of EOP 2) exceeded.

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
2. RPV Water Level		**2. RPV Water Level**		**2. RPV Water Level**	
A. SAG entry is required~~Primary containment flooding required.~~	A. RPV water level cannot be restored and maintained above ~~(site-specific RPV water level corresponding to the top of active fuel)~~+15 inches **OR** cannot be determined.	A. RPV water level cannot be restored and maintained above ~~(site-specific RPV water level corresponding to the top of active fuel)~~+15 inches **OR** cannot be determined.	Not Applicable	Not Applicable	A. SAG entry is required~~Primary containment flooding required.~~
3. Not Applicable		**3. RCS Leak Rate**		**3. Primary Containment Isolation Failure**	
Not Applicable	Not Applicable	A. UNISOLABLE break in ~~ANY of the following: (site-specific systems with potential for high-energy line breaks)~~Main Steam, HPCI, Feedwater, RWCU, or RCIC as indicated by the failure of both isolation valves in ~~any~~ANY one line to close **AND** **EITHER**: • High MSL flow or steam	A. UNISOLABLE primary system leakage that results in exceeding the Max Normal Operating Limit (MNOL) of EOP 3, Table 6 for **EITHER** of the following: • ~~1. Max Normal Operating~~ Temperature **OR** • ~~2. Max Normal Operating Area~~	A. UNISOLABLE primary system leakage that results in exceeding the Max Safe Operating Limit (MSOL) of EOP 3, Table 6 for **EITHER** of the following: • Temperature **OR** • Radiation Level ~~• A. UNISOLABLE direct downstream~~	Not Applicable

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
		tunnel temperature annunciators **OR** • Direct report of steam release **OR** ~~B.~~ Emergency RPV Depressurization required.	Radiation ~~Level~~Level~~.~~	~~pathway to the environment exists after primary containment isolation signal~~ ~~**OR**~~ ~~B.~~ ~~Intentional primary containment venting per EOPs~~ ~~**OR**~~ ~~C.~~ ~~UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following:~~ ~~1. Max Safe Operating Temperature.~~ ~~**OR**~~ ~~2. Max Safe Operating Area Radiation Level.~~	

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
4. Primary Containment Radiation		**4. Primary Containment Radiation**		**4. Primary Containment Radiation**	
A. Drywell Monitor (9184A/B) reading greater than 1250 R/hr. **OR** B. Torus Monitor (9185A/B) reading greater than 125 R/hr	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 5 R/hr after reactor shutdown~~A. Primary containment radiation monitor reading greater than (site-specific value).~~	Not Applicable	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 5000 R/hr. **OR** B. Torus Monitor (9185A/B) reading greater than 500 R/hr~~A. Primary containment radiation monitor reading greater than (site-specific value).~~
5. Other Indications		**5. Other Indications**		**5. Other Indications**	
A. Fuel damage assessment indicates at least 5% fuel clad damage.~~(site-specific as applicable)~~	Not Applicable~~A. (site-specific as applicable)~~	Not Applicable~~A. (site-specific as applicable)~~	Not Applicable~~A. (site-specific as applicable)~~	Not Applicable~~A. (site-specific as applicable)~~	A. Fuel damage assessment ~~(PASAP 7.2)~~ indicates at least 20% fuel clad damage.~~(site-specific as applicable)~~
6. Emergency Director Judgment		**6. Emergency Director Judgment**		**6. Emergency Director Judgment**	
A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
the Fuel Clad Barrier.	Loss of the Fuel Clad Barrier.	the RCS Barrier.	Loss of the RCS Barrier.	the Containment Barrier.	Loss of the Containment Barrier.

Basis Information For
~~BWR~~ DAEC EAL Fission Product Barrier Table ~~9-F~~F-1-~~2~~

~~BWR~~ DAEC FUEL CLAD BARRIER THRESHOLDS:

The Fuel Clad barrier consists of the zircalloy or stainless steel fuel bundle tubes that contain the fuel pellets.

1. **RCS Activity**

Loss 1.A

This threshold indicates that RCS radioactivity concentration is greater than 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes ~~and corresponds to an approximate range of 2% to 5% fuel clad damage~~. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.

There is no Potential Loss threshold associated with RCS Activity.

~~**Developer Notes:**~~

~~Threshold values should be determined assuming RCS radioactivity concentration equals 300 µCi/gm dose equivalent I-131. Other site-specific units may be used (e.g., µCi/cc).~~

~~Depending upon site-specific capabilities, this threshold may have a sample analysis component and/or a radiation monitor reading component.~~

~~Add this paragraph (or similar wording) to the Basis if the threshold includes a sample analysis component, "It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications."~~

2. **RPV Water Level**

Loss 2.A

The Loss threshold represents any EOP requirement for entry into the Severe Accident Guidelines.

This is identified in the BWROG EPGs/SAGs when adequate core cooling cannot be assured.~~The Loss threshold represents the EOP requirement for primary containment flooding. This is identified in the BWROG EPGs/SAGs when the phrase, "Primary Containment Flooding Is Required," appears. Since a site-specific RPV water level is not specified here, the Loss threshold phrase, "Primary containment flooding required," also accommodates the EOP need to flood the primary containment when RPV water level cannot be determined and core damage due to inadequate core cooling is believed to be occurring.~~

Potential Loss 2.A

This water level corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling.

~~**BWR FUEL CLAD BARRIER THRESHOLDS:**~~

The RPV water level threshold is the same as RCS barrier Loss threshold 2.A. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of the RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

DAEC FUEL CLAD BARRIER THRESHOLDS (cont.):

This threshold is considered to be exceeded when, as specified in the ~~site-specific~~ EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

~~**DAEC FUEL CLAD BARRIER THRESHOLDS (cont.):**~~**:**

The term "cannot be restored and maintained above" means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value *cannot* be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation below the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs ~~SA5~~ SA6 or SS~~5~~6 will dictate the need for emergency classification.

Since the loss of ability to determine if adequate core cooling is being provided presents a significant challenge to the fuel clad barrier, a potential loss of the fuel clad barrier is specified.

~~**BWR FUEL CLAD BARRIER THRESHOLDS:**~~

~~**Developer Notes:**~~

~~Loss 2.A~~

~~The phrase, "Primary containment flooding required," should be modified to agree with the site-specific EOP phrase indicating exit from all EOPs and entry to the SAGs (e.g., drywell flooding required, etc.).~~

~~Potential Loss 2.A~~

~~The decision that "RPV water level cannot be determined" is directed by guidance given in the RPV water level control sections of the EOPs.~~

3. Not Applicable (included for numbering consistency between barrier tables)

DAEC FUEL CLAD BARRIER THRESHOLDS (cont.):

4. Primary Containment Radiation

Loss 4.A and

Loss 4.B

The Drywell and Torus radiation monitor readings corresponds~~s~~ to an instantaneous release of all reactor coolant mass into the Drywell or ~~primary~~ Torus~~containment~~, assuming that reactor coolant activity ~~equals 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and~~ corresponds to ~~an~~ approximately ~~range of 2% to~~ 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor readings in this threshold ~~is~~ are higher than that specified for RCS Barrier Loss threshold 4.A since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Radiation.

~~**Developer Notes:**~~

~~The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS radioactivity concentration equal to 300 μCi/gm dose equivalent I-131, into the primary containment atmosphere.~~

~~BWR~~ DAEC ~~FUEL CLAD BARRIER THRESHOLDS~~ (cont.):

Other Indications~~:~~

5.

~~1. Other Indications~~

Loss ~~and/or Potential Loss~~ 5.A

Results obtained from procedure PASAP 7.2, Fuel Damage Assessment, indicate at least 5% fuel clad damage. ~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Fuel Clad barrier based on plant-specific design characteristics not considered in the generic guidance.~~

There is no Potential Loss threshold associated with Other Indications.

~~Developer Notes:~~

~~Loss and/or Potential Loss 5.A~~

~~Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.~~

~~Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.~~

~~5.~~6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

~~**Developer Notes:**~~

~~None~~

~~BWR~~ DAEC RCS BARRIER THRESHOLDS:

The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping up to and including the isolation valves.

1. Primary Containment ~~Pressure~~Conditions

Loss 1.A

~~The (site-specific value) primary containment~~2 psig ~~pressure~~ is the drywell high pressure scram setpoint which indicates a LOCA by automatically initiating ~~the~~ ECCS ~~or equivalent makeup system.~~.

There is no Potential Loss threshold associated with Primary Containment Pressure.

~~**Developer Notes:**~~

~~None~~

2. RPV Water Level

Loss 2.A

~~This water l~~+15 inches ~~evel~~ corresponds to the top of active fuel (TAF) and is used in the EOPs to indicate challenge to core cooling.

The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold 2.A. Thus, this threshold indicates a Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier and that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the ~~site-specific~~ EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

~~BWR~~ DAEC ~~RCS BARRIER THRESHOLDS:~~

The term, "cannot be restored and maintained above," means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value *cannot* be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active ~~fuel, but~~fuel but does not permit extended operation beyond the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

DAEC RCS BARRIER THRESHOLDS (cont.):

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA5 or SS5 will dictate the need for emergency classification.

There is no RCS Potential Loss threshold associated with RPV Water Level.

3. RCS Leak Rate

Loss Threshold 3.A

Large high-energy lines that rupture outside primary containment can discharge significant amounts of inventory and jeopardize the pressure-retaining capability of the RCS until they are isolated. If it is determined that the ruptured line cannot be promptly isolated from the Control Room, the RCS barrier Loss threshold is met.

Loss Threshold 3.B

Emergency RPV Depressurization in accordance with the EOPs is indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is performed, the plant operators are directed to open safety relief valves (SRVs) and keep them open. Even though the RCS is being vented into the suppression pool, a Loss of the RCS barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

Potential Loss Threshold 3.A

Potential loss of RCS based on primary system leakage outside the primary containment is determined from EOP temperature or radiation Max Normal Operating values in areas such as main steam line tunnel, RCIC, HPCI, etc., which indicate a direct path from the RCS to areas outside primary containment.

A Max Normal Operating Limit (MNOL) value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

~~BWR DAEC RCS BARRIER THRESHOLDS:~~

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage from a primary system warrant an Alert classification. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

An UNISOLABLE leak which is indicated by ~~Max Normal Operating~~MNOL values escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold 3.A (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

DAEC RCS BARRIER THRESHOLDS (cont.):

~~Developer Notes:~~

~~Loss Threshold 3.A~~

~~The list of systems included in this threshold should be the high energy lines which, if ruptured and remain unisolated, can rapidly depressurize the RPV. These lines are typically isolated by actuation of the Leak Detection system.~~

~~Large high energy line breaks such as Main Steam Line (MSL), High Pressure Coolant Injection (HPCI), Feedwater, Reactor Water Cleanup (RWCU), Isolation Condenser (IC) or Reactor Core Isolation Cooling (RCIC) that are UNISOLABLE represent a significant loss of the RCS barrier.~~

4. Primary Containment Radiation

Loss 4.A

The Drywell~~radiation~~ monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold 4.A since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with Primary Containment Radiation.

~~Developer Notes:~~

~~The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS activity at Technical Specification allowable limits, into the primary containment atmosphere. Using RCS activity at Technical Specification allowable limits aligns this threshold with IC SU3. Also, RCS activity at this level will typically result in primary containment~~

~~radiation levels that can be more readily detected by primary containment radiation monitors, and more readily differentiated from those caused by piping or component "shine" sources. If desired, a plant may use a lesser value of RCS activity for determining this value.~~

~~**BWR RCS BARRIER THRESHOLDS:**~~

~~In some cases, the site-specific physical location and sensitivity of the primary containment radiation monitor(s) may be such that radiation from a cloud of released RCS gases cannot be distinguished from radiation emanating from piping and components containing elevated reactor coolant activity. If so, refer to the Developer Guidance for Loss/Potential Loss 5.A and determine if an alternate indication is available.~~

5. Other Indications

There are no Loss or Potential Loss thresholds associated with Other Indications.

~~**Developer Notes:**~~

~~Loss and/or Potential Loss 5.A~~

~~Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.~~

~~Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.~~

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

~~**Developer Notes:**~~

~~None~~

~~BWR~~ DAEC CONTAINMENT BARRIER THRESHOLDS:

The Primary Containment Barrier includes the drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

1. Primary Containment Conditions

Loss 1.A and 1.B

Rapid UNPLANNED loss of ~~primary containment~~ drywell pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase indicates a loss of ~~primary containment~~drywell integrity. ~~Primary containment~~Drywell pressure should increase as a result of mass and energy release into the primary containment from a LOCA. Thus, ~~primary containment~~drywell pressure not increasing under these conditions indicates a loss of primary containment integrity.

These thresholds rely on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Loss 1.C

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS).

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Following the leakage of RCS mass into primary containment and a rise in primary containment pressure, there may be minor radiological releases associated with allowable primary containment leakage through various penetrations or system components. Minor releases may also occur if a primary containment isolation valve(s) fails to close but the primary containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of primary containment but should be evaluated using the Recognition Category R ICs.

~~DAEC CONTAINMENT BARRIER THRESHOLDS:~~

Loss 1.D

EOPs may direct primary containment isolation valve logic(s) to be intentionally bypassed, even if offsite radioactivity release rate limits will be exceeded. Under these conditions with a valid primary containment isolation signal, the containment should also be considered lost if primary containment venting is actually performed.

Intentional venting of primary containment for primary containment pressure or combustible gas control to the secondary containment and/or the environment is a Loss of the Containment. Venting for primary containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.

DAEC CONTAINMENT BARRIER THRESHOLDS (cont.):

~~DAEC CONTAINMENT BARRIER THRESHOLDS:~~

Potential Loss 1.A

The threshold pressure is the ~~primary containment~~Torus internal design pressure. Structural acceptance testing demonstrates the capability of the primary containment to resist pressures greater than the internal design pressure. A pressure of this magnitude is greater than those expected to result from any design basis accident and, thus, represent a Potential Loss of the Containment barrier.

Potential Loss 1.B

If hydrogen concentration reaches or exceeds the lower flammability limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the primary containment, loss of the Containment barrier could occur.

Potential Loss 1.C

The Heat Capacity ~~Temperature~~ Limit (HC~~T~~L) is the highest suppression pool temperature from which Emergency RPV Depressurization will not raise:

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,

  OR

~~**BWR CONTAINMENT BARRIER THRESHOLDS:**~~

- ~~Suppression chamber pressure above Primary Containment Pressure Limit A, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.~~

-

The HC~~T~~L is a function of RPV pressure, suppression pool temperature and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

DAEC CONTAINMENT BARRIER THRESHOLDS (cont.):~~:~~

~~**Developer Notes:**~~

~~Potential Loss 1.B~~

~~BWR EPGs/SAGs specifically define the limits associated with explosive mixtures in terms of deflagration concentrations of hydrogen and oxygen. For Mk I/II containments the deflagration limits are "6% hydrogen and 5% oxygen in the drywell or suppression chamber". For Mk III containments, the limit is the "Hydrogen Deflagration Overpressure Limit". The threshold term "explosive mixture" is synonymous with the EPG/SAG "deflagration limits".~~

~~Potential Loss 1.C~~

~~Since the HCTL is defined assuming a range of suppression pool water levels as low as the elevation of the downcomer openings in Mk I/II containments, or 2 feet above the elevation of the horizontal vents in a Mk III containment, it is unnecessary to consider separate Containment barrier Loss or Potential Loss thresholds for abnormal suppression pool water level conditions. If desired, developers may include a separate Containment Potential Loss threshold based on the inability to maintain suppression pool water level above the downcomer openings in Mk I/II containments, or 2 feet above the elevation of the horizontal vents in a Mk III containment with RPV pressure above the minimum decay heat removal pressure, if it will simplify the assessment of the suppression pool level component of the HCTL.~~

2. RPV Water Level

There is no Loss threshold associated with RPV Water Level.

Potential Loss 2.A

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold 2.A. The Potential Loss requirement for Primary Containment Flooding indicates adequate core cooling cannot be restored and maintained and that core damage is possible. BWR EPGs/SAGs specify the conditions that require primary containment flooding. When primary containment flooding is required, the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to restore and maintain adequate core cooling.

~~BWR CONTAINMENT BARRIER THRESHOLDS:~~

PRA studies indicate that the condition of this Potential Loss threshold could be a core melt sequence which, if not corrected, could lead to RPV failure and increased potential for primary containment failure. In conjunction with the RPV water level Loss thresholds in the Fuel Clad and RCS barrier columns, this threshold results in the declaration of a General Emergency.

~~Developer Notes:~~

~~The phrase, "Primary containment flooding required," should be modified to agree with the site-specific EOP phrase indicating exit from all EOPs and entry to the SAGs (e.g., drywell flooding required, etc.).~~

3. Primary Containment Isolation Failure

These thresholds address incomplete containment isolation that allows an UNISOLABLE direct release to the environment.

~~Loss 3.A~~

~~The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS).~~

~~The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.~~

~~Following the leakage of RCS mass into primary containment and a rise in primary containment pressure, there may be minor radiological releases associated with allowable primary containment leakage through various penetrations or system components. Minor releases may also occur if a primary containment isolation valve(s) fails to close but the primary containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of primary containment but should be evaluated using the Recognition Category A ICs.~~

~~Loss 3.B~~

~~EOPs may direct primary containment isolation valve logic(s) to be intentionally bypassed, even if offsite radioactivity release rate limits will be exceeded. Under these conditions with a valid primary containment isolation signal, the containment should also be considered lost if primary containment venting is actually performed.~~

~~Intentional venting of primary containment for primary containment pressure or combustible gas control to the secondary containment and/or the environment is a Loss of the Containment. Venting for primary containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.~~

Loss 3.~~C~~A

The Max Safe Operating Limit (MSOL) for Temperature and ~~the Max Safe Operating~~ Radiation Level are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. EOPs utilize these temperatures and radiation levels to establish conditions under which RPV depressurization is required.

~~**BWR CONTAINMENT BARRIER THRESHOLDS:**~~

The temperatures and radiation levels should be confirmed to be caused by RCS leakage from a primary system. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

In combination with RCS potential loss 3.A this threshold would result in a Site Area Emergency.

There is no Potential Loss threshold associated with ~~Primary Containment Isolation Failure~~RCS Leak Rate.

DAEC CONTAINMENT BARRIER THRESHOLDS (cont.):~~:~~

~~**Developer Notes:**~~

~~Loss 3.B~~

~~Consideration may be given to specifying the specific procedural step within the Primary Containment Control EOP that defines intentional venting of the Primary Containment regardless of offsite radioactivity release rate.~~

4. Primary Containment Radiation

There is no Loss threshold associated with Primary Containment Radiation.

Potential Loss 4.A

The drywell radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the ~~primary containment~~drywell, assuming that 20% of the fuel cladding has failed. The radiation monitor reading for the torus corresponds to an instantaneous release of all reactor coolant mass directly into the torus, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the emergency classification level to a General Emergency.

~~**Developer Notes:**~~

~~NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, provides the basis for using the 20% fuel cladding failure value. Unless there is a site-specific analysis justifying a different value, the reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad failure into the primary containment atmosphere.~~

~~**BWR CONTAINMENT BARRIER THRESHOLDS:**~~

5. Other Indications

There is no Loss threshold associated with Other Indications

~~Loss and/or~~ Potential Loss 5.A

Results obtained from procedure PASAP 7.2, Fuel Damage Assessment, indicate at least 25% fuel clad damage. ~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Containment barrier based on plant-specific design characteristics not considered in the generic guidance.~~ PASAP 7.2 only shows whether fuel damage is greater than or less than 25%, thus this indication is not likely to be declared before containment barrier potential loss 4.A which indicates 20% fuel damage. However, this potential loss threshold adds an additional layer of diversity to the scheme.

~~**Developer Notes:**~~

~~Loss and/or Potential Loss 5.A~~

~~Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.~~

~~Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.~~

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

~~**Developer Notes:**~~
~~None~~

~~Table 9-F-3: PWR EAL Fission Product Barrier Table~~
~~Thresholds for LOSS or POTENTIAL LOSS of Barriers~~

~~FA1 ALERT~~	~~FS1 SITE AREA EMERGENCY~~	~~FG1 GENERAL EMERGENCY~~
~~Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.~~	~~Loss or Potential Loss of any two barriers.~~	~~Loss of any two barriers and Loss or Potential Loss of the third barrier.~~

~~Fuel Clad Barrier~~		~~RCS Barrier~~		~~Containment Barrier~~	
~~LOSS~~	**~~POTENTIAL LOSS~~**	**~~LOSS~~**	**~~POTENTIAL LOSS~~**	**~~LOSS~~**	**~~POTENTIAL LOSS~~**
~~1. RCS or SG Tube Leakage~~		**~~1. RCS or SG Tube Leakage~~**		**~~1. RCS or SG Tube Leakage~~**	
~~Not Applicable~~	~~A. RCS/reactor vessel level less than (site-specific level).~~	~~A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: UNISOLABLE RCS leakage **OR** SG tube RUPTURE.~~	~~A. Operation of a standby charging (makeup) pump is required by **EITHER** of the following: UNISOLABLE RCS leakage **OR** SG tube leakage.~~ ~~**OR**~~ ~~B. RCS cooldown rate greater than (site-specific pressurized thermal shock criteria/limits defined by site-specific indications).~~	~~A. A leaking or RUPTURED SG is FAULTED outside of containment.~~	~~Not Applicable~~
~~2. Inadequate Heat Removal~~		**~~2. Inadequate Heat Removal~~**		**~~2. Inadequate Heat Removal~~**	
~~A. Core exit thermocouple readings~~	~~A. Core exit thermocouple readings~~	~~Not Applicable~~	~~A. Inadequate RCS heat removal~~	~~Not Applicable~~	~~A. 1. (Site-specific criteria for entry into~~

~~Fuel Clad Barrier~~		~~RCS Barrier~~		~~Containment Barrier~~	
~~LOSS~~	~~POTENTIAL LOSS~~	~~LOSS~~	~~POTENTIAL LOSS~~	~~LOSS~~	~~POTENTIAL LOSS~~
~~greater than (site specific temperature value).~~	~~greater than (site specific temperature value).~~ ~~OR~~ ~~B.~~ ~~Inadequate RCS heat removal capability via steam generators as indicated by (site specific indications).~~		~~capability via steam generators as indicated by (site specific indications).~~		~~core cooling restoration procedure)~~ ~~AND~~ ~~2.~~ ~~Restoration procedure not effective within 15 minutes.~~
~~3. RCS Activity / Containment Radiation~~		~~3. RCS Activity / Containment Radiation~~		~~3. RCS Activity / Containment Radiation~~	
~~Containment radiation monitor reading greater than .~~ ~~OR~~ ~~(Site specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).~~	~~Not Applicable~~	~~A.~~ ~~Containment radiation monitor reading greater than (site specific value)..~~	~~Not Applicable~~	~~Not Applicable~~	~~A.~~ ~~Containment radiation monitor reading greater than (site specific value).~~
~~4. Containment Integrity or Bypass~~		~~4. Containment Integrity or Bypass~~		~~4. Containment Integrity or Bypass~~	
~~Not Applicable~~	~~Not Applicable~~	~~Not Applicable~~	~~Not Applicable~~	~~A.~~ ~~Containment isolation is~~	~~A.~~ ~~Containment pressure~~

~~Fuel Clad Barrier~~		~~RCS Barrier~~		~~Containment Barrier~~	
~~LOSS~~	~~POTENTIAL LOSS~~	~~LOSS~~	~~POTENTIAL LOSS~~	~~LOSS~~	~~POTENTIAL LOSS~~
				~~required~~ ~~AND~~ ~~EITHER of the following:~~ ~~Containment integrity has been lost based on Emergency Director judgment.~~ ~~OR~~ ~~UNISOLABLE pathway from the containment to the environment exists.~~ ~~OR~~ ~~B.~~ ~~Indications of RCS leakage outside of containment.~~	~~greater than (site-specific value)~~ ~~OR~~ ~~B.~~ ~~Explosive mixture exists inside containment~~ ~~OR~~ ~~C. 1.~~ ~~Containment pressure greater than (site-specific pressure setpoint)~~ ~~AND~~ ~~2. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.~~
~~5. Other Indications~~		~~5. Other Indications~~		~~5. Other Indications~~	
~~A. (site-specific as applicable)~~	~~A. (site-specific as applicable)~~	~~A. (site-specific as applicable)~~	~~A. (site-specific as applicable)~~	~~A. (site-specific as applicable)~~	~~A. (site-specific as applicable)~~
~~6. Emergency Director Judgment~~		~~6. Emergency Director Judgment~~		~~6. Emergency Director Judgment~~	
~~A. ANY condition in the opinion of the Emergency Director that indicates Loss~~	~~A. ANY condition in the opinion of the Emergency Director that indicates~~	~~A. ANY condition in the opinion of the Emergency Director that indicates Loss of~~	~~A. ANY condition in the opinion of the Emergency Director that indicates~~	~~A. ANY condition in the opinion of the Emergency Director that indicates Loss of~~	~~A. ANY condition in the opinion of the Emergency Director that indicates~~

~~Fuel Clad Barrier~~		~~RCS Barrier~~		~~Containment Barrier~~	
~~LOSS~~	~~POTENTIAL LOSS~~	~~LOSS~~	~~POTENTIAL LOSS~~	~~LOSS~~	~~POTENTIAL LOSS~~
~~of the Fuel Clad Barrier.~~	~~Potential Loss of the Fuel Clad Barrier.~~	~~the RCS Barrier.~~	~~Potential Loss of the RCS Barrier.~~	~~the Containment Barrier.~~	~~Potential Loss of the Containment Barrier.~~

~~Basis Information For~~
~~PWR EAL Fission Product Barrier Table 9-F-~~
~~Developer Notes:~~
~~Threshold Parameters and Values~~

~~Each PWR owner's group has developed a methodology for guiding the development and implementation of EOPs (i.e., assessing plant parameters, and determining and prioritizing operator actions). Many of the thresholds contained in the PWR EAL Fission Product Barrier Table reflect conditions that are specifically addressed in EOPs (e.g., a loss of heat removal capability by the steam generators). When developing a site-specific threshold, developers should use the parameters and values specified within their EOPs that align with the condition described by the generic threshold and basis, and related developer notes. This approach will ensure consistency between the site-specific EOPs and emergency classification scheme, and thus facilitate more timely and accurate classification assessments.~~

~~In support of EOP development and implementation, the Westinghouse Owners Group (WOG) developed a defined set of Critical Safety Functions as part of their Emergency Response Guidelines. The WOG approach structures EOPs to maintain and/or restore these Critical Safety Functions, and to do so in a prioritized and systematic manner. The WOG Critical Safety Functions are presented below.~~

~~Subcriticality~~
~~Core Cooling~~
~~Heat Sink~~
~~RCS Integrity~~
~~Containment~~
~~RCS Inventory~~

~~The WOG ERGs provide a methodology for monitoring the status of the Critical Safety Functions and classifying the significance of a challenge to a function; this methodology is referred to as the Critical Safety Function Status Trees (CSFSTs). For plants that have implemented the WOG ERGs, the guidance in NEI 99-01 allows for use of certain CSFST assessment results as EALs and fission product barrier loss/potential loss thresholds. In this manner, an emergency classification assessment may flow directly from a CSFST assessment. It is important to understand that the CSFSTs are evaluated using plant parameters, and that they are simply a vendor-specific method for collectively evaluating a set of parameters for purposes of driving emergency operating procedure usage. For the emergency conditions of interest, the generic thresholds within the PWR EAL Fission Product Barrier Table specify the plant parameters that define a potential loss or loss of a fission product barrier; however, as described in the associated Developer Notes, a CSFST terminus may be used as well. For this reason, inclusion of the CSFST-related thresholds would be redundant to the parameter-based thresholds for plants that employ the WOG ERGs.~~

~~Sites that employ the WOG ERGs may, at their discretion, include the CSFST-based loss and potential loss thresholds as described in the Developer Notes. Developers at these sites should consult with their classification decision-makers to determine if inclusion would assist with timely and accurate emergency classification. This decision should consider the effects of any site-specific changes to the generic WOG CSFST evaluation logic and setpoints, as well as those arising from user rules applicable to emergency operating procedures (e.g., exceptions to procedure entry or transition due to specific accident conditions or loss of a support system).~~

~~The CSFST thresholds may be addressed in one of 3 ways:~~

~~1) Not incorporated; thresholds will use parameters and values as discussed in the Developer Notes.~~

~~2) Incorporated along with parameter and value thresholds (e.g., a fuel clad loss would have 2 thresholds such as "CETs > 1200°F" and "Core Cooling Red entry conditions met".~~

~~3) Used in lieu of parameters and values for all thresholds.~~

~~With one exception, if a decision is made to include the CSFST-based thresholds, then all such allowed thresholds must be used in the table (e.g., it is not permissible to use only the C Orange terminus as a potential loss of the fuel clad barrier threshold and disregard all other CSFST-based thresholds). The one exception is the RCS Integrity (P) CSFST. Because of the complexity of the P Red decision point that relies on an assessment a pressure-temperature curve, a P Red condition may be used as an RCS potential loss threshold without the need to incorporate the other CSFST-based thresholds.~~

~~**PWR FUEL CLAD BARRIER THRESHOLDS:**~~

~~The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.~~

~~**RCS or SG Tube Leakage**~~

~~There is no Loss threshold associated with RCS or SG Tube Leakage.~~

~~Potential Loss 1.A~~

~~This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.~~

~~**Developer Notes:**~~

~~Potential Loss 1.A~~

~~Enter the site-specific reactor vessel water level value(s) used by EOPs to identify a degraded core cooling condition (e.g., requires prompt restoration action). The reactor vessel level that corresponds to approximately the top of active fuel may also be used.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the reactor vessel level(s) used for the Core Cooling Orange Path (including dependencies upon the status of RCPs, if applicable).~~

~~Westinghouse ERG Plants~~

~~Developers should consider including a threshold the same as, or similar to, "Core Cooling Orange entry conditions met" in accordance with the guidance at the front of this section. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.~~

~~**PWR FUEL CLAD BARRIER THRESHOLDS:**~~

~~Meeting this threshold results in a Site Area Emergency because this threshold is identical to RCS Barrier Potential Loss threshold 2.A; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.~~~~Potential Loss 1.A~~

~~This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.~~

~~**Developer Notes:**~~

~~Potential Loss 1.A~~

~~Enter the site-specific reactor vessel water level value(s) used by EOPs to identify a degraded core cooling condition (e.g., requires prompt restoration action). The reactor vessel level that corresponds to approximately the top of active fuel may also be used.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the reactor vessel level(s) used for the Core Cooling Orange Path (including dependencies upon the status of RCPs, if applicable).~~

~~Westinghouse ERG Plants~~

~~Developers should consider including a threshold the same as, or similar to, "Core Cooling Orange entry conditions met" in accordance with the guidance at the front of this section. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.~~

~~**Developer Notes:**~~

~~Some site-specific EOPs and/or EOP user guidelines may establish decision-making criteria concerning the number or other attributes of thermocouple readings necessary to drive actions (e.g., 5 CETs reading greater than 1,200°F is required before transitioning to an inadequate core cooling procedure). To maintain consistency with EOPs, these decision-making criteria may be used in the core exit thermocouple reading thresholds.~~

~~Loss 2.A~~

~~Enter a site-specific temperature value that corresponds to significant in-core superheating of reactor coolant. 1,200°F may also be used.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Core Cooling Red Path.~~

~~Potential Loss 2.A~~

~~Enter a site-specific temperature value that corresponds to core conditions at the onset of heat-induced cladding damage (e.g., the temperature allowing for the formation of superheated steam assuming that the RCS is intact). 700°F may also be used.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Core Cooling Orange Path.~~

~~Potential Loss 2.B~~

~~Enter the site-specific parameters and values that define an extreme challenge to the ability to remove heat from the RCS via the steam generators. These will typically be parameters and values that would require operators to take prompt action to address this condition.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Heat Sink Red Path.~~

~~Westinghouse ERG Plants~~

~~As a loss indication, developers should consider including a threshold the same as, or similar to, "Core Cooling Red entry conditions met" in accordance with the guidance at the front of this section.~~

~~PWR FUEL CLAD BARRIER THRESHOLDS:~~

~~As a potential loss indication, developers should consider including a threshold the same as, or similar to, "Core Cooling Orange entry conditions met" in accordance with the guidance at the front of this section.~~

~~As a potential loss indication, developers should consider including a threshold the same as, or similar to, "Heat Sink Red entry conditions met" in accordance with the guidance at the front of this section.~~

~~RCS Activity / Containment Radiation~~

~~Loss 3.A~~

~~The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.~~

~~The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold 3.A since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.~~

~~Loss 3.B~~

~~This threshold indicates that RCS radioactivity concentration is greater than 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.~~

~~There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.~~

~~Developer Notes:~~

~~Loss 3.A~~

~~The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS radioactivity concentration equal to 300 µCi/gm dose equivalent I-131, into the containment atmosphere.~~

~~PWR FUEL CLAD BARRIER THRESHOLDS:~~

~~Loss 3.B~~

~~Threshold values should be determined assuming RCS radioactivity concentration equals 300 µCi/gm dose equivalent I-131. Other site-specific units may be used (e.g., µCi/cc).~~

~~Depending upon site-specific capabilities, this threshold may have a sample analysis component and/or a radiation monitor reading component.~~

~~Add this paragraph (or similar wording) to the Basis if the threshold includes a sample analysis component, "It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications."~~

~~Containment Integrity or Bypass~~

~~**Not Applicable** (included for numbering consistency)~~

~~Other Indications~~

~~Loss and/or Potential Loss 5.A~~

~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Fuel Clad barrier based on plant-specific design characteristics not considered in the generic guidance.~~

~~Developer Notes:~~

~~Loss and/or Potential Loss 5.A~~

~~Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.~~

~~Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.~~

~~Emergency Director Judgment~~

~~Loss 6.A~~

~~This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost.~~

~~**PWR FUEL CLAD BARRIER THRESHOLDS:**~~

~~Potential Loss 6.A~~

~~This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.~~

~~**Developer Notes:**~~

~~None~~

~~PWR RCS BARRIER THRESHOLDS:~~

~~The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.~~

~~RCS or SG Tube Leakage~~

~~Loss 1.A~~

~~This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.~~

~~This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary side (i.e., steam generator tube leakage) or outside of containment.~~

~~A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.~~

~~Potential Loss 1.A~~

~~This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level.~~

~~This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary side (i.e., steam generator tube leakage) or outside of containment.~~

~~If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.~~

~~Potential Loss 1.B~~

~~This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).~~

~~PWR RCS BARRIER THRESHOLDS:~~

~~Developer Notes:~~

~~Loss 1.A~~

~~Actuation of the ECCS may also be referred to as Safety Injection (SI) actuation or other appropriate site-specific term.~~

~~Potential Loss 1.A~~

~~Depending upon charging pump flow capacities and RCS volume control parameters, developers may use an RCS leak rate value of 50 gpm, or an appropriate site-specific value, as an alternate Potential Loss threshold. If used, the threshold wording should reflect that the determination of the leak rate value excludes normal reductions in RCS inventory (e.g., by the letdown system or RCP seal leakoff).~~

~~Potential Loss 1.B~~

~~Enter the site-specific indications that define an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized). These will typically be parameters and values that would require operators to take prompt action to address a pressurized thermal shock condition. Developers should also determine if the threshold needs to reflect any dependencies used as EOP transition/entry decision points or condition validation criteria (e.g., an EOP used to respond to an excessive RCS cooldown may not be entered or immediately exited if RCS pressure is below a certain value).~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the RCS Integrity Red Path. Because of the complexity of certain decision points within the Red Path of this CSFST, developers at these plants may elect to not include the specific parameters and values, and instead follow the guidance below.~~

~~Westinghouse ERG Plants~~

~~As a potential loss indication, developers should consider including a threshold the same as, or similar to, "RCS Integrity Red entry conditions met" in accordance with the guidance at the front of this section. As noted above, developers should ensure that the threshold wording reflects any EOP transition/entry decision points or condition validation criteria. For example, a threshold might read "RCS Integrity (P) Red entry conditions met with RCS pressure > 300 psig."~~

~~Inadequate Heat Removal~~

~~There is no Loss threshold associated with Inadequate Heat Removal.~~

~~PWR RCS BARRIER THRESHOLDS:~~

~~Potential Loss 2.A~~

~~This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.~~

~~Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold 2.B; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.~~

~~Developer Notes:~~

~~Potential Loss 2.A~~

~~Enter the site-specific parameters and values that define an extreme challenge to the ability to remove heat from the RCS via the steam generators. These will typically be parameters and values that would require operators to take prompt action to address this condition.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Heat Sink Red Path.~~

~~Westinghouse ERG Plants~~

~~Developers should consider including a threshold the same as, or similar to, "Heat Sink Red entry conditions met" in accordance with the guidance at the front of this section.~~

~~RCS Activity / Containment Radiation~~

~~Loss 3.A~~

~~The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold 3.A since it indicates a loss of the RCS Barrier only.~~

~~There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.~~

~~PWR RCS BARRIER THRESHOLDS:~~

~~Developer Notes:~~

~~Loss 3.A~~

~~The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS activity at Technical Specification allowable limits, into the containment atmosphere. Using RCS activity at Technical Specification allowable limits aligns this threshold with IC SU3. Also, RCS activity at this level will typically result in containment radiation levels that can be more readily detected by containment radiation monitors, and more readily differentiated from those caused by piping or component "shine" sources. If desired, a plant may use a lesser value of RCS activity for determining this value.~~

~~In some cases, the site-specific physical location and sensitivity of the containment radiation monitor(s) may be such that radiation from a cloud of released RCS gases cannot be distinguished from radiation emanating from piping and components containing elevated reactor coolant activity. If so, refer to the Developer Notes for Loss/Potential Loss 5.A and determine if an alternate indication is available.~~

~~Containment Integrity or Bypass~~

~~**Not Applicable** (included for numbering consistency)~~

~~Other Indications~~

~~Loss and/or Potential Loss 5.A~~

~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the RCS barrier based on plant-specific design characteristics not considered in the generic guidance.~~

~~Developer Notes:~~

~~Loss and/or Potential Loss 5.A~~

~~Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.~~

~~Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.~~

~~PWR RCS BARRIER THRESHOLDS:~~

~~Emergency Director Judgment~~

~~Loss 6.A~~

~~This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is lost.~~

~~Potential Loss 6.A~~

~~This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.~~

~~Developer Notes:~~

~~None~~

~~PWR CONTAINMENT BARRIER THRESHOLDS:~~

~~The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.~~

~~RCS or SG Tube Leakage~~

~~Loss 1.A~~

~~This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss 1.A and Loss 1.A, respectively. This condition represents a bypass of the containment barrier.~~

~~FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably [*part of the FAULTED definition*] and the faulted steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.~~

~~The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU3 for the fuel clad barrier (i.e., RCS activity values) and IC SU4 for the RCS barrier (i.e., RCS leak rate values).~~

~~This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.~~

~~Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.~~

~~PWR CONTAINMENT BARRIER THRESHOLDS:~~

~~Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category A ICs.~~

~~The emergency classification levels resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.~~

	~~Affected SG is FAULTED Outside of Containment?~~	
~~P-to-S Leak Rate~~	~~Yes~~	~~No~~
~~Less than or equal to 25 gpm (or other value per SU4 Developer Notes)~~	~~No classification~~	~~No classification~~
~~Greater than 25 gpm (or other value per SU4 Developer Notes)~~	~~Unusual Event per SU4~~	~~Unusual Event per SU4~~
~~Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)~~	~~Site Area Emergency per FS1~~	~~Alert per FA1~~
~~Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)~~	~~Site Area Emergency per FS1~~	~~Alert per FA1~~

~~There is no Potential Loss threshold associated with RCS or SG Tube Leakage.~~
~~Developer Notes:~~
~~Loss-1.A~~

~~A steam generator power operated relief valve may also be referred to as an atmospheric steam dump valve or other appropriate site specific term.~~

~~Developers may include an additional site-specific threshold(s) to address prolonged steam releases necessitated by operational considerations if AOPs or EOPs could require that a leaking or RUPTURED steam generator be used to support plant cooldown.~~
~~Developers may wish to consider incorporating the above table into user aids (e.g., a wallboard) or other locations within their basis document.~~

~~PWR CONTAINMENT BARRIER Thresholds:~~

~~Inadequate Heat Removal~~

~~There is no Loss threshold associated with Inadequate Heat Removal.~~

~~Potential Loss 2.A~~

~~This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.~~

~~The restoration procedure is considered "effective" if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Director should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.~~

~~Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.~~

~~Developer Notes:~~

~~Some site-specific EOPs and/or EOP user guidelines may establish decision-making criteria concerning the number or other attributes of thermocouple readings necessary to drive actions (e.g., 5 CETs reading greater than 1,200oF is required before transitioning to an inadequate core cooling procedure). To maintain consistency with EOPs, these decision-making criteria may be used in the core exit thermocouple reading thresholds.~~

~~Potential Loss 2.A.1~~

~~Enter site-specific criteria requiring entry into a core cooling restoration procedure or prompt implementation of core cooling restoration actions. A reading of 1,200oF on the CETs may also be used.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters and values used in the Core Cooling Red Path.~~

~~PWR CONTAINMENT BARRIER Thresholds:~~

~~Westinghouse ERG Plants~~

~~Developers should consider including a threshold the same as, or similar to, "Core Cooling Red entry conditions met for 15 minutes or longer" in accordance with the guidance at the front of this section.~~

~~RCS Activity / Containment Radiation~~

~~There is no Loss threshold associated with RCS Activity / Containment Radiation.~~

~~Potential Loss 3.A~~

~~The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.~~

~~NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the emergency classification level to a General Emergency.~~

~~Developer Notes:~~

~~Potential Loss 3.A~~

~~NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, provides the basis for using the 20% fuel cladding failure value. Unless there is a site-specific analysis justifying a different value, the reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad failure into the containment atmosphere.~~

~~Containment Integrity or Bypass~~

~~Loss 4.A~~

~~These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both thresholds 4.A.1 and 4.A.2.~~

~~PWR CONTAINMENT BARRIER Thresholds:~~

~~4.A.1 Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency Director will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).~~

~~Refer to the middle piping run of Figure 9-F-4. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.~~

~~Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.~~

~~Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category A ICs.~~

~~4.A.2 Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.~~

~~Refer to the top piping run of Figure 9-F-4. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.~~

~~The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.~~

~~PWR CONTAINMENT BARRIER Thresholds:~~

~~Leakage between two interfacing liquid systems, by itself, does not meet this threshold.~~

~~Refer to the bottom piping run of Figure 9-F-4. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump or system piping developed a leak that allowed steam/water to enter the Auxiliary Building, then threshold 4.B would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.~~

~~Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. Minor releases may also occur if a~~

~~containment isolation valve(s) fails to close but the containment atmosphere escapes to a closed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category A ICs.~~

~~The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold 1.A.~~

~~Loss 4.B~~

~~Containment sump, temperature, pressure and/or radiation levels will increase if reactor coolant mass is leaking into the containment. If these parameters have not increased, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Increases in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.~~

~~Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.~~

~~Refer to the middle piping run of Figure 9-F-4. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.~~

~~To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold 1.A to be met.~~

~~PWR CONTAINMENT BARRIER Thresholds:~~

~~Potential Loss 4.A~~

~~If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.~~

~~Potential Loss 4.B~~

~~The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a potential loss of the Containment Barrier.~~

~~Potential Loss 4.C~~

~~This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.~~

~~Developer Notes:~~

~~Loss 4.A.1~~

~~Developers may include a list of site-specific radiation monitors to better define this threshold. Expected monitor alarms or readings may also be included.~~

~~Potential Loss 4.A~~

~~The site-specific pressure is the containment design pressure.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, the pressure value in Potential Loss 4.A is that used for the Containment Red Path. If the Containment CSFST contains more than one Red Path due to other dependencies (e.g., status of containment isolation), enter the highest containment pressure value shown on the tree. This is typically the containment design pressure.~~

~~PWR CONTAINMENT BARRIER Thresholds:~~

~~Potential Loss 4.B~~

~~Developers may enter the minimum containment atmospheric hydrogen concentration necessary to support a hydrogen burn (i.e., the lower deflagration limit). A concurrent containment oxygen concentration may be included if the plant has this indication available in the Control Room.~~

~~Potential Loss 4.C~~

~~Enter the site-specific pressure setpoint value that actuates containment pressure control systems (e.g., containment spray). Also enter the site-specific containment pressure control system/equipment that should be operating per design if the containment pressure setpoint is reached. If desired, specific condition indications such as parameter values can also be entered (e.g., a containment spray flow rate less than a certain value).~~

~~This threshold is not applicable to the U.S. Evolutionary Power Reactor (EPR) design.~~

~~Westinghouse ERG Plants~~

~~As a potential loss indication, developers should consider including a threshold the same as, or similar to, "Containment Red entry conditions met" in accordance with the guidance at the front of this section.~~

~~Other Indications~~

~~Loss and/or Potential Loss 5.A~~

~~This subcategory addresses other site-specific thresholds that may be included to indicate loss or potential loss of the Containment barrier based on plant-specific design characteristics not considered in the generic guidance.~~

~~Developer Notes:~~

~~Loss and/or Potential Loss 5.A~~

~~If site emergency operating procedures provide for venting of the containment as a means of preventing catastrophic failure, a Loss threshold should be included for the containment barrier. This threshold would be met as soon as such venting is IMMINENT. Containment venting as part of recovery actions is classified in accordance with the radiological effluent ICs.~~

~~Developers should determine if other reliable indicators exist to evaluate the status of this fission product barrier (e.g., review accident analyses described in the site Final Safety Analysis Report, as updated). The goal is to identify any unique or site-specific indications that will promote timely and accurate assessment of barrier status.~~

~~PWR CONTAINMENT BARRIER Thresholds:~~

~~Any added thresholds should represent approximately the same relative threat to the barrier as the other thresholds in this column. Basis information for the other thresholds may be used to gauge the relative barrier threat level.~~

~~Emergency Director Judgment~~

~~Loss 6.A~~

~~This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is lost.~~

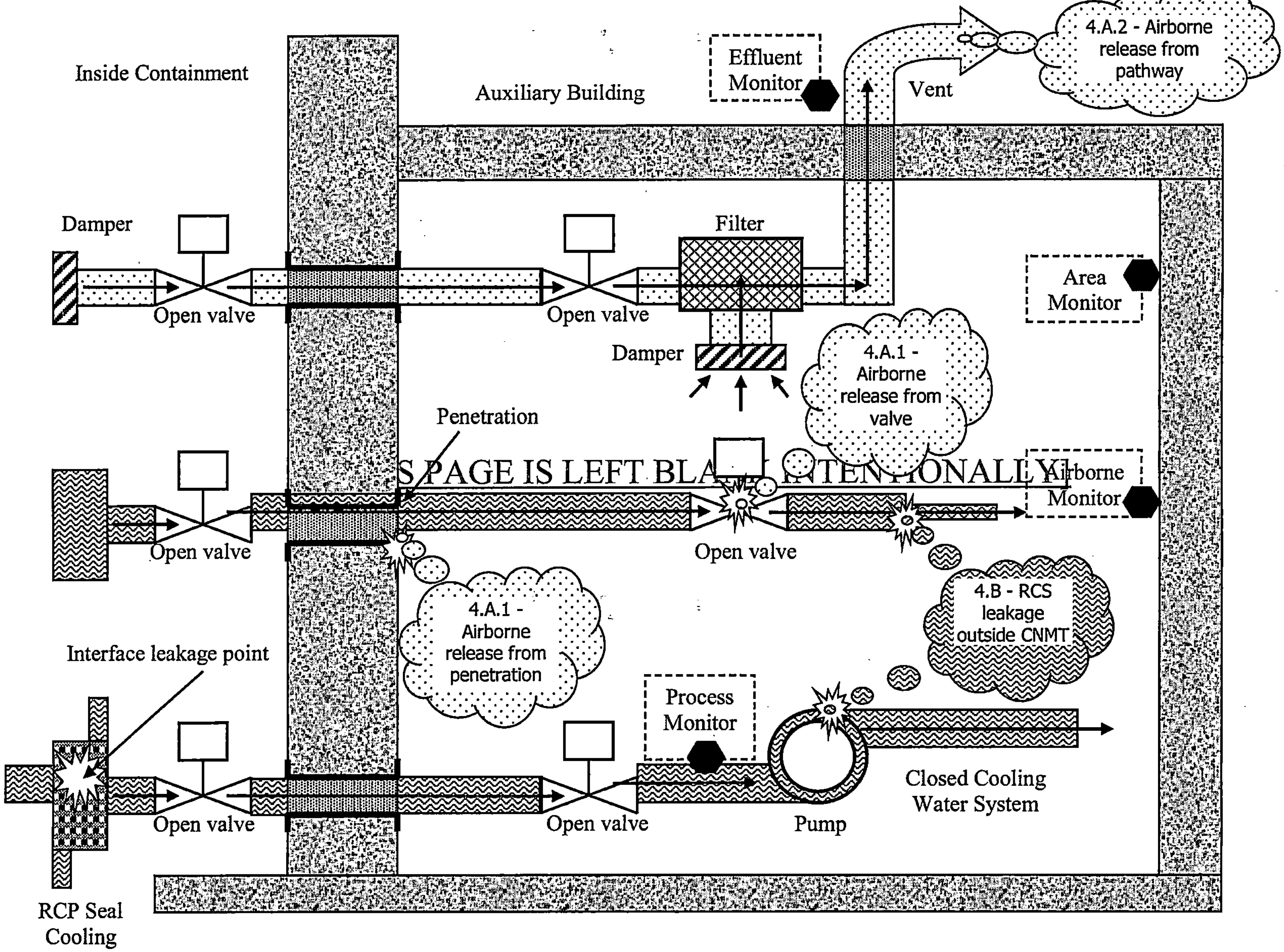
~~Potential Loss 6.A~~

~~This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.~~

~~Developer Notes:~~

~~None~~

~~Figure 9-F-4: PWR Containment Integrity or Bypass Examples~~



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10 HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

~~Table H-1: Recognition Category "H" Initiating Condition Matrix~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
~~**HU1** Confirmed SECURITY CONDITION or threat.~~ ~~*Op. Modes: All*~~	~~**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.~~ ~~*Op. Modes: All*~~	~~**HS1** HOSTILE ACTION within the PROTECTED AREA.~~ ~~*Op. Modes: All*~~	~~**HG1** HOSTILE ACTION resulting in loss of physical control of the facility.~~ ~~*Op. Modes: All*~~
~~**HU2** Seismic event greater than OBE levels.~~ ~~*Op. Modes: All*~~	·	·	·
~~**HU3** Hazardous event.~~ ~~*Op. Modes: All*~~	·	·	·
~~**HU4** FIRE potentially degrading the level of safety of the plant.~~ ~~*Op. Modes: All*~~	·	·	·
·	~~**HA5** Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.~~ ~~*Op. Modes: All*~~	·	
·	~~**HA6** Control Room evacuation resulting in transfer of plant control to alternate locations.~~ ~~*Op. Modes: All*~~	~~**HS6** Inability to control a key safety function from outside the Control Room.~~ ~~*Op. Modes: All*~~	·

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
~~HU7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE.~~ *~~Op. Modes: All~~*	~~HA7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.~~ *~~Op. Modes: All~~*	~~HS7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.~~ *~~Op. Modes: All~~*	~~HG7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.~~ *~~Op. Modes: All~~*

HU1

ECL: Notification of Unusual Event

Initiating Condition: Confirmed SECURITY CONDITION or threat.

Operating Mode Applicability: All

Emergency Action Levels:

~~Example Emergency Action Levels: (1 or 2 or 3)~~

HU1.1 A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by ~~the (site-specific security shift supervision).~~DAEC Security Shift Supervision.

HU1.2 Notification of a credible security threat directed at ~~the site~~DAEC.

HU1.3 A validated notification from the NRC providing information of an aircraft threat.

Definitions:

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non--terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, ~~including the~~including the ECCS. These systems are classified as safety–related.

Basis:

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR ~~§~~-73.71 or 10-CFR-~~§~~ 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between DAEC Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and ~~ORO~~offsite response organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

EAL HU1.1 references ~~(site-specific security shift supervision)~~DAEC Security Shift Supervision because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.390 information.

EAL HU1.2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with Abnormal Operating Procedure (AOP) 914, Security Events. ~~(site-specific procedure).~~

EAL HU1.3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with ~~(site-specific procedure)~~ Abnormal Operating Procedure (AOP) 914, Security Events.

Emergency plans and implementing procedures are public documents; therefore, EALs ~~should~~ do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information ~~should be~~is contained in ~~non-public documents such as~~ the Security Plan.

Escalation of the emergency classification level would be via IC HA1.

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~The (site-specific procedure) is the procedure(s) used by Control Room and/or Security personnel to determine if a security threat is credible, and to validate receipt of aircraft threat information.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~ECL Assignment Attributes: 3.1.1.A~~

HU2

ECL: Notification of Unusual Event

Initiating Condition: Seismic event greater than OBE levels.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

HU2.1 Seismic event greater than Operating Basis Earthquake (OBE) as indicated by~~:~~

receipt of the Amber Operating Basis Earthquake Light and the wailing seismic alarm on 1C35.

~~(1) (site-specific indication that a seismic event met or exceeded OBE limits)~~

Definitions:

DESIGN BASIS EARTHQUAKE (DBE): A DBE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.

OPERATING BASIS EARTHQUAKE (OBE): An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.

~~None~~

Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE)~~[1]~~. An earthquake greater than an OBE but less than a ~~Safe Shutdown~~Design Basis Earthquake (~~SSE~~DBE)~~[2]~~ should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event~~(e.g., typical lateral accelerations are in excess of 0.08g)~~. The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the

~~[1] An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.~~

~~[2] An SSE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.~~

USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

OBE events are detected in accordance with AOP 901. The OBE is associated with a peak horizontal acceleration of ± 0.06g.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or ~~SA9~~SA8.

~~**Developer Notes:**~~

~~This "site-specific indication that a seismic event met or exceeded OBE limits" should be based on the indications, alarms and displays of site-specific seismic monitoring equipment.~~

~~Indications described in the EAL should be limited to those that are immediately available to Control Room personnel and which can be readily assessed. Indications available outside the Control Room and/or which require lengthy times to assess (e.g., processing of scratch plates or recorded data) should not be used. The goal is to specify indications that can be assessed within 15 minutes of the actual or suspected seismic event.~~

~~For sites that do not have readily assessable OBE indications within the Control Room, developers should use the following alternate EAL (or similar wording).~~

~~(1) a. Control Room personnel feel an actual or potential seismic event.~~

~~**AND**~~

~~b. The occurrence of a seismic event is confirmed in manner deemed appropriate by the Shift Manager or Emergency Director.~~

~~The EAL 1.b statement is included to ensure that a declaration does not result from felt vibrations caused by a non-seismic source (e.g., a dropped heavy load). The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration. It is recognized that this alternate EAL wording may cause a site to declare an Unusual Event while another site, similarly affected but with readily assessable OBE indications in the Control Room, may not.~~

~~The above alternate wording may also be used to develop a compensatory EAL for use during periods when a seismic monitoring system capable of detecting an OBE is out of service for maintenance or repair.~~

~~ECL Assignment Attributes: 3.1.1.A~~

HU3

ECL: Notification of Unusual Event

Initiating Condition: Hazardous events

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example Emergency Action Levels:** (1 or 2 or 3 or 4 or 5 or 6)~~

Note: EAL HU3.4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

HU3.1 A tornado strike within the PROTECTED AREA.

HU3.2 Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.

HU3.3 Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).

HU3.4 A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

~~HU3.5 (Site-specific list of natural or technological hazard events)6~~

Definitions:

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety--related.~~

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL HU3.1 addresses a tornado striking (touching down) within the Protected Area.

EAL HU3.2 addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

EAL HU3.3 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

EAL HU3.4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

~~EAL HU3.5 addresses (site-specific description).~~

Escalation of the emergency classification level would be based on ICs in Recognition Categories ~~A~~R, F, S or C.

~~**Developer Notes:**~~

~~The "Site-specific list of natural or technological hazard events" should include other events that may be a precursor to a more significant event or condition, and that are appropriate to the site location and characteristics.~~

~~Notwithstanding the events specifically included as EALs above, a "Site-specific list of natural or technological hazard events" need not include short-lived events for which the extent of the damage and the resulting consequences can be determined within a relatively short time frame. In these cases, a damage assessment can be performed soon after the event, and the plant staff will be able to identify potential or actual impacts to plant systems and structures. This will enable prompt definition and implementation of compensatory or corrective measures with no appreciable increase in risk to the public.~~

~~To the extent that a short-lived event does cause immediate and significant damage to plant systems and structures, it will be classifiable under the Recognition Category F, S and C ICs and EALs. Events of lesser impact would be expected to cause only small and localized damage. The consequences from these types of events are adequately assessed and addressed in accordance with Technical Specifications. In addition, the occurrence or effects of the event may be reportable under the requirements of 10 CFR 50.72.~~

~~ECL Assignment Attributes: 3.1.1.A and 3.1.1.C~~

HU4

ECL: Notification of Unusual Event

Initiating Condition: FIRE potentially degrading the level of safety of the plant.

Operating Mode Applicability: All

Emergency Action Levels:

~~**Example Emergency Action Levels:** (1 or 2 or 3 or 4)~~

Notes:

- The Emergency Director should declare the ~~Unusual Event~~event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

HU4.1 a. A FIRE is **NOT** extinguished within 15-minutes of **ANY** of the following FIRE detection indications:
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

b. The FIRE is located within **ANY** ~~of the following~~Table H-1 plant rooms or areas~~:~~

~~(site-specific list of plant rooms or areas)~~

HU4.2 a. Receipt of a single fire alarm ~~(i.e.,~~with no other indications of a FIRE~~)~~.

AND

b. The FIRE is located within **ANY** ~~of the following~~Table H-1 plant rooms or areas

~~(site-specific list of plant rooms or areas)~~

AND

c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.

HU4.3 A FIRE within the plant or ISFSI ~~[*for plants with an ISFSI outside the plant Protected Area*]~~PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

HU4.4 A FIRE within the plant or ISFSI ~~[*for plants with an ISFSI outside the plant Protected Area*]~~PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Table H-1 Fire Areas
~~**Area**~~

- 1G31 DG and Day Tank Rooms~~,~~
- 1G21 DG and Day Tank Rooms~~,~~
- Battery Rooms~~,~~
- Essential Switchgear Rooms~~,~~
- Cable Spreading Room
- Torus Room
- ~~,~~Intake Structure~~,~~
- Pumphouse
- Drywell~~,~~
- Torus
- NE, NW, SE Corner Rooms~~,~~
- HPCI Room~~,~~
- RCIC Room~~,~~
- RHR Valve Room~~,~~
- North CRD Area~~,~~
- South CRD Area~~,~~
- CSTs
- Control Building~~,~~
- Remote Shutdown Panel 1C388 Area~~,~~
- Panel 1C55/56 Area~~,~~
- SBGT Room

Definitions:

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

EAL HU4.1

The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

EAL HU4.2

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then EAL HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

EAL HU4.3

In addition to a FIRE addressed by EAL HU4.1 or EAL HU4.2, a FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA. ~~[Sentence for plants with an ISFSI outside the plant Protected Area]~~

EAL HU4.4

If a FIRE within the plant or ISFSI ~~[*for plants with an ISFSI outside the plant Protected Area*]~~ PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Basis-Related Requirements from Appendix R and NFPA-805

Criterion 3 of Appendix A to 10 CFR 50 states in part that "structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

The Nuclear Safety Goal ("NSG") in NFPA 805, Section 1.3.1 states, "The nuclear safety goal is to provide reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition."

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance because a safe shutdown success path, free of fire damage, must be available to meet the nuclear safety goals, objectives and performance criteria for a fire under any plant operational mode or configuration.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). Even though DAEC has adopted the alternate approach provided by NFPA-805 in lieu of the deterministic requirements of Appendix R, the 30-minutes to verify a single alarm as used in EAL HU4.2 is considered a reasonable amount of time to determine if an actual FIRE exists without presenting a challenge to the nuclear safety performance criteria.~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

~~Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."~~

~~When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.~~

~~Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.~~

~~In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.~~

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or ~~SA9~~SA8.

~~**Developer Notes:**~~

~~The "site-specific list of plant rooms or areas" should specify those rooms or areas that contain SAFETY SYSTEM equipment.~~

~~As noted in the EALs and Basis section, include the term ISFSI if the site has an ISFSI outside the plant Protected Area.~~

~~ECL Assignment Attributes: 3.1.1.A~~

HU~~7~~6

ECL: Notification of Unusual Event

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a ~~(~~NO~~)~~UE.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

HU~~7~~6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS ~~safety systems~~ occurs.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety - related.~~

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a NOUE.

HA1

ECL: Alert

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:~~: (1 or 2)~~

HA1.1 A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the ~~(site-specific security shift supervision)~~DAEC Security Shift Supervision.

HA1.2 A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

Definitions:

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

OWNER CONTROLLED AREA: The site property owned by or otherwise under the control of the licensee.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between DAEC Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations, allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR ~~§~~73.71 or 10 CFR ~~§~~50.72.

EAL HA1.1 is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against ~~an~~ the ISFSI ~~that~~ which is located outside the plant PROTECTED AREA.

EAL HA1.2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and ~~ORO~~offsite response organizations are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with ~~(~~Abnormal Operating Procedure (AOP) 914, Security Events~~site-specific procedure).~~s.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.

In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs ~~should~~ do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information ~~should be~~ is contained in ~~non-public documents such as~~ the Security Plan.

Escalation of the emergency classification level would be via IC HS1.

~~Developer Notes:~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the OWNER CONTROLLED AREA.~~

~~ECL Assignment Attributes: 3.1.2.D~~

~~HA5~~

~~**ECL:** Alert~~

~~**Initiating Condition:** Gaseous release impeding access to equipment necessary for normal plant operations, cooldown, or shutdown.~~

~~**Operating Mode Applicability:** All~~

~~**Example Emergency Action Levels:**~~

~~**Note:** If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.~~

~~1 a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas:~~

~~(site-specific list of plant rooms or areas with entry-related mode applicability identified)~~

~~**AND**~~

~~b. Entry into the room or area is prohibited or impeded.~~

~~**Basis:**~~

~~This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.~~

~~An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.~~

~~Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Director's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).~~

~~An emergency declaration is not warranted if any of the following conditions apply.~~

- ~~The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.~~

- ~~The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).~~
- ~~The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).~~
- ~~The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.~~

~~An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness, or even death.~~

~~This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area, or to intentional inerting of containment (BWR only).~~

~~Escalation of the emergency classification level would be via Recognition Category AR, C or F ICs.~~

~~Developer Notes:~~

~~The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.~~

~~The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).~~

~~The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.~~

~~If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.~~

~~ECL Assignment Attributes: 3.1.2.B~~

~~HA6~~HA5

ECL: Alert

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

HA~~6~~5.1 An event has resulted in plant control being transferred from the Control Room to ~~(site-specific remote shutdown panels and local control stations)~~the Remote Shutdown Panel (1C388).

Definitions:

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS~~6~~5.

~~Developer Notes:~~

~~The "site-specific remote shutdown panels and local control stations" are the panels and control stations referenced in plant procedures used to cooldown and shutdown the plant from a location(s) outside the Control Room.~~

~~ECL Assignment Attributes: 3.1.2.B~~

~~HA7~~HA6

ECL: Alert

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Level~~s~~:

HA~~7~~6.1 Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Definitions:

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for an Alert.

HS1

ECL: Site Area Emergency

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: All

~~Example Emergency~~Emergency Action Levels:

HS1.1 A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the ~~(site-specific security shift supervision)~~DAEC Security Shift Supervision.

Definitions:

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between DAEC Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize ~~ORO~~offsite response organization resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to a HOSTILE ACTION directed at ~~an~~ the ISFSI PROTECTED AREA which is located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR ~~§~~ 73.71 or 10 CFR ~~§~~ 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs ~~should~~ do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information ~~should be~~is contained in ~~non-public documents such as~~ the Security Plan.

Escalation of the emergency classification level would be via IC HG1.

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the PROTECTED AREA.~~

~~ECL Assignment Attributes: 3.1.3.D~~

~~HS6~~HS5

ECL: Site Area Emergency

Initiating Condition: Inability to control a key safety function from outside the Control Room.

Operating Mode Applicability: All

~~Example~~ Emergency Action Levels:

Note: The Emergency Director should declare the ~~Site Area Emergency~~event promptly upon determining that ~~(site-specific number~~ the applicable time~~of~~20 ~~minutes)~~ has been exceeded, or will likely be exceeded.

HS5~~6~~.1 a. An event has resulted in plant control being transferred from the Control Room to ~~(site-specific remote shutdown panels and control stations)~~ the Remote Shutdown Panel (1C388).

AND

b. Control of **ANY** of the following key safety functions is not reestablished within ~~(site-specific number of~~20 minutes~~)~~.

- Reactivity control
- ~~Core cooling [*PWR*] /~~ RPV water level ~~[*BWR*]~~
- RCS heat removal

Definitions:

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the ~~remote safe shutdown location(s)~~Remote Shutdown Panel (1C388 ~~is~~) is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within ~~(the site-specific time for transfer)~~ ~~3~~20 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

AOP 915, "Shutdown Outside Control Room" provides the following CAUTION – "*For Control Room evacuation as the result of a fire, transfer of control at panels 1C388, 1C389, 1C390, 1C391, ~~1C392~~and 1C392 is required to be completed within 20 minutes.*"

Escalation of the emergency classification level would be via IC FG1 or CG1.

~~**Developer Notes:**~~

~~The "site-specific remote shutdown panels and local control stations" are the panels and control stations referenced in plant procedures used to cooldown and shutdown the plant from a location(s) outside the Control Room.~~

~~The "site-specific number of minutes" is the time in which plant control must be (or is expected to be) reestablished at an alternate location as described in the site-specific fire response analyses. Absent a basis in the site-specific analyses, 15 minutes should be used. Another time period may be used with appropriate basis/justification.~~

~~ECL Assignment Attributes: 3.1.3.B~~

~~HS7~~HS6

ECL: Site Area Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.

Operating Mode Applicability: All

~~Example~~ Emergency Action Level~~s~~:

HS~~7~~6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Definitions:

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non–terrorism–based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a Site Area Emergency.

HG1

ECL: General Emergency

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility.

Operating Mode Applicability: All

~~Example~~ Emergency Action Level~~s~~:

HG1.1 a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the ~~(site-specific security shift supervision)~~DAEC Security Shift Supervision.

AND

b. **EITHER** of the following has occurred:

1. **ANY** of the following safety functions cannot be controlled or maintained.

- Reactivity control
- ~~Core cooling [*PWR*] /~~ RPV water level ~~[*BWR*]~~
- RCS heat removal

OR

2. Damage to spent fuel has occurred or is IMMINENT.

Definitions:

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non–terrorism–based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

Timely and accurate communications between the DAEC Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

Emergency plans and implementing procedures are public documents; therefore, EALs ~~should~~ do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information ~~should be~~is contained in ~~non-public documents such as~~ the Security Plan.

~~Developer Notes:~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the PROTECTED AREA.~~

~~ECL Assignment Attributes: 3.1.4.D~~

~~HG7~~HG6

ECL: General Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.

Operating Mode Applicability: All

~~Example~~ Emergency Action Level~~s~~:

HG~~7~~6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Definitions:

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a General Emergency.

11 SYSTEM MALFUNCTION ICS/EALS

~~Table S-1: Recognition Category "S" Initiating Condition Matrix~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
~~**SU1** Loss of all offsite AC power capability to emergency buses for 15 minutes or longer.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~	~~**SA1** Loss of all but one AC power source to emergency buses for 15 minutes or longer.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~	~~**SS1** Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~	~~**SG1** Prolonged loss of all offsite and all onsite AC power to emergency buses.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~
~~**SU2** UNPLANNED loss of Control Room indications for 15 minutes or longer.~~ ~~*Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown1, 2, 3, 4*~~	~~**SA2** UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.~~ ~~*Op. Modes: 1, 2, 3, 4 Power Operation, Startup, Hot Standby, Hot Shutdown*~~	.	.
~~**SU3** Reactor coolant activity greater than Technical Specification allowable limits.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~	.	.	.
~~**SU4** RCS leakage for 15 minutes or longer.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~	.	.	.
~~**SU5** Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor.~~ ~~*Op. Modes: Power Operation1*~~	~~**SA5** Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.~~ ~~*Op. Modes: Power Operation1*~~	~~**SS5** Inability to shutdown the reactor causing a challenge to (core cooling [*PWR*] / RPV water level [*BWR*]) or RCS heat removal.~~ ~~*Op. Modes: Power Operation1*~~	.

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

~~UNUSUAL EVENT~~	~~ALERT~~	~~SITE AREA EMERGENCY~~	~~GENERAL EMERGENCY~~
~~**SU6** Loss of all onsite or offsite communications capabilities.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~			
~~**SU7** Failure to isolate containment or loss of containment pressure control. [*PWR*]~~ ~~*Op. Modes: 1, 2, 3, 4 Power Operation, Startup, Hot Standby, Hot Shutdown*~~			
		~~**SS8** Loss of all Vital DC power for 15 minutes or longer.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~	~~**SG8** Loss of all AC and Vital DC power sources for 15 minutes or longer.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~
	~~**SA9** Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.~~ ~~*Op. Modes: 1, 2, 3, 4Power Operation, Startup, Hot Standby, Hot Shutdown*~~		

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

SU1

ECL: Notification of Unusual Event

Initiating Condition: Loss of ~~all~~ ALL offsite AC power capability to ~~emergency~~ essential buses for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Level~~s~~:

Note: The Emergency Director should declare the ~~Unusual E~~event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

SU1.1 Loss of **ALL** offsite AC power capability to ~~(site-specific emergency buses)~~1A3 **AND** 1A4 buses for 15 minutes or longer.

Definitions:

None

Basis:

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC ~~emergency~~ essential buses. This condition represents a potential reduction in the level of safety of the plant.

The intent of this EAL is to declare ~~an~~ Notification of Unusual Event when offsite power has been lost and both of the emergency diesel generators have successfully started and energized their respective 4kv essential bus.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the ~~emergency~~ essential buses, whether or not the buses are powered from it.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Escalation of the emergency classification level would be via IC SA1.

~~Developer Notes:~~

~~The "site specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an~~

~~affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~SU2~~SU3

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Levels:

Note: The Emergency Director should declare the ~~Unusual E~~event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

SU3.1 ~~a.~~ An UNPLANNED event results in the inability to monitor one or more of the Table S-1 parameters from within the Control Room for 15 minutes or longer.~~An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.~~

~~Reactor Power~~
~~RPV Water Level~~
~~RPV Pressure~~
~~Primary Containment Pressure~~
~~Suppression Pool Level~~
- ~~Suppression Pool Temperature~~

~~[*BWR parameter list*]~~	~~[*PWR parameter list*]~~
~~Reactor Power~~	~~Reactor Power~~
~~RPV Water Level~~	~~RCS Level~~
~~RPV Pressure~~	~~RCS Pressure~~
~~Primary Containment Pressure~~	~~In-Core/Core Exit Temperature~~
~~Suppression Pool Level~~	~~Levels in at least (site-specific number) steam generators~~
~~Suppression Pool Temperature~~	~~Steam Generator Auxiliary or Emergency Feed Water Flow~~

Table S-1 Safety System Parameters
• Reactor power
• RPV Water Level
• RPV Pressure
• Primary Containment Pressure
• Suppression Pool Level
• Suppression Pool Temperature

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety related.~~

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, ~~core cooling [*PWR*] /~~ RPV level ~~[*BWR*]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for ~~reactor vessel level [PWR] /~~ RPV water level ~~[BWR]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via IC SA~~2~~3.

~~**Developer Notes:**~~

~~In the PWR parameter list column, the "site-specific number" should reflect the minimum number of steam generators necessary for plant cooldown and shutdown. This criterion may also specify whether the level value should be wide-range, narrow-range or both, depending upon the monitoring requirements in emergency operating procedures.~~

~~Developers may specify either pressurizer or reactor vessel level in the PWR parameter column entry for RCS Level.~~

~~The number, type, location and layout of Control Room indications, and the range of possible failure modes, can challenge the ability of an operator to accurately determine, within the time period available for emergency classification assessments, if a specific percentage of indications have been lost. The approach used in this EAL facilitates prompt and accurate emergency classification assessments by focusing on the indications for a selected subset of parameters.~~

~~By focusing on the availability of the specified parameter values, instead of the sources of those values, the EAL recognizes and accommodates the wide variety of indications in nuclear power plant Control Rooms. Indication types and sources may be analog or digital, safety-related or not, primary or alternate, individual meter value or computer group display, etc.~~

~~A loss of plant annunciators will be evaluated for reportability in accordance with 10 CFR 50.72 (and the associated guidance in NUREG-1022), and reported if it significantly impairs the capability to perform emergency assessments. Compensatory measures for a loss of annunciation can be readily implemented and may include increased monitoring of main control boards and more frequent plant rounds by non-licensed operators. Their alerting function notwithstanding, annunciators do not provide the parameter values or specific component status information used to operate the plant, or process through AOPs or EOPs. Based on these considerations, a loss of annunciation is considered to be adequately addressed by reportability criteria, and therefore not included in this IC and EAL.~~

~~With respect to establishing event severity, the response to a loss of radiation monitoring data (e.g., process or effluent monitor values) is considered to be adequately bounded by the requirements of 10 CFR 50.72 (and associated guidance in NUREG-1022). The reporting of this event will ensure adequate plant staff and NRC awareness, and drive the establishment of appropriate compensatory measures and corrective actions. In addition, a loss of radiation monitoring data, by itself, is not a precursor to a more significant event.~~

~~Personnel at sites that have a Failure Modes and Effects Analysis (FMEA) included within the design basis of a digital I&C system should consider the FMEA information when developing their site-specific EALs.~~

~~Due to changes in the configurations of SAFETY SYSTEMS, including associated instrumentation and indications, during the cold shutdown, refueling, and defueled modes, no analogous IC is included for these modes of operation.~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~SU3~~SU4

ECL: Notification of Unusual Event

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Levels: ~~(1 or 2)~~

SU4.1 ~~(Site specific radiation monitor) reading greater than (site specific value).~~ Pretreatment Offgas System (RM-4104) Hi-Hi Radiation Alarm.

SU4.2 Sample analysis indicates that reactor coolant specific activity is greater than 2.0 µCi/gm dose equivalent I-131 for 12 hours or longer~~Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.~~

Definitions:

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

For EAL SU4.1, RM-4104 Hi-Hi Radiation Alarm has been chosen because it is operationally significant, is readily recognizable by the Control Room Operations Staff, and is set at a level corresponding to noble gas release rate, after 30-minute delay and decay of 1 Ci/sec.

For EAL SU4.2, coolant samples exceeding the 2.0 µCi/gm dose equivalent I-131concentration require prompt action by DAEC Technical Specifications and are representative of minor fuel cladding degradation.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category ~~A~~ R ICs.

~~Developer Notes:~~

~~For EAL #1 – Enter the radiation monitor(s) that may be used to readily identify when RCS activity levels exceed Technical Specification allowable limits. This EAL may be developed using different methods and sites should use existing capabilities to address it (e.g., development of new capabilities is not required). Examples of existing methods/capabilities include:~~

- ~~An installed radiation monitor on the letdown system or air ejector.~~
- ~~A hand-held monitor or deployed detector reading with pre-calculated conversion values or readily implementable conversion calculation capability.~~

~~The monitor reading values should correspond to an RCS activity level approximately at Technical Specification allowable limits.~~

~~If there is no existing method/capability for determining this EAL, then it should not be included. IC evaluation will be based on EAL #2.~~

~~For EAL#2 – Developers may reword the EAL to include the reactor coolant activity parameter(s) specified in Technical Specifications and the associated allowable limit(s) (e.g., values for dose equivalent I-131 and gross activity, time-dependent or transient values, etc.). If this approach is selected, all RCS activity allowable limits should be included.~~

~~ECL Assignment Attributes: 3.1.1.A and 3.1.1.B~~

~~SU4~~SU5

ECL: Notification of Unusual Event

Initiating Condition: RCS leakage for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Levels: ~~(1 or 2 or 3)~~

Note: The Emergency Director should declare the ~~Unusual E~~event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

SU5.1 RCS unidentified or pressure boundary leakage greater than ~~(site-specific value)~~10 gpm for 15 minutes or longer.

SU5.2 RCS identified leakage greater than ~~(site-specific value)~~25 gpm for 15 minutes or longer.

SU5.3 Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.

Definitions:

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

EAL SU5.1 and EAL SU5.2 are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications).

EAL SU5.3 addresses a RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These EALs thus apply to leakage into the containment, a secondary-side system ~~(e.g., steam generator tube leakage in a PWR)~~ or a location outside of containment.

The leak rate values for each EAL were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). EAL SU5.1 uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. ~~For PWRs, an emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated). For BWRs, a~~A stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category ~~A~~ R or F.

~~**Developer Notes:**~~

~~EAL #1 – For the site-specific leak rate value, enter the higher of 10 gpm or the value specified in the site's Technical Specifications for this type of leakage.~~

~~EAL #2 – For the site-specific leak rate value, enter the higher of 25 gpm or the value specified in the site's Technical Specifications for this type of leakage.~~

~~For sites that have Technical Specifications that do not specify a leakage type for steam generator tube leakage, developers should include an EAL for tube leakage greater than 25 gpm for 15 minutes or longer.~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~SU5~~SU6

ECL: Notification of Unusual Event

Initiating Condition: Automatic or manual ~~(trip [PWR] /~~ scram ~~[BWR])~~ fails to shutdown the reactor.

Operating Mode Applicability: ~~Power Operation~~1, 2

~~**Note:** A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.~~

~~Example~~ Emergency Action Levels: ~~(1 or 2)~~

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

SU6.1

a. An automatic ~~(trip [PWR] /~~ scram ~~[BWR])~~ did not shutdown the reactor.

AND

b. **ANY** of the following manual actions taken at 1C05 are successful in lowering reactor power below 5% power
- Manual Scram Pushbuttons
- Mode Switch to Shutdown
- Alternate Rod Insertion (ARI)~~A subsequent manual action taken at the reactor control consoles (1C05) is successful in shutting down the reactor.~~

SU6.2 a. A manual ~~trip ([PWR] /~~ scram ~~[BWR])~~ did not shutdown the reactor.

AND

b. **EITHER** of the following:

1. **ANY** of the following subsequent manual actions taken at 1C05 are successful in lowering reactor power below 5% power
   - Manual Scram Pushbuttons
   - Mode Switch to Shutdown
   - Alternate Rod Insertion (ARI)~~A subsequent manual action taken at the reactor control console (1C05)s is successful in shutting down the reactor.~~

OR

2. —A subsequent automatic ~~(trip [PWR] /~~ scram ~~[BWR])~~ is successful in shutting down the reactor.

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(trip [*PWR*] /~~ scram ~~[*BWR*])~~ that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic ~~(trip [*PWR*] /~~ scram ~~[*BWR*])~~ is successful in shutting down the reactor. _This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor ~~(trip [*PWR*] /~~ scram ~~[*BWR*])~~, operators will promptly initiate manual actions at the reactor control console~~s~~ to shutdown the reactor (e.g., initiate a manual reactor ~~(trip [*PWR*] / scram [*BWR*])). If these manual actions are successful in shutting down the reactor, core heat generation will~~ scram quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor ~~(trip [*PWR*] / scram [*BWR*])~~ is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control console~~s~~ to shutdown the reactor (e.g., initiate a manual reactor ~~(trip [*PWR*] / scram [*BWR*])~~ using a different switch). Depending upon several factors, the initial or subsequent effort to manually ~~(trip [*PWR*] / scram [*BWR*])~~ the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor ~~(trip [*PWR*] / scram [*BWR*])~~ signal. If a subsequent manual or automatic ~~(trip [*PWR*] / scram [*BWR*])~~ is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control console~~s~~ is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(trip [*PWR*] / scram [*BWR*])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control console~~s~~".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. ~~[*BWR*]~~

The plant response to the failure of an automatic or manual reactor ~~(trip [PWR] / scram [BWR])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA~~5~~6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA~~5~~6 or FA1, an Unusual Event declaration is appropriate for this event.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the SAFETY SYSTEMS are designed (typically 3 to 5% power). ~~A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.~~

Should a reactor ~~(trip [PWR] / scram [BWR])~~ signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor ~~(trip [PWR] / scram [BWR])~~ and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the ~~(trip [PWR] / scram [BWR])~~ failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

~~**Developer Notes:**~~

~~This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. A PWR with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, then the IC is also applicable in Startup Mode.~~

~~Developers may include site specific EOP criteria indicative of a successful reactor shutdown in~~

~~an EAL statement, the Basis or both (e.g., a reactor power level).~~

~~The term "reactor control consoles" may be replaced with the appropriate site-specific term (e.g., main control boards).~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~SU6~~SU7

ECL: Notification of Unusual Event

Initiating Condition: Loss of ~~all~~ ALL onsite or offsite communications capabilities.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Levels: ~~(1 or 2 or 3)~~

SU7.1 Loss of **ALL** of the following onsite communication methods:

- ~~(site-specific list of communications methods)~~ Plant Operations Radio System
- In-Plant Phone System
- Plant Paging System (Gaitronics)

SU7.2 Loss of **ALL** of the following ~~ORO~~offsite response organization communications methods:

- ~~(site-specific list of communications methods)~~ DAEC All-Call phone
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system
- FTS Phone system

SU7.3 Loss of **ALL** of the following NRC communications methods:

- ~~(site-specific list of communications methods)~~ FTS Phone system
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system

Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to ~~ORO~~offsite response organizations and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

EAL SU7.1 addresses a total loss of the communications methods used in support of routine plant operations.

EAL SU7.2 addresses a total loss of the communications methods used to notify all ~~ORO~~offsite response organizations of an emergency declaration. The ~~ORO~~offsite response organizations referred to here are~~-~~ the State of Iowa, Linn County, and Benton County~~-(see Developer Notes)~~.

EAL SU7.3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

~~**Developer Notes:**~~

~~EAL #1 The "site-specific list of communications methods" should include all communications methods used for routine plant communications (e.g., commercial or site telephones, page-party systems, radios, etc.). This listing should include installed plant equipment and components, and not items owned and maintained by individuals.~~

~~EAL #2 The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to OROs as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. Example methods are ring-down/dedicated telephone lines, commercial telephone lines, radios, satellite telephones and internet-based communications technology.~~

~~In the Basis section, insert the site-specific listing of the OROs requiring notification of an emergency declaration from the Control Room in accordance with the site Emergency Plan, and typically within 15 minutes.~~

~~EAL #3 The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to the NRC as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. These methods are typically the dedicated Emergency Notification System (ENS) telephone line and commercial telephone lines.~~

~~ECL Assignment Attributes: 3.1.1.C~~

~~SU7~~

~~ECL: Notification of Unusual Event~~

~~Initiating Condition: Failure to isolate containment or loss of containment pressure control. [PWR]~~

~~Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown~~

~~Example Emergency Action Levels: (1 or 2)~~

~~1 a. Failure of containment to isolate when required by an actuation signal.~~

~~AND~~

~~b. ALL required penetrations are not closed within 15 minutes of the actuation signal.~~

~~2 a. Containment pressure greater than (site-specific pressure).~~

~~AND~~

~~b. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.~~

~~Basis:~~

~~This IC addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a~~

~~concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.~~

~~For EAL 1, the containment isolation signal must be generated as the result on an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.~~

~~EAL 2 addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays or ice condenser fans) are either lost or performing in a degraded manner.~~

~~This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss~~

~~of either the Fuel Clad or RCS fission product barriers.~~

~~Developer Notes:~~

~~Enter the "site-specific pressure" value that actuates containment pressure control systems (e.g., containment spray). Also enter the site-specific containment pressure control system/equipment that should be operating per design if the containment pressure actuation setpoint is reached. If desired, specific condition indications such as parameter values can also be entered (e.g., a containment spray flow rate less than a certain value).~~

~~EAL #2 is not applicable to the U.S. Evolutionary Power Reactor (EPR) design.~~

~~ECL Assignment Attributes: 3.1.1.A~~

SA1

ECL: Alert

Initiating Condition: Loss of ~~all~~ ALL but one AC power source to ~~emergency~~ essential buses for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Levels:

Note: The Emergency Director should declare the ~~Alert~~ event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

SA1.1 a. AC power capability to ~~(site-specific emergency buses)~~1A3 and 1A4 buses is reduced to a single power source for 15 minutes or longer.

AND

b. ~~Any~~ ~~ANY~~**ANY** additional single power source failure will result in a loss of ~~all~~ **ALL** AC power to SAFETY SYSTEMS.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety related.~~

Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- ~~A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.~~
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of essential~~emergency~~ buses being ~~back-~~fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

Escalation of the emergency classification level would be via IC SS1.

~~Developer Notes:~~

~~For a power source that has multiple generators, the EAL and/or Basis section should reflect the minimum number of operating generators necessary for that source to provide required power to~~

~~an AC emergency bus. For example, if a backup power source is comprised of two generators (i.e., two 50%-capacity generators sized to feed 1 AC emergency bus), the EAL and Basis section must specify that both generators for that source are operating.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~Developers should modify the bulleted examples provided in the basis section, above, as needed to reflect their site-specific plant designs and capabilities.~~

~~The EALs and Basis should reflect that each independent offsite power circuit constitutes a single power source. For example, three independent 345kV offsite power circuits (i.e., incoming power lines) comprise three separate power sources. Independence may be determined from a review of the site-specific UFSAR, SBO analysis or related loss of electrical power studies.~~

~~The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is recognized in AOPs and EOPs, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.2.B~~

~~SA2~~SA3

ECL: Alert

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Level~~s~~:

Note: The Emergency Director should declare the ~~Alert~~ event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

SA3.1 a. An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for 15 minutes or longer~~An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer~~.

Table S-1 Safety System Parameters
• Reactor power
• RPV Water Level
• RPV Pressure
• Primary Containment Pressure
• Suppression Pool Level
• Suppression Pool Temperature

~~Reactor Power~~
~~RPV Water Level~~
~~RPV Pressure~~
~~Primary Containment Pressure~~
~~Suppression Pool Level~~
~~Suppression Pool Temperature~~

~~*[BWR parameter list]*~~	~~*[PWR parameter list]*~~
~~Reactor Power~~	~~Reactor Power~~
~~RPV Water Level~~	~~RCS Level~~
~~RPV Pressure~~	~~RCS Pressure~~
~~Primary Containment Pressure~~	~~In-Core/Core Exit Temperature~~
~~Suppression Pool Level~~	~~Levels in at least (site-specific number) steam generators~~
~~Suppression Pool Temperature~~	~~Steam Generator Auxiliary or Emergency Feed Water Flow~~

AND

b. **ANY** of the Table S-2 transient events are in progress.

Table S-2 Significant Transients
• Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load • Reactor scram • ECCS actuation • Thermal power oscillations greater than 10%

~~of the following transient events in progress.~~

~~Automatic or manual runback greater than 25% thermal reactor power~~

~~Electrical load rejection greater than 25% full electrical load~~

~~Reactor scram [BWR] / trip [PWR]~~

~~ECCS (SI) actuation~~

~~Thermal power oscillations greater than 10% [BWR]~~

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety-related.~~

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, ~~core cooling [*PWR*] /~~ RPV level ~~[*BWR*]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for ~~reactor vessel level [*PWR*] /~~ RPV water level ~~[*BWR*]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via ICs FS1 or IC ~~AS1~~RS1.

~~**Developer Notes:**~~

~~In the PWR parameter list column, the "site-specific number" should reflect the minimum number of steam generators necessary for plant cooldown and shutdown. This criterion may also specify whether the level value should be wide range, narrow range or both, depending upon the monitoring requirements in emergency operating procedures.~~

~~Developers may specify either pressurizer or reactor vessel level in the PWR parameter column entry for RCS Level.~~

~~Developers should consider if the "transient events" list needs to be modified to better reflect site-specific plant operating characteristics and expected responses.~~

~~The number, type, location and layout of Control Room indications, and the range of possible failure modes, can challenge the ability of an operator to accurately determine, within the time period available for emergency classification assessments, if a specific percentage of indications have been lost. The approach used in this EAL facilitates prompt and accurate emergency classification assessments by focusing on the indications for a selected subset of parameters.~~

~~By focusing on the availability of the specified parameter values, instead of the sources of those values, the EAL recognizes and accommodates the wide variety of indications in nuclear power plant Control Rooms. Indication types and sources may be analog or digital, safety-related or not, primary or alternate, individual meter value or computer group display, etc.~~

~~A loss of plant annunciators will be evaluated for reportability in accordance with 10 CFR 50.72 (and the associated guidance in NUREG 1022), and reported if it significantly impairs the capability to perform emergency assessments. Compensatory measures for a loss of annunciation can be readily implemented and may include increased monitoring of main control boards and more frequent plant rounds by non-licensed operators. Their alerting function notwithstanding, annunciators do not provide the parameter values or specific component status information used to operate the plant, or process through AOPs or EOPs. Based on these considerations, a loss of annunciation is considered to be adequately addressed by reportability criteria, and therefore not included in this IC and EAL.~~

~~With respect to establishing event severity, the response to a loss of radiation monitoring data (e.g., process or effluent monitor values) is considered to be adequately bounded by the requirements of 10 CFR 50.72 (and associated guidance in NUREG 1022). The reporting of this event will ensure adequate plant staff and NRC awareness, and drive the establishment of appropriate compensatory measures and corrective actions. In addition, a loss of radiation monitoring data, by itself, is not a precursor to a more significant event.~~

~~Personnel at sites that have a Failure Modes and Effects Analysis (FMEA) included within the design basis of a digital I&C system should consider the FMEA information when developing their site-specific EALs.~~

~~Due to changes in the configurations of SAFETY SYSTEMS, including associated instrumentation and indications, during the cold shutdown, refueling, and defueled modes, no analogous IC is included for these modes of operation.~~

~~ECL Assignment Attributes: 3.1.2.B~~

~~SA5~~SA6

ECL: Alert

Initiating Condition: Automatic or manual ~~(trip [PWR] / scram [BWR])~~ fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.

Operating Mode Applicability: ~~Power Operation~~1, 2

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

~~Example Emergency~~Emergency Action Level~~s~~:

SA6.1 a. An automatic or manual ~~(trip [PWR] / scram [BWR])~~ did not shutdown the reactor.

AND

b. **ALL** of the following manual actions taken at 1C05 are not successful in lowering reactor power below 5% power

- Manual Scram Pushbuttons
- Mode Switch to Shutdown
- Alternate Rod Insertion (ARI)~~Manual actions taken at the reactor control consoles (1C05) are not successful in shutting down the reactor.~~

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(trip [PWR] /~~scram ~~[BWR])~~ that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(trip [PWR] /~~scram~~-[BWR])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control consoles (e.g., locally opening breakers). Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles."

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. ~~[*BWR*]~~

The plant response to the failure of an automatic or manual reactor ~~(trip [*PWR*] /~~ scram ~~[*BWR*])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shutdown the reactor is prolonged enough to cause a challenge to the ~~core cooling [PWR] /~~ RPV water level ~~[BWR]~~ or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS~~5~~6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS~~5~~6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the SAFETY SYSTEMS are designed (typically 3 to 5% power). ~~A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.~~

~~**Developer Notes:**~~

~~This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. A PWR with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, then the IC is also applicable in Startup Mode.~~

~~Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level).~~

~~The term "reactor control consoles" may be replaced with the appropriate site-specific term (e.g., main control boards).~~

~~ECL Assignment Attributes: 3.1.2.B~~

~~SA9~~SA8

ECL: Alert

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Level~~s~~:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of the SAFETY SYSTEM, then this emergency classification is not warranted.

SA8.1 a. The occurrence of **ANY** of the Table S-3 hazardous events:

Table S-3 Hazardous Events
• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

~~The occurrence of ANY of the following hazardous events:~~
~~Seismic event (earthquake)~~
~~Internal or external flooding event~~
~~High winds or tornado strike~~
~~FIRE~~
~~EXPLOSION~~
~~(site-specific hazards)~~
~~Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director~~

AND

b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:

- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode.

OR

- The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.~~,~~

~~Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or~~

~~The event has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode.~~

~~Loss of the safety function of a single train SAFETY SYSTEM.~~

Definitions:

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety--related.~~

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

~~. **EITHER** of the following:~~

~~1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.~~

~~**OR**~~

~~2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.~~

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria SA~~9~~8.1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

An event affecting a single-train SAFETY SYSTEM (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under SA8 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train SAFETY SYSTEM, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

~~This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.~~

~~EAL 1.b.1 addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.~~

~~EAL 1.b.2 addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.~~

Escalation of the emergency classification level would be via IC FS1 or ~~AS1~~RS1.

~~**Developer Notes:**~~

~~For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).~~

~~Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.~~

~~ECL Assignment Attributes: 3.1.2.B~~

SS1

ECL: Site Area Emergency

Initiating Condition: Loss of ALL~~all~~ offsite and ~~all~~ ALL onsite AC power to ~~emergency~~ essential buses for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Level~~s~~:

Note: The Emergency Director should declare the ~~Site Area Emergency~~event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

SS1.1 Loss of **ALL** offsite and **ALL** onsite AC power to ~~(site-specific emergency buses)~~1A3 and 1A4 buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety- related.~~

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs ~~AG1~~RG1, FG1 or SG1.

~~**Developer Notes:**~~

~~For a power source that has multiple generators, the EAL and/or Basis section should reflect the minimum number of operating generators necessary for that source to provide adequate power to an AC emergency bus. For example, if a backup power source is comprised of two generators (i.e., two 50% capacity generators sized to feed 1 AC emergency bus), the EAL and Basis section must specify that both generators for that source are operating.~~

~~The "site specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.3.B~~

~~SS8~~SS2

ECL: Site Area Emergency

Initiating Condition: Loss of ALL Vital DC power for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Levels:

Note: The Emergency Director should declare the ~~Site Area Emergency~~event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

SS2.1 Indicated voltage is less than ~~(site-specific bus voltage value)~~ 105 VDC on ~~ALL(site-specific Vital DC busses)~~ **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety- related.~~

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs ~~AG1~~RG1, FG1 or SG2.

~~SS5~~SS6

ECL: Site Area Emergency

Initiating Condition: Inability to shutdown the reactor causing a challenge to ~~(core cooling [PWR] /~~ RPV water level ~~[BWR])~~ or RCS heat removal.

Operating Mode Applicability: ~~Power Operation~~1, 2

~~Example~~ Emergency Action Levels:

SS6.1 a. An automatic or manual ~~(trip [PWR] /~~ scram ~~[BWR])~~ did not shutdown the reactor.

AND

b. ALL of the following manual actions taken at 1C05 are not successful in lowering reactor power below 5% power:

- Manual Scram Pushbuttons
- Mode Switch to Shutdown
- Alternate Rod Insertion (ARI)~~All manual actions to shutdown the reactor have been unsuccessful.~~

AND

c. **EITHER** of the following conditions exist:

- ~~(Site specific indication of an inability to adequately remove heat from the core) Reactor vessel water~~RPV level cannot be restored and maintained above -25 inches.

OR

- ~~(Site specific indication of an inability to adequately remove heat from the RCS)~~HCL (Graph 4 of EOP 2) exceeded.

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(trip [PWR] /~~ scram ~~[BWR])~~ that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition

Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shutdown the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the SAFETY SYSTEMS are designed (typically 3 to 5% power).

~~A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.~~

Escalation of the emergency classification level would be via IC ~~AG1~~ RG1 or FG1.

~~**Developer Notes:**~~

~~This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. A PWR with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, then the IC is also applicable in Startup Mode.~~

~~Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level).~~

~~Site-specific indication of an inability to adequately remove heat from the core:~~

~~[*BWR*] – Reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level (as described in the EOP bases).~~

~~[*PWR*] – Insert site-specific values for an incore/core exit thermocouple temperature and/or reactor vessel water level that drives entry into a core cooling restoration procedure (or otherwise requires implementation of prompt restoration actions). Alternately, a site may use incore/core exit thermocouple temperatures greater than 1,200°F and/or a reactor vessel water level that corresponds to approximately the middle of active fuel. Plants with reactor vessel level instrumentation that cannot measure down to approximately the middle of active fuel should use the lowest on-scale reading that is not above the top of active fuel. If the lowest on-scale reading is above the top of active fuel, then a reactor vessel level value should not be included.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters used in the Core Cooling Red Path.~~

~~Site-specific indication of an inability to adequately remove heat from the RCS:~~

~~[*BWR*] – Use the Heat Capacity Temperature Limit. This addresses the inability to remove heat via the main condenser and the suppression pool due to high pool water temperature.~~

~~[*PWR*] – Insert site-specific parameters associated with inadequate RCS heat removal via the steam generators. These parameters should be identical to those used for the Inadequate Heat Removal threshold Fuel Clad Barrier Potential Loss 2.B and threshold RCS Barrier Potential Loss 2.A in the PWR EAL Fission Product Barrier Table.~~

~~ECL Assignment Attributes: 3.1.3.B~~

~~SS8~~

~~**ECL:** Site Area Emergency~~

~~**Initiating Condition:** Loss of all Vital DC power for 15 minutes or longer.~~

~~**Operating Mode Applicability:** Power Operation, Startup, Hot Standby, Hot Shutdown~~~~1, 2, 3, 4~~

~~**Example Emergency Action Levels:**~~

~~**Note:** The Emergency Director should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.~~

~~1 Indicated voltage is less than (site-specific bus voltage value)~~~~115 VDC~~ ~~on **ALL** (site-specific Vital DC busses)~~~~1(2)-D-01, D-02, D-03, and D-04~~ ~~for 15 minutes or longer.~~

~~**Basis:**~~

~~SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety-related.~~

~~This IC addresses a loss of Vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.~~

~~Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.~~

~~Escalation of the emergency classification level would be via ICs AG1~~~~RG1~~~~, FG1 or SG8.~~

~~**Developer Notes:**~~

~~The "site-specific bus voltage value" should be based on the minimum bus voltage necessary for adequate operation of SAFETY SYSTEM equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed. The typical value for an entire battery set is approximately 105 VDC. For a 60 cell string of batteries, the cell voltage is approximately 1.75 Volts per cell. For a 58 string battery set, the minimum voltage is approximately 1.81 Volts per cell.~~

~~The "site-specific Vital DC busses" are the DC busses that provide monitoring and control capabilities for SAFETY SYSTEMS.~~

~~ECL Assignment Attributes: 3.1.3.B~~

SG1

ECL: General Emergency

Initiating Condition: Prolonged loss of ~~all~~ ALL offsite and ALL~~all~~ onsite AC power to ~~emergency~~ essential buses.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Level~~s~~:

Note: The Emergency Director should declare the ~~General Emergency~~event promptly upon determining that ~~(site-specific hours)~~ the applicable time ~~4 hours~~ has been exceeded, or will likely be exceeded.

SG1.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses~~(site-specific emergency buses)~~.

AND

b. **EITHER** of the following:

- Restoration of at least one AC ~~emergency~~ essential bus in less than ~~(site-specific hours)~~4 hours is not likely.

OR

- ~~(Site-specific indication of an inability to adequately remove heat from the core)~~RPV level cannot be restored and maintained above -25 inches.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety--related.~~

Basis:

This IC addresses a prolonged loss of all power sources to AC ~~emergency~~ essential buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC ~~emergency~~ essential bus by the end of the ~~analyzed~~ 4 hour station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one essential~~emergency~~ bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

~~**Developer Notes:**~~

~~Although this IC and EAL may be viewed as redundant to the Fission Product Barrier ICs, it is included to provide for a more timely escalation of the emergency classification level.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The "site-specific hours" to restore AC power to an emergency bus should be based on the station blackout coping analysis performed in accordance with 10 CFR § 50.63 and Regulatory Guide 1.155, *Station Blackout*.~~

~~Site-specific indication of an inability to adequately remove heat from the core:~~

~~[*BWR*] – Reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level (as described in the EOP bases).~~

~~[*PWR*] – Insert site-specific values for an incore/core exit thermocouple temperature and/or reactor vessel water level that drive entry into a core cooling restoration procedure (or otherwise requires implementation of prompt restoration actions). Alternately, a site may use incore/core exit thermocouple temperatures greater than 1,200°F and/or a reactor vessel water level that corresponds to approximately the middle of active fuel. Plants with reactor vessel level instrumentation that cannot measure down to approximately the middle of active fuel should use the lowest on-scale reading that is not above the top of active fuel. If the lowest on-scale reading is above the top of active fuel, then a reactor vessel level value should not be included.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters used in the Core Cooling Red Path.~~

~~ECL Assignment Attributes: 3.1.4.B~~

~~SG8~~SG2

ECL: General Emergency

Initiating Condition: Loss of ~~all~~ ALL AC and Vital DC power sources for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~1, 2, 3

~~Example~~ Emergency Action Levels:

Note: The Emergency Director should declare the ~~General Emergency~~event promptly upon determining that the applicable time ~~15 minutes~~ has been exceeded, or will likely be exceeded.

SG2.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to ~~(site-specific emergency buses)~~ 1A3 and 1A4 buses for 15 minutes or longer.

AND

b. Indicated voltage is less than ~~(site-specific bus voltage value)~~105 VDC on ~~**ALL (site-specific Vital DC busses)**~~ **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related:~~A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. Systems classified as safety-related.~~

Basis:

This IC addresses a concurrent and prolonged loss of both AC and Vital DC power. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of Vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both AC and DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

~~**Developer Notes:**~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The "site-specific bus voltage value" should be based on the minimum bus voltage necessary for adequate operation of SAFETY SYSTEM equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed.~~

~~The typical value for an entire battery set is approximately 105 VDC. For a 60 cell string of batteries, the cell voltage is approximately 1.75 Volts per cell. For a 58 string battery set, the minimum voltage is approximately 1.81 Volts per cell.~~

~~The "site-specific Vital DC busses" are the DC busses that provide monitoring and control capabilities for SAFETY SYSTEMS.~~

~~This IC and EAL were added to Revision 6 to address operating experience from the March, 2011 accident at Fukushima Daiichi.~~

~~ECL Assignment Attributes: 3.1.4.B~~

APPENDIX A – ACRONYMS AND ABBREVIATIONS

AC .... Alternating Current
AOP .... Abnormal Operating Procedure
~~.......... A~~
~~PRM .... Average Power Range Meter~~
ATWS .... Anticipated Transient Without Scram
~~.......... B~~
~~&W .... Babcock and Wilcox~~
~~.......... B~~
~~IIT .... Boron Injection Initiation Temperature~~
BWR .... Boiling Water Reactor
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
~~CTMT/~~CNMT .... Containment
~~.......... C~~
~~SF .... Critical Safety Function~~
~~.......... C~~
~~SFST .... Critical Safety Function Status Tree~~
~~.......... D~~
~~BA .... Design Basis Accident~~
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPG .... Emergency Procedure Guideline
~~.......... E~~
~~PIP .... Emergency Plan Implementing Procedure~~
~~.......... E~~
~~PR .... Evolutionary Power Reactor~~
~~.......... E~~
~~PRI .... Electric Power Research Institute~~
~~.......... E~~
~~RG .... Emergency Response Guideline~~
~~.......... F~~
EMA .... Federal Emergency Management Agency
~~FSAR .... Final Safety Analysis Report~~
GE .... General Emergency
HC~~T~~L .... Heat Capacity ~~Temperature~~ Limit
HPCI .... High Pressure Coolant Injection
~~.......... H~~
~~SI .... Human System Interface~~
IC .... Initiating Condition

~~..........~~I
D.......... Inside Diameter
~~IPEEE.......... Individual Plant Examination of External Events (Generic Letter 88-20)~~
ISFSI.......... Independent Spent Fuel Storage Installation
Keff.......... Effective Neutron Multiplication Factor
LCO.......... Limiting Condition of Operation
~~..........~~L
OCA.......... Loss of Coolant Accident
~~..........M~~
~~CR.......... Main Control Room~~
~~..........M~~
~~SIV.......... Main Steam Isolation Valve~~
~~MSL.......... Main Steam Line~~
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MW.......... Megawatt
NEI.......... Nuclear Energy Institute
~~..........N~~
~~PP.......... Nuclear Power Plant~~
~~..........~~N
RC.......... Nuclear Regulatory Commission
~~NSSS.......... Nuclear Steam Supply System~~
NORAD.......... North American Aerospace Defense Command
~~(NO)~~UE.......... ~~(Notification Of)~~ Unusual Event
NUMARC[1] .... Nuclear Management and Resources Council
OBE.......... Operating Basis Earthquake
OCA.......... Owner Controlled Area
~~..........O~~
~~DCM/~~ODAM.......... Offsite Dose ~~Calculation (~~Assessment~~)~~ Manual
~~ORO.......... Off-site Response Organization~~
PA.......... Protected Area
~~..........P~~
~~ACS.......... Priority Actuation and Control System~~
PAG.......... Protective Action Guideline
~~..........P~~
~~ICS.......... Process Information and Control System~~
PRA/PSA.......... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR.......... Pressurized Water Reactor
~~..........P~~
~~S.......... Protection System~~
PSIG.......... Pounds per Square Inch Gauge
R.......... Roentgen
~~..........R~~
~~CC.......... Reactor Control Console~~
RCIC.......... Reactor Core Isolation Cooling

[1] NUMARC was a predecessor organization of the Nuclear Energy Institute (NEI).

RCS ... Reactor Coolant System
Rem, rem, REM ... Roentgen Equivalent Man
~~R~~
~~ETS ... Radiological Effluent Technical Specifications~~
RPS ... Reactor Protection System
RPV ... Reactor Pressure Vessel
~~R~~
~~VLIS ... Reactor Vessel Level Instrumentation System~~
RWCU ... Reactor Water Cleanup
~~S~~
~~AR ... Safety Analysis Report~~
~~S~~
~~AS ... Safety Automation System~~
~~S~~
~~BO ... Station Blackout~~
SCBA ... Self-Contained Breathing Apparatus
~~S~~
~~G ... Steam Generator~~
~~S~~
~~I ... Safety Injection~~
~~S~~
~~ICS ... Safety Information and Control System~~
S
PDS ... Safety Parameter Display System
~~SRO ... Senior Reactor Operator~~
TEDE ... Total Effective Dose Equivalent
T~~O~~AF ... Top of Active Fuel
TSC ... Technical Support Center

U
FSAR ... Updated Final Safety Analysis Report
~~WOG ... Westinghouse Owners Group~~

APPENDIX B – DEFINITIONS

The following definitions are taken from Title 10, Code of Federal Regulations, and related regulatory guidance documents.

Alert: Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.

General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Notification of Unusual Event (NOUE)~~[1]~~: Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of ~~safety systems~~SAFETY SYSTEMS occurs.

Site Area Emergency: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.

The following are key terms necessary for overall understanding the ~~NEI 99-01~~DAEC emergency classification scheme.

Emergency Action Level (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Emergency Classification Level (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are:

Notification of Unusual Event (NOUE)
Alert
Site Area Emergency (SAE)
General Emergency (GE)

~~[1] This term is sometimes shortened to Unusual Event (UE) or other similar site-specific terminology.~~

Fission Product Barrier Threshold: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

Selected terms used in Initiating Condition and Emergency Action Level statements are set in all capital letters (e.g., ALL CAPS). These words are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

CONFINEMENT BOUNDARY:~~(Insert a site-specific definition for this term.) **Developer Note**~~ – The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. This corresponds to the pressure boundary for the Dry Shielded Canister (DSC) shell (including the inner bottom cover plate) base metal and associated confinement boundary welds.

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

~~CONTAINMENT CLOSURE: (Insert a site-specific definition for this term.) **Developer Note** – Site specific procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAECs, this is considered to be Secondary Containment as required by Technical Specifications.The procedurally defined conditions or actions taken to secure containment (primary or secondary for BWR) and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.~~

DESIGN BASIS EARTHQUAKE (DBE): A DBE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

~~FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized. **Developer Note** – This term is applicable to PWRs only.~~

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION: An act toward a ~~NPP~~nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the ~~NPP~~nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

~~NORMAL LEVELS: As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.~~

OPERATING BASIS EARTHQUAKE (OBE): An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.

OWNER CONTROLLED AREA: ~~(Insert a site-specific definition for this term.) **Developer Note**~~ This term is typically taken to mean the site property owned by~~,~~ or otherwise under the control of~~,~~ the licensee. ~~In some cases, it may be appropriate for a licensee to define a smaller area with a perimeter closer to the plant Protected Area perimeter (e.g., a site with a large OCA where some portions of the boundary may be a significant distance from the Protected Area). In these cases, developers should consider using the boundary defined by the Restricted or Secured Owner Controlled Area (ROCA/SOCA). The area and boundary selected for scheme use must be consistent with the description of the same area and boundary contained in the Security Plan.~~

PROJECTILE: An object directed toward a ~~NPP~~nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: ~~(Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean t~~The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

REFUELING PATHWAY: ~~(Insert a site-specific definition for this term.) **Developer Note** – This description should include all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.~~ Includes all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.

~~RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection. **Developer Note** – This term is applicable to PWRs only.~~

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These ~~are typically~~ systems are classified as safety-related. ~~**Developer Note** – This term may be modified to include the attributes of "safety-related" in accordance with 10 CFR 50.2 or other site-specific terminology, if desired.~~

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the Company. UFSAR Figure 1.2-1 identifies the DAEC SITE BOUNDARY.

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

~~APPENDIX C – PERMANENTLY DEFUELED STATION ICs/EALs~~

~~Recognition Category PD provides a stand-alone set of ICs/EALs for a Permanently Defueled nuclear power plant to consider for use in developing a site-specific emergency classification scheme. For development, it was assumed that the plant had operated under a 10 CFR § 50 license and that the operating company has permanently ceased plant operations. Further, the company intends to store the spent fuel within the plant for some period of time.~~

~~When in a permanently defueled condition, the plant licensee typically receives approval from the NRC for exemption from specific emergency planning requirements. These exemptions reflect the lowered radiological source term and risks associated with spent fuel pool storage relative to reactor at-power operation. Source terms and accident analyses associated with plausible accidents are documented in the station's Final Safety Analysis Report (FSAR), as updated. As a result, each licensee will need to develop a site-specific emergency classification scheme using the NRC-approved exemptions, revised source terms, and revised accident analyses as documented in the station's FSAR.~~

~~Recognition Category PD uses the same ECLs as operating reactors; however, the source term and accident analyses typically limit the ECLs to an Unusual Event and Alert. The Unusual Event ICs provide for an increased awareness of abnormal conditions while the Alert ICs are specific to actual or potential impacts to spent fuel. The source terms and release motive forces associated with a permanently defueled plant would not be sufficient to require declaration of a Site Area Emergency or General Emergency.~~

~~A permanently defueled station is essentially a spent fuel storage facility with the spent fuel is stored in a pool of water that serves as both a cooling medium (i.e., removal of decay heat) and shield from direct radiation. These primary functions of the spent fuel storage pool are the focus of the Recognition Category PD ICs and EALs. Radiological effluent IC and EALs were included to provide a basis for classifying events that cannot be readily classified based on an observable events or plant conditions alone.~~

~~Appropriate ICs and EALs from Recognition Categories A, C, F, H, and S were modified and included in Recognition Category PD to address a spectrum of the events that may affect a spent fuel pool. The Recognition Category PD ICs and EALs reflect the relevant guidance in Section 3 of this document (e.g., the importance of avoiding both over-classification and under-classification). Nonetheless, each licensee will need to develop their emergency classification scheme using the NRC approved exemptions, and the source terms and accident analyses specific to the licensee. Security-related events will also need to be considered.~~

~~Table PD-1: Recognition Category "PD" Initiating Condition Matrix~~

~~UNUSUAL EVENT~~	~~ALERT~~
~~**PD-AU1** Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-AA1** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.~~ ~~*Op. Modes: Not Applicable*~~
~~**PD-AU2** UNPLANNED rise in plant radiation levels.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-AA2** UNPLANNED rise in plant radiation levels that impedes plant access required to maintain spent fuel integrity.~~ ~~*Op. Modes: Not Applicable*~~
~~**PD-SU1** UNPLANNED spent fuel pool temperature rise.~~ ~~*Op. Modes: Not Applicable*~~	
~~**PD-HU1** Confirmed SECURITY CONDITION or threat.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.~~ ~~*Op. Modes: Not Applicable*~~
~~**PD-HU2** Hazardous event affecting SAFETY SYSTEM equipment necessary for spent fuel cooling.~~ ~~*Op. Modes: Not Applicable*~~	
~~**PD-HU3** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-HA3** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.~~ ~~*Op. Modes: Not Applicable*~~

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required.~~

~~PD-AU1~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2)~~

~~**Notes:**~~

- ~~The Emergency Director should declare the Unusual Event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.~~
- ~~If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes.~~
- ~~If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.~~

~~(1) Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.~~

~~(2) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.~~

~~**Basis:**~~

~~This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.~~

~~Nuclear power plants incorporate design features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, and to control and monitor intentional releases. The occurrence of an extended, uncontrolled radioactive release to the environment is indicative of degradation in these features and/or controls.~~

~~Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.~~

~~Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.~~

~~Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.~~

~~EAL #1 – This EAL addresses radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. This EAL will typically be associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).~~

~~EAL #2 – This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).~~

~~Escalation of the emergency classification level would be via IC PD-AA1.~~

~~Developer Notes:~~

~~The "site-specific effluent release controlling document" is the Radiological Effluent Technical Specifications (RETS) or, for plants that have implemented Generic Letter 89-01[11], the Offsite Dose Calculation Manual (ODCM). These documents implement regulations related to effluent controls (e.g., 10 CFR Part 20 and 10 CFR Part 50, Appendix I). As appropriate, the RETS or ODCM methodology should be used for establishing the monitor thresholds for this IC.~~

~~Listed monitors should include the effluent monitors described in the RETS or ODCM.~~

~~Developers may also consider including installed monitors associated with other potential effluent pathways that are not described in the RETS or ODCM[12,13]. If included, EAL values for these monitors should be determined using the most applicable dose/release limits presented in the RETS or ODCM. It is recognized that a calculated EAL value may be below what the monitor can read; in that case, the monitor does not need to be included in the list. Also, some monitors may not be governed by Technical Specifications or other license-related related requirements; therefore, it is important that the associated EAL and basis section clearly identify any limitations on the use or availability of these monitors.~~

~~Some sites may find it advantageous to address gaseous and liquid releases with separate EALs.~~

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[11] ~~*Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program*~~

[12] ~~This includes consideration of the effluent monitors described in the site emergency plan section(s) which address the requirements of 10 CFR 50.47(b)(8) and (9).~~

[13] ~~Developers should keep in mind the requirements of 10 CFR 50.54(q) and the guidance provided by INPO related to emergency response equipment when considering the addition of other effluent monitors.~~

~~Radiation monitor readings should reflect values that correspond to a radiological release exceeding 2 times a release control limit. The controlling document typically describes methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values. In cases where a methodology is not adequately defined, developers should determine values consistent with effluent control regulations (e.g., 10 CFR Part 20 and 10 CFR Part 50 Appendix I) and related guidance.~~

~~For EAL #1 - Values in this EAL should be 2 times the setpoint established by the radioactivity discharge permit to warn of a release that is not in compliance with the specified limits. Indexing the value in this manner ensures consistency between the EAL and the setpoint established by a specific discharge permit.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.1.B~~

~~PD-AU2~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** UNPLANNED rise in plant radiation levels.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2)~~

~~(1) a. UNPLANNED water level drop in the spent fuel pool as indicated by ANY of the following:~~

~~(site-specific level indications).~~

~~**AND**~~

~~b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors.~~

~~(site-specific list of area radiation monitors).~~

~~(2) Area radiation monitor reading or survey result indicates an UNPLANNED rise of 25 mR/hr over NORMAL LEVELS.~~

~~**Basis:**~~

~~This IC addresses elevated plant radiation levels caused by a decrease in water level above irradiated (spent) fuel or other UNPLANNED events. The increased radiation levels are indicative of a minor loss in the ability to control radiation levels within the plant or radioactive materials. Either condition is a potential degradation in the level of safety of the plant.~~

~~A water level decrease will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.~~

~~The effects of planned evolutions should be considered. Note that EAL #1 is applicable only in cases where the elevated reading is due to an UNPLANNED water level drop. EAL #2 excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials.~~

~~Escalation of the emergency classification level would be via IC PD-AA1 or PD-AA2.~~

~~**Developer Notes:**~~

~~For EAL #1 – Site-specific indications may include instrumentation values such as water level and area radiation monitor readings, and personnel reports. If available, video cameras may allow for remote observation. Depending on available instrumentation, the declaration may also be based on indications of water makeup rate and/or decreases in the level of a water storage tank.~~

~~For EAL #2 – The specified value of 25 mR/hr may be set to another value for a specific application with appropriate justification.~~

~~ECL Assignment Attributes: 3.1.1.B~~

~~PD-SU1~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** UNPLANNED spent fuel pool temperature rise.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:**~~

~~(1) UNPLANNED spent fuel pool temperature rise to greater than (site-specific °F).~~

~~**Basis:**~~

~~This IC addresses a condition that is a precursor to a more serious event and represents a potential degradation in the level of safety of the plant. If uncorrected, boiling in the pool will occur, and result in a loss of pool level and increased radiation levels.~~

~~Escalation of the emergency classification level would be via IC PD-AA1 or PD-AA2.~~

~~**Developer Notes:**~~

~~The site-specific temperature should be chosen based on the starting point for fuel damage calculations in the SAR. Typically, this temperature is 125° to 150°F. Spent Fuel Pool temperature is normally maintained well below this point thus allowing time to correct the cooling system malfunction prior to classification.~~

~~ECL Assignment Attributes: 3.1.1.A~~

~~PD-HU1~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Confirmed SECURITY CONDITION or threat.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2 or 3)~~

~~(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).~~

~~(2) Notification of a credible security threat directed at the site.~~

~~(3) A validated notification from the NRC providing information of an aircraft threat.~~

~~**Basis:**~~

~~This IC addresses events that pose a threat to plant personnel or the equipment necessary to maintain cooling of spent fuel, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under IC PD-HA1.~~

~~Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and OROs.~~

~~Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.~~

~~EAL #1 references (site-specific security shift supervision) because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.~~

~~EAL #2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with (site-specific procedure).~~

~~EAL #3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with (site-specific procedure).~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be~~

~~advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~Escalation of the emergency classification level would be via IC PD-HA1.~~

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~The (site-specific procedure) is the procedure(s) used by Control Room and/or Security personnel to determine if a security threat is credible, and to validate receipt of aircraft threat information.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~ECL Assignment Attributes: 3.1.1.A~~

PD-HU2

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Hazardous event affecting SAFETY SYSTEM equipment necessary for spent fuel cooling.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:**~~

~~(1) a. The occurrence of **ANY** of the following hazardous events:~~

- ~~Seismic event (earthquake)~~
- ~~Internal or external flooding event~~
- ~~High winds or tornado strike~~
- ~~FIRE~~
- ~~EXPLOSION~~
- ~~(site-specific hazards)~~
- ~~Other events with similar hazard characteristics as determined by the Shift Manager~~

~~**AND**~~

~~b. The event has damaged at least one train of a SAFETY SYSTEM needed for spent fuel cooling.~~

~~**AND**~~

~~c. The damaged SAFETY SYSTEM train(s) cannot, or potentially cannot, perform its design function based on **EITHER**:~~

- ~~Indications of degraded performance~~
- ~~VISIBLE DAMAGE~~

~~**Basis:**~~

~~This IC addresses a hazardous event that causes damage to at least one train of a SAFETY SYSTEM needed for spent fuel cooling. The damage must be of sufficient magnitude that the system(s) train cannot, or potentially cannot, perform its design function. This condition reduces the margin to a loss or potential loss of the fuel clad barrier, and therefore represents a potential degradation of the level of safety of the plant.~~

~~For EAL 1.c, the first bullet addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available.~~

~~For EAL 1.c, the second bullet addresses damage to a SAFETY SYSTEM train that is not in service/operation or readily apparent through indications alone. Operators will make this~~

~~determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.~~

~~Escalation of the emergency classification level could, depending upon the event, be based on any of the Alert ICs; PD-AA1, PD-AA2, PD-HA1 or PD-HA3.~~

~~**Developer Notes:**~~

~~For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).~~

~~Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.~~

~~ECL Assignment Attributes: 3.1.1.A and 3.1.1C~~

~~PD-HU3~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:**~~

~~(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.~~

~~**Basis:**~~

~~This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a NOUE.~~

~~PD-AA1~~

~~**ECL:** Alert~~

~~**Initiating Condition:** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2 or 3 or 4)~~

~~**Notes:**~~

- ~~The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.~~
- ~~If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.~~
- ~~If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.~~
- ~~The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.~~

~~(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:~~

~~(site-specific monitor list and threshold values)~~

~~(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).~~

~~(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.~~

~~(4) Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point):~~

- ~~Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.~~
- ~~Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.~~

~~**Basis:**~~

~~This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual~~

~~offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).~~

~~Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.~~

~~The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.~~

~~Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.~~

~~Developer Notes:~~

~~While this IC may not be met absent challenges to the cooling of spent fuel, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant conditions alone.~~

~~The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE….".~~

~~The EPA PAG guidance provides for the use adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.~~

~~The "site-specific monitor list and threshold values" should be determined with consideration of the following:~~

- ~~Selection of the appropriate installed gaseous and liquid effluent monitors.~~
- ~~The effluent monitor readings should correspond to a dose of 10 mrem TEDE or 50 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.~~
- ~~Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for IC PD-AU1.~~

- ~~The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for IC PD-AU1.~~
- ~~Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.~~

~~The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.2.C~~

~~PD-AA2~~

~~**ECL:** Alert~~

~~**Initiating Condition:** UNPLANNED rise in plant radiation levels that impedes plant access required to maintain spent fuel integrity.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2)~~

~~(1) UNPLANNED dose rate greater than 15 mR/hr in **ANY** of the following areas requiring continuous occupancy to maintain control of radioactive material or operation of systems needed to maintain spent fuel integrity:~~

~~(site-specific area list)~~

~~(2) UNPLANNED Area Radiation Monitor readings or survey results indicate a rise by 100 mR/hr over NORMAL LEVELS that impedes access to **ANY** of the following areas needed to maintain control of radioactive material or operation of systems needed to maintain spent fuel integrity.~~

~~(site-specific area list)~~

~~**Basis:**~~

~~This IC addresses increased radiation levels that impede necessary access to areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain systems needed to maintain spent fuel integrity. As used here, 'impede' includes hindering or interfering, provided that the interference or delay is sufficient to significantly threaten necessary plant access. It is this impaired access that results in the actual or potential substantial degradation of the level of safety of the plant.~~

~~This IC does not apply to anticipated temporary increases due to planned events.~~

~~**Developer Notes:**~~

~~The value of 15mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, *Clarification of TMI Action Plan Requirements*, provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.~~

~~The specified value of 100 mR/hr may be set to another value for a specific application with appropriate justification.~~

~~ECL Assignment Attributes: 3.1.2.C~~

~~PD-HA1~~

~~**ECL:** Alert~~

~~**Initiating Condition:** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:** (1 or 2)~~

~~(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision).~~

~~(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.~~

~~**Basis:**~~

~~This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.~~

~~Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.~~

~~Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.~~

~~As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations, allowing them to be better prepared should it be necessary to consider further actions.~~

~~This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.~~

~~EAL #1 is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located within the OWNER CONTROLLED AREA.~~

~~EAL #2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related~~

~~notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with (site-specific procedure).~~

~~The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.~~

~~In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security sensitive information should be contained in non-public documents such as the Security Plan.~~

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security sensitive information should be contained in non-public documents such as the Security Plan.~~

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the OWNER CONTROLLED AREA.~~

~~ECL Assignment Attributes: 3.1.2.D~~

~~PD-HA3~~

~~**ECL:** Alert~~

~~**Initiating Condition:** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.~~

~~**Operating Mode Applicability:** Not Applicable~~

~~**Example Emergency Action Levels:**~~

~~(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.~~

~~**Basis:**~~

~~This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for an Alert.~~

ATTACHMENT 2

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATING TO

LICENSE AMENDMENT REQUEST TSCR-166

UPDATED CLEAN COPY OF THE PROPOSED DAEC EAL SCHEME

125 pages follow

Duane Arnold Energy Center (DAEC) Emergency Action Levels Technical Bases Document

TBD, 2018

TABLE OF CONTENTS

DUANE ARNOLD EMERGENCY ACTION LEVELS TECHNICAL BASIS DOCUMENT

1 BASIS FOR EMERGENCY ACTION LEVELS

1.1 OPERATING REACTORS

Title 10, Code of Federal Regulations (CFR), Energy, contains the U.S. Nuclear Regulatory Commission (NRC) regulations that apply to nuclear power facilities. Several of these regulations govern various aspects of an emergency classification scheme. A review of the relevant sections listed below will aid the reader in understanding the key terminology provided in Section 3.0 of this document.

- 10 CFR § 50.47(a)(1)(i)
- 10 CFR § 50.47(b)(4)
- 10 CFR § 50.54(q)
- 10 CFR § 50.72(a)
- 10 CFR § 50, Appendix E, IV.B, Assessment Actions
- 10 CFR § 50, Appendix E, IV.C, Activation of Emergency Organization

The above regulations are supplemented by various regulatory guidance documents. Three documents of particular relevance to NEI 99-01 are:

NUREG-0654/FEMA-REP-1, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, October 1980. [Refer to Appendix 1, *Emergency Action Level Guidelines for Nuclear Power Plants*]
NUREG-1022, *Event Reporting Guidelines 10 CFR § 50.72 and § 50.73*
Regulatory Guide 1.101, *Emergency Response Planning and Preparedness for Nuclear Power Reactors*

1.2 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

Selected guidance in NEI 99-01 is applicable to licensees electing to use their 10 CFR 50 emergency plan to fulfill the requirements of 10 CFR 72.32 for a stand-alone ISFSI. The emergency classification levels applicable to an ISFSI are consistent with the requirements of 10 CFR 50 and the guidance in NUREG 0654/FEMA-REP-1. The initiating conditions germane to a 10 CFR 72.32 emergency plan (as described in NUREG-1567) are subsumed within the classification scheme for a 10 CFR 50.47 emergency plan.

The generic ICs and EALs for an ISFSI are presented in Section 8, ISFSI ICs/EALs. IC E-HU1 covers the spectrum of credible natural and man-made events included within the scope of an ISFSI design. This IC is not applicable to installations or facilities that may process and/or repackage spent fuel (e.g., a Monitored Retrievable Storage Facility or an ISFSI at a spent fuel processing facility). In addition, appropriate aspects of IC HU1 and IC HA1 should also be included to address a HOSTILE ACTION directed against an ISFSI.

The analysis of potential onsite and offsite consequences of accidental releases associated with the operation of an ISFSI is contained in NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees.* NUREG-1140 concluded that the postulated worst-case accident involving an ISFSI has insignificant consequences to public health and safety. This evaluation shows that the maximum offsite dose to a member of the public due to an accidental release of radioactive materials would not exceed 1 rem Effective Dose Equivalent.

Regarding the above information, the expectations for an offsite response to an Alert classified under a 10 CFR 72.32 emergency plan are generally consistent with those for a Notification of Unusual Event in a 10 CFR 50.47 emergency plan (e.g., to provide assistance if requested). Also, the licensee's Emergency Response Organization (ERO) required for 10 CFR 72.32 emergency plan is different than that prescribed for a 10 CFR 50.47 emergency plan (e.g., no emergency technical support function).

1.3 NRC ORDER EA-12-051

The Fukushima Daiichi accident of March 11, 2012, was the result of a tsunami that exceeded the plant's design basis and flooded the site's emergency electrical power supplies and distribution systems. This caused an extended loss of power that severely compromised the key safety functions of core cooling and containment integrity, and ultimately led to core damage in three reactors. While the loss of power also impaired the spent fuel pool cooling function, sufficient water inventory was maintained in the pools to preclude fuel damage from the loss of cooling.

Following a review of the Fukushima Daiichi accident, the NRC concluded that several measures were necessary to ensure adequate protection of public health and safety under the provisions of the backfit rule, 10 CFR 50.109(a)(4)(ii). Among them was to provide each spent fuel pool with reliable level instrumentation to significantly enhance the ability of key decision-makers to allocate resources effectively following a beyond design basis event. To this end, the NRC issued Order EA-12-051, *Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,* on March 12, 2012, to all US nuclear plants with an operating license, construction permit, or combined construction and operating license.

NRC Order EA-12-051 states, in part, "All licensees ... shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred." To this end, all licensees must provide:

- A primary and back-up level instrument that will monitor water level from the normal level to the top of the used fuel rack in the pool;
- A display in an area accessible following a severe event; and
- Independent electrical power to each instrument channel and provide an alternate remote power connection capability.

NEI 12-02, *Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,"* provides guidance for complying with NRC Order EA-12-051.

NEI 99-01, Revision 6, includes three EALs that reflect the availability of the enhanced spent fuel pool level instrumentation associated with NRC Order EA-12-051. These EALs are included within ICs RA2, RS2, and RG2.

2 KEY TERMINOLOGY USED IN DAEC EAL SCHEME

There are several key terms that appear throughout the EAL methodology. These terms are introduced in this section to support understanding of subsequent material. As an aid to the reader, the following table is provided as an overview to illustrate the relationship of the terms to each other.

Emergency Classification Level			
Unusual Event	Alert	SAE	GE
↓	↓	↓	↓
Initiating Condition	Initiating Condition	Initiating Condition	Initiating Condition
↓	↓	↓	↓
Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis
(1) - When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition. This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL.			

2.1 EMERGENCY CLASSIFICATION LEVEL (ECL)

One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are:

Notification of Unusual Event (NOUE)
Alert
Site Area Emergency (SAE)
General Emergency (GE)

2.1.1 Notification of Unusual Event (NOUE)

Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Purpose: The purpose of this classification is to assure that the first step in future response has been carried out, to bring the operations staff to a state of readiness, and to provide systematic handling of unusual event information and decision-making.

2.1.2 Alert

Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.

Purpose: The purpose of this classification is to assure that emergency personnel are readily available to respond if the situation becomes more serious or to perform confirmatory radiation monitoring if required, and provide offsite authorities current information on plant status and parameters.

2.1.3 Site Area Emergency

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.

Purpose: The purpose of the Site Area Emergency declaration is to assure that emergency response centers are staffed, to assure that monitoring teams are dispatched, to assure that personnel required for evacuation of near-site areas are at duty stations if the situation becomes more serious, to provide consultation with offsite authorities, and to provide updates to the public through government authorities.

2.1.4 General Emergency (GE)

Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Purpose: The purpose of the General Emergency declaration is to initiate predetermined protective actions for the public, to provide continuous assessment of information from the licensee and offsite organizational measurements, to initiate additional measures as indicated by actual or potential releases, to provide consultation with offsite authorities, and to provide updates for the public through government authorities.

2.2 INITIATING CONDITION (IC)

An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

Discussion: An IC describes an event or condition, the severity or consequences of which meets the definition of an emergency classification level. An IC can be expressed as a continuous, measurable parameter (e.g., RCS leakage), an event (e.g., an earthquake) or the status of one or more fission product barriers (e.g., loss of the RCS barrier).

Appendix 1 of NUREG-0654 does not contain example Emergency Action Levels (EALs) for each ECL, but rather Initiating Conditions (i.e., plant conditions that indicate that a radiological emergency, or events that could lead to a radiological emergency, has occurred). NUREG-0654 states that the Initiating Conditions form the basis for establishment by a licensee of the specific plant instrumentation readings (as applicable) which, if exceeded, would initiate the emergency classification. Thus, it is the specific instrument readings that would be the EALs.

2.3 EMERGENCY ACTION LEVEL (EAL)

A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Discussion: EAL statements may utilize a variety of criteria including instrument readings and status indications; observable events; results of calculations and analyses; entry into particular procedures; and the occurrence of natural phenomena.

2.4 FISSION PRODUCT BARRIER THRESHOLD

A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

Discussion: Fission product barrier thresholds represent threats to the defense in depth design concept that precludes the release of radioactive fission products to the environment. This concept relies on multiple physical barriers, any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

- Fuel Clad
- Reactor Coolant System (RCS)
- Containment

Upon determination that one or more fission product barrier thresholds have been exceeded, the combination of barrier loss and/or potential loss thresholds is compared to the fission product barrier IC/EAL criteria to determine the appropriate ECL. In some accident sequences, the ICs and EALs presented in the Abnormal Radiation Levels/ Radiological Effluent (R) Recognition Category will be exceeded at the same time, or shortly after, the loss of one or more fission product barriers. This redundancy is intentional as the former ICs address radioactivity releases that result in certain offsite doses from whatever cause, including events that might not be fully encompassed by fission product barriers (e.g., spent fuel pool accidents, design containment leakage following a LOCA, etc.).

3 DESIGN OF THE DAEC EMERGENCY CLASSIFICATION SCHEME

3.1 ASSIGNMENT OF EMERGENCY CLASSIFICATION LEVELS (ECLs)

An effective emergency classification scheme must incorporate a realistic and accurate assessment of risk, both to plant workers and the public. There are obvious health and safety risks in underestimating the potential or actual threat from an event or condition; however, there are also risks in overestimating the threat as well (e.g., harm that may occur during an evacuation). The DAEC emergency classification scheme attempts to strike an appropriate balance between reasonably anticipated event or condition consequences, potential accident trajectories, and risk avoidance or minimization.

There are a range of "non-emergency events" reported to the US Nuclear Regulatory Commission (NRC) staff in accordance with the requirements of 10 CFR 50.72. Guidance concerning these reporting requirements, and example events, are provided in NUREG-1022. Certain events reportable under the provisions of 10 CFR 50.72 may also require the declaration of an emergency.

In order to align each Initiating Conditions (IC) with the appropriate ECL, it was necessary to determine the attributes of each ECL. The goal of this process is to answer the question, "What events or conditions should be placed under each ECL?" The following sources provided information and context for the development of ECL attributes.

- Assessments of the effects and consequences of different types of events and conditions
- DAEC abnormal and emergency operating procedure setpoints and transition criteria
- DAEC Technical Specification limits and controls
- Offsite Dose Assessment Manual (ODAM) radiological release limits
- Review of selected Updated Final Safety Analysis Report (UFSAR) accident analyses
- Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs)
- NUREG 0654, Appendix 1, *Emergency Action Level Guidelines for Nuclear Power Plants*
- Industry Operating Experience
- Input from DAEC subject matter experts

The following ECL attributes are used to aid in the development of ICs and Emergency Action Levels (EALs). The attributes may be useful in briefing and training settings (e.g., helping an Emergency Director understand why a particular condition is classified as an Alert).

3.1.1 Notification of Unusual Event (NOUE)

A Notification of Unusual Event, as defined in section 2.1.1, includes but is not limited to an event or condition that involves:

(A) A precursor to a more significant event or condition.

(B) A minor loss of control of radioactive materials or the ability to control radiation levels within the plant.

(C) A consequence otherwise significant enough to warrant notification to local, State and Federal authorities.

3.1.2 Alert

An Alert, as defined in section 2.1.2, includes but is not limited to an event or condition that involves:

(A) A loss or potential loss of either the fuel clad or Reactor Coolant System (RCS) fission product barrier.

(B) An event or condition that significantly reduces the margin to a loss or potential loss of the fuel clad or RCS fission product barrier.

(C) A significant loss of control of radioactive materials resulting in an inability to control radiation levels within the plant, or a release of radioactive materials to the environment that could result in doses greater than 1% of an EPA PAG at or beyond the site boundary.

(D) A HOSTILE ACTION occurring within the OWNER CONTROLLED AREA, including those directed at an Independent Spent Fuel Storage Installation (ISFSI).

3.1.3 Site Area Emergency (SAE)

A Site Area Emergency, as defined in section 2.1.3, includes but is not limited to an event or condition that involves:

(A) A loss or potential loss of any two fission product barriers - fuel clad, RCS and/or containment.

(B) A precursor event or condition that may lead to the loss or potential loss of multiple fission product barriers within a relatively short period of time. Precursor events and conditions of this type include those that challenge the monitoring and/or control of multiple SAFETY SYSTEMS.

(C) A release of radioactive materials to the environment that could result in doses greater than 10% of an EPA PAG at or beyond the site boundary.

(D) A HOSTILE ACTION occurring within the plant PROTECTED AREA.

3.1.4 General Emergency (GE)

A General Emergency, as defined in section 2.1.4, includes but is not limited to an event or condition that involves:

(A) Loss of any two fission product barriers AND loss or potential loss of the third barrier - fuel clad, RCS and/or containment.

(B) A precursor event or condition that, unmitigated, may lead to a loss of all three fission product barriers. Precursor events and conditions of this type include those that lead directly to core damage and loss of containment integrity.

(C) A release of radioactive materials to the environment that could result in doses greater than an EPA PAG at or beyond the site boundary.

(D) A HOSTILE ACTION resulting in the loss of key safety functions (reactivity control, core cooling/RPV water level or RCS heat removal) or damage to spent fuel.

3.1.5 Risk-Informed Insights

Emergency preparedness is a defense-in-depth measure that is independent of the assessed risk from any particular accident sequence; however, the development of an effective emergency classification scheme can benefit from a review of risk-based assessment results. To that end, the development and assignment of certain ICs and EALs also considered insights from several site-specific probabilistic safety assessments. Some generic insights from this review included:

1. Accident sequences involving a prolonged loss of all AC power are significant contributors to core damage frequency at many Boiling Water Reactors (BWRs). For this reason, a loss of all AC power for greater than 15 minutes, with the plant at or above Hot Shutdown, was assigned an ECL of Site Area Emergency. Precursor events to a loss of all AC power were also included as an Unusual Event and an Alert.

   A station blackout coping analyses performed in response to 10 CFR 50.63 and Regulatory Guide 1.155, *Station Blackout*, may be used to determine a time-based criterion to demarcate between a Site Area Emergency and a General Emergency. The time dimension is critical to a properly anticipatory emergency declaration since the goal is to maximize the time available for State and local officials to develop and implement offsite protective actions.

2. For severe core damage events, uncertainties exist in phenomena important to accident progressions leading to containment failure. Because of these uncertainties, predicting the status of containment integrity may be difficult under severe accident conditions. This is why maintaining containment integrity alone following sequences leading to severe core damage is an insufficient basis for not escalating to a General Emergency.

3. PSAs indicated that leading contributors to latent fatalities were sequences involving a containment bypass, a large Loss of Coolant Accident (LOCA) with early containment failure, a Station Blackout lasting longer than the DAEC coping period, and a reactor coolant pump seal failure. The generic EAL methodology needs to be sufficiently rigorous to address these sequences in a timely fashion.

3.2 TYPES OF INITIATING CONDITIONS AND EMERGENCY ACTION LEVELS

The NEI 99-01 methodology makes use of symptom-based, barrier-based and event-based ICs and EALs. Each type is discussed below.

Symptom-based ICs and EALs are parameters or conditions that are measurable over some range using plant instrumentation (e.g., core temperature, reactor coolant level, radiological effluent, etc.). When one or more of these parameters or conditions are off-normal, reactor operators will implement procedures to identify the probable cause(s) and take corrective action.

Fission product barrier-based ICs and EALs are the subset of symptom-based EALs that refer specifically to the level of challenge to the principal barriers against the release of radioactive material from the reactor core to the environment. These barriers are the fuel cladding, the reactor coolant system pressure boundary, and the containment. The barrier-based ICs and EALs consider the level of challenge to each individual barrier - potentially lost and lost - and the total number of barriers under challenge.

Event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. These include the failure of an automatic reactor scram to shut down the reactor, natural phenomena (e.g., an earthquake), or man-made hazards such as a toxic gas release.

3.3 DAEC-SPECIFIC ORGANIZATION AND PRESENTATION OF GENERIC INFORMATION

The scheme's generic information is organized by Recognition Category in the following order.

R - Abnormal Radiation Levels / Radiological Effluent – Section 6
C - Cold Shutdown / Refueling System Malfunction – Section 7
E - Independent Spent Fuel Storage Installation (ISFSI) – Section 8
F - Fission Product Barrier – Section 9
H - Hazards and Other Conditions Affecting Plant Safety – Section 10
S - System Malfunction – Section 11

Each Recognition Category section contains a matrix showing the ICs and their associated emergency classification levels.

The following information and guidance is provided for each IC:

ECL – the assigned emergency classification level for the IC.

Initiating Condition – provides a summary description of the emergency event or condition.

Operating Mode Applicability – Lists the modes during which the IC and associated EAL(s) are applicable (i.e., are to be used to classify events or conditions).

Emergency Action Level(s) – Provides examples of reports and indications that are considered to meet the intent of the IC.

For Recognition Category F, the fission product barrier thresholds are presented in tables and arranged by fission product barrier and the degree of barrier challenge (i.e., potential loss or loss). This presentation method shows the synergism among the thresholds, and supports accurate assessments.

Basis – Provides background information that explains the intent and application of the IC and EALs. In some cases, the basis also includes relevant source information and references.

3.4 IC AND EAL MODE APPLICABILITY

The DAEC emergency classification scheme was developed recognizing that the applicability of ICs and EALs will vary with plant mode. For example, some symptom-based ICs and EALs can be assessed only during the power operations, startup, or hot standby/shutdown modes of operation when all fission product barriers are in place, and plant instrumentation and SAFETY SYSTEMS are fully operational. In the cold shutdown and refueling modes, different symptom-based ICs and EALs will come into play to reflect the opening of systems for routine maintenance, the unavailability of some SAFETY SYSTEM components and the use of alternate instrumentation.

The following table shows which Recognition Categories are applicable in each plant mode. The ICs and EALs for a given Recognition Category are applicable in the indicated modes.

MODE APPLICABILITY MATRIX

	Recognition Category					
Mode	**R**	**C**	**E**	**F**	**H**	**S**
Power Operations	X		X	X	X	X
Startup	X		X	X	X	X
Hot Shutdown	X		X	X	X	X
Cold Shutdown	X	X	X		X	
Refueling	X	X	X		X	
Defueled	X	X	X		X	

DAEC Operating Modes

Power Operations (1):	Mode Switch in Run
Startup (2):	Mode Switch in Startup/Hot Standby or Refuel (with all vessel head closure bolts fully tensioned)
Hot Shutdown (3):	Mode Switch in Shutdown, Average Reactor Coolant Temperature >212 °F (with all vessel head closure bolts fully tensioned)
Cold Shutdown (4):	Mode Switch in Shutdown, Average Reactor Coolant Temperature ≤ 212 °F (with all vessel head closure bolts fully tensioned)
Refueling (5):	Mode Switch in Shutdown or Refuel (with one or more vessel head closure bolts less than fully tensioned)

4 DEVELOPMENT OF THE DAEC EMERGENCY CLASSIFICATION SCHEME

4.1 GENERAL DEVELOPMENT PROCESS

The DAEC ICs and EALs were developed to be unambiguous and readily assessable.

The IC is the fundamental event or condition requiring a declaration. The EAL(s) is the pre-determined threshold that defines when the IC is met.

Useful acronyms and abbreviations associated with the DAEC emergency classification scheme are presented in Appendix A, Acronyms and Abbreviations.

Many words or terms used in the DAEC emergency classification scheme have scheme-specific definitions. These words and terms are identified by being set in all capital letters (i.e., ALL CAPS). The definitions are presented in Appendix B, Definitions.

4.2 CRITICAL CHARACTERISTICS

When crafting the scheme, DAEC ensured that certain critical characteristics have been met. These critical characteristics are listed below.

- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are consistent with industry guidance; while the actual wording may be different, the classification intent is maintained. With respect to Recognition Category F, DAEC includes a user-aid to facilitate timely and accurate classification of fission product barrier losses and/or potential losses. The user-aid logic is consistent with the classification logic presented in Section 9.
- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are technically complete and accurate (i.e., they contain the information necessary to make a correct classification).
- EAL statements use objective criteria and observable values.
- ICs, EALs, Operating Mode Applicability and Note statements and formatting consider human factors and are user-friendly.
- The scheme facilitates upgrading and downgrading of the emergency classification where necessary.
- The scheme facilitates classification of multiple concurrent events or conditions.

4.3 INSTRUMENTATION USED FOR EALS

DAEC incorporated instrumentation that is reliable and routinely maintained in accordance with site programs and procedures. Alarms referenced in EAL statements are those that are the most operationally significant for the described event or condition.

EAL setpoints are within the calibrated range of the referenced instrumentation, and consider any automatic instrumentation functions that may impact accurate EAL assessment. In addition, EAL setpoint values do not use terms such as "off-scale low" or "off-scale high" since that type of reading may not be readily differentiated from an instrument failure.

4.4 EAL/THRESHOLD REFERENCES TO AOP AND EOP SETPOINTS/CRITERIA

Some of the criteria/values used in several EALs and fission product barrier thresholds are drawn from DAEC AOPs and EOPs. This approach is intended to maintain good alignment between operational diagnoses and emergency classification assessments. Appropriate administrative controls are in place to ensure that a subsequent change to an AOP or EOP is screened to determine if an evaluation pursuant to 10 CFR 50.54(q) is required.

5 GUIDANCE ON USING THE DAEC EALS

5.1 GENERAL CONSIDERATIONS

When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL.

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, Interim Staff Guidance, *Emergency Planning for Nuclear Power Plants*.

All emergency classification assessments should be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, validation could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel. The validation of indications should be completed in a manner that supports timely emergency declaration.

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. If an ongoing radiological release is detected and the release start time is unknown, it should be assumed that the release duration specified in the IC/EAL has been exceeded, absent data to the contrary.

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that 1) the activity proceeds as planned and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10 CFR 50.72.

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, RCS leak rate calculation, etc.); the EAL and/or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. This scheme provides the Emergency Director with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Emergency Director will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated into the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

5.2 CLASSIFICATION METHODOLOGY

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL(s) must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, then the IC is considered met and the associated ECL is declared in accordance with plant procedures.

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01.

5.3 CLASSIFICATION OF MULTIPLE EVENTS AND CONDITIONS

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. Additionally, there is no "additive" effect from multiple EALs meeting the same ECL. For example:

> If two Alert EALs are met, an Alert should be declared.

Related guidance concerning classification of rapidly escalating events or conditions is provided in Regulatory Issue Summary (RIS) 2007-02, *Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events.*

5.4 CONSIDERATION OF MODE CHANGES DURING CLASSIFICATION

The mode in effect at the time that an event or condition occurred, and prior to any plant or operator response, is the mode that determines whether or not an IC is applicable. If an event or condition occurs, and results in a mode change before the emergency is declared, the emergency classification level is still based on the mode that existed at the time that the event or condition was initiated (and not when it was declared). Once a different mode is reached, any new event or condition, not related to the original event or condition, requiring emergency classification should be evaluated against the ICs and EALs applicable to the operating mode at the time of the new event or condition.

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Hot Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

5.5 CLASSIFICATION OF IMMINENT CONDITIONS

Although EALs provide specific thresholds, the Emergency Director must remain alert to events or conditions that could lead to meeting or exceeding an EAL within a relatively short period of time (i.e., a change in the ECL is IMMINENT). If, in the judgment of the Emergency Director, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all emergency classification levels, this approach is particularly important at the higher emergency classification levels since it provides additional time for implementation of protective measures.

5.6 EMERGENCY CLASSIFICATION LEVEL UPGRADING AND DOWNGRADING

An ECL may be downgraded when the event or condition that meets the highest IC and EAL no longer exists, and other site-specific downgrading requirements are met. If downgrading the ECL is deemed appropriate, the new ECL would then be based on a lower applicable IC(s) and EAL(s). The ECL may also simply be terminated.

The following approach to downgrading or terminating an ECL is recommended.

ECL	Action When Condition No Longer Exists
Unusual Event	Terminate the emergency in accordance with plant procedures.
Alert	Downgrade or terminate the emergency in accordance with plant procedures.
Site Area Emergency with no long-term plant damage	Downgrade or terminate the emergency in accordance with plant procedures.
Site Area Emergency with long-term plant damage	Terminate the emergency and enter recovery in accordance with plant procedures.
General Emergency	Terminate the emergency and enter recovery in accordance with plant procedures.

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02.

5.7 CLASSIFICATION OF SHORT-LIVED EVENTS

As discussed in Section 3.2, event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. By their nature, some of these events may be short-lived and, thus, over before the emergency classification assessment can be completed. If an event occurs that meets or exceeds an EAL, the associated ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include a failure of the reactor protection system to automatically scram the reactor followed by a successful manual scram or an earthquake.

5.8 CLASSIFICATION OF TRANSIENT CONDITIONS

Many of the ICs and/or EALs contained in this document employ time-based criteria. These criteria will require that the IC/EAL conditions be present for a defined period of time before an emergency declaration is warranted. In cases where no time-based criterion is specified, it is recognized that some transient conditions may cause an EAL to be met for a brief period of time (e.g., a few seconds to a few minutes). The following guidance should be applied to the classification of these conditions.

<u>EAL momentarily met during expected plant response</u> - In instances where an EAL is briefly met during an expected (normal) plant response, an emergency declaration is not warranted provided that associated systems and components are operating as expected, and operator actions are performed in accordance with procedures.

EAL momentarily met but the condition is corrected prior to an emergency declaration -- If an operator takes prompt manual action to address a condition, and the action is successful in correcting the condition prior to the emergency declaration, then the applicable EAL is not considered met and the associated emergency declaration is not required. For illustrative purposes, consider the following example.

> An ATWS occurs and the auxiliary feedwater system fails to automatically start. Steam generator levels rapidly decrease and the plant enters an inadequate RCS heat removal condition (a potential loss of both the fuel clad and RCS barriers). If an operator manually starts the auxiliary feedwater system in accordance with an EOP step and clears the inadequate RCS heat removal condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event; emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations where an operator is able to take a successful corrective action prior to the Emergency Director completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

5.9 AFTER-THE-FACT DISCOVERY OF AN EMERGENCY EVENT OR CONDITION

In some cases, an EAL may be met but the emergency classification was not made at the time of the event or condition. This situation can occur when personnel discover that an event or condition existed which met an EAL, but no emergency was declared, and the event or condition no longer exists at the time of discovery. This may be due to the event or condition not being recognized at the time or an error that was made in the emergency classification process.

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR 50.72 within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

5.10 RETRACTION OF AN EMERGENCY DECLARATION

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022.

6 ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

RU1

ECL: Notification of Unusual Event

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the ODAM limits for 60 minutes or longer.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

RU1.1 Reading on **ANY** Table R-1 effluent radiation monitor greater than column "NOUE" for 60 minutes or longer:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
Gaseous	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
Liquid	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
Liquid	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

RU1.2 Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

RU1.3 Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODAM limits for 60 minutes or longer.

Definitions:

None

Basis:

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

DAEC incorporates design features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, and to control and monitor intentional releases. The occurrence of an extended, uncontrolled radioactive release to the environment is indicative of degradation in these features and/or controls.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

EAL RU1.1 - This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.

EAL RU1.2 - This EAL addresses radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. This EAL will typically be associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

EAL RU1.3 - This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analysis or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).

Escalation of the emergency classification level would be via IC RA1.

RU2

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel.

Operating Mode Applicability: All

Emergency Action Levels:

RU2.1 a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:

- Report to control room (visual observation)
- Fuel pool level indication (LI-3413) less than 36 feet and lowering
- WR GEMAC Floodup indication (LI-4541) coming on scale

AND

b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors.

- Spent Fuel Pool Area, RI-9178
- North Refuel Floor, RI-9163
- New Fuel Vault Area, RI-9153
- South Refuel Floor, RI-9164
- NW Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REFUELING PATHWAY: The reactor refueling cavity, spent fuel pool and fuel transfer canal.

Basis:

This IC addresses a decrease in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level decrease will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations. A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an UNPLANNED loss of water level.

During preparation for reactor cavity flood up prior to entry into refuel mode, reactor vessel level instrument LI-4541 (WR GEMAC, FLOODUP) on control room panel 1C04 is placed in service by I&C personnel connecting a compensating air signal after the reference leg is disconnected from the reactor head. Normal refuel water level is above the top of the span of this flood up level indicator. A valid indication (e.g., not due to loss of compensating air signal or other instrument channel failure) of reactor cavity level coming on span for this instrument is used at DAEC as an indicator of uncontrolled reactor cavity level decrease.

DAEC Technical Specifications require a minimum of 36 feet of water in the spent fuel pool when moving irradiated fuel into the secondary containment. During refueling, the gates between the reactor cavity and the refueling cavity are removed and the spent fuel pool level indicator LI- 3413 is used to monitor refueling water level. Procedures require that a normal refueling water level be maintained at 37 feet 5 inches. A low level alarm actuates when spent fuel pool level drops below 37 feet 1 inch. Symptoms of inventory loss at DAEC include visual observation of decreasing water levels in reactor cavity or spent fuel storage pool, Reactor Building (RB) fuel storage pool radiation monitor or refueling area radiation monitor alarms, observation of a decreasing trend on the spent fuel pool water level indicator, and actuation of the spent fuel pool low water level alarm. To eliminate minor level perturbations from concern, DAEC uses LI-3413 indicated water level below 36 feet and lowering.

Increased radiation levels can be detected by the local area radiation monitors surrounding the spent fuel pool and refueling cavity areas. Applicable area radiation monitors are those listed in AOP 981.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RA2.

RA1

ECL: Alert

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RA1.1 Reading on **ANY** Table R-1 effluent radiation monitor greater than column "Alert" for 15 minutes or longer:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
Gaseous	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
Liquid	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
Liquid	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

RA1.2 Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond SITE BOUNDARY. [Preferred]

RA1.3 Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for one hour of exposure.

RA1.4 Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

This IC is modified by a note that EAL RA1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

ECL: Alert

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel.

Operating Mode Applicability: All

Emergency Action Levels:

RA2.1 Uncovery of irradiated fuel in the REFUELING PATHWAY.

RA2.2 Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by Hi Rad alarm for **ANY** of the following ARMs:

- Spent Fuel Pool Area, RI-9178
- North Refuel Floor, RI-9163
- New Fuel Vault Area, RI-9153
- South Refuel Floor, RI-9164

OR

Reading greater than 5 R/hr on **ANY** of the following radiation monitors (in Mode 5 only):

- NW Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

RA2.3 Lowering of spent fuel pool level to 25.17 feet.

Definitions:

REFUELING PATHWAY – The reactor refueling cavity, spent fuel pool and fuel transfer canal.

Basis:

This IC addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

Expected radiation monitor alarm(s) during preplanned transfer of highly radioactive material through the affected areas are not considered valid alarms for the purpose of comparison to these EALs.

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

EAL RA2.1

This EAL escalates from RU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used. Classification of an event using this EAL should be based on the totality of available indications, reports, and observations.

While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

EAL RA2.2

This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. An alarm on these radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).

Threshold values for the Drywell monitors are only applicable in Mode 5 since the calculated radiation levels from damage to irradiated fuel would be masked by the typical background levels on these monitors during plant operation, and mechanical damage to a fuel assembly in the vessel can only happen with the reactor head removed.

EAL RA2.3

Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.

Escalation of the emergency classification level would be via ICs RS1 or RS2.

ECL: Alert

Initiating Condition: Radiation levels that impede access to areas necessary for normal plant operation.

Operating Mode Applicability: All

Emergency Action Levels:

RA3.1 Dose rate greater than 15 mR/hr in **ANY** of the following areas:

- Control Room (RM-9162)
- Central Alarm Station (by survey)

Definitions:

None

Basis:

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Director should consider the cause of the increased radiation levels and determine if another IC may be applicable.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

ECL: Site Area Emergency

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RS1.1 Reading on **ANY** Table R-1 effluent radiation monitor greater than column "SAE" for 15 minutes or longer:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

RS1.2 Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]

RS1.3 Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

This IC is modified by a note that EAL RS1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions. However, if Kaman monitor readings are sustained for 15 minutes or longer and the required MIDAS dose assessments cannot be completed within this period, then the declaration can be made using Kaman readings PROVIDED the readings are not from an isolated flow path.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. If Kaman readings are not valid, field survey results may be utilized to assess this IC using EAL RS1.3.

Escalation of the emergency classification level would be via IC RG1.

ECL: Site Area Emergency

Initiating Condition: Spent fuel pool level at 16.36 feet.

Operating Mode Applicability: All

Emergency Action Levels:

RS2.1 Lowering of spent fuel pool level to 16.36 feet.

Definitions:

None

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

It is recognized that this IC would likely not be met until well after another Site Area Emergency IC was met; however, it is included to provide classification diversity.

Escalation of the emergency classification level would be via IC RG1 or RG2.

RG1

ECL: General Emergency

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RG1.1 Reading on **ANY** Table R-1 effluent radiation monitor greater than column "GE" for 15 minutes or longer:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
Gaseous	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
Liquid	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
Liquid	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

RG1.2 Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]

RG1.3 Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.

Definitions:

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

Basis:

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

This IC is modified by a note that EAL RG1.1 is only assessed for emergency classification until a qualified dose assessor is performing assessments using dose projection software incorporating actual meteorological data and current radiological conditions. However, if Kaman monitor readings are sustained for 15 minutes or longer and the required MIDAS dose assessments cannot be completed within this period, then the declaration can be made using Kaman readings PROVIDED the readings are not from an isolated flow path.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. If Kaman readings are not valid, field survey results may be utilized to assess this IC using EAL RG1.3.

RG2

ECL: General Emergency

Initiating Condition: Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer.

Operating Mode Applicability: All

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

RG2.1 Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer.

Definitions:

None

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another General Emergency IC was met; however, it is included to provide classification diversity.

7 COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of RPV inventory for 15 minutes or longer.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CU1.1 UNPLANNED loss of reactor coolant results in RPV level less than a required lower limit for 15 minutes or longer.

CU1.2 a. RPV level cannot be monitored.

AND

b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RPV level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

EAL CU1.1 recognizes that the minimum required RPV level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

EAL CU1.2 addresses a condition where all means to determine RPV level have been lost. If all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RSC inventory loss was occurring by observing sump and Suppression Pool level changes. The drywell floor and equipment drain sumps, reactor building equipment and floor drain sumps receive all liquid waste from floor and equipment drains inside the primary containment and reactor building. A rise in Suppression Pool water level may be indicative of valve misalignment or leakage in systems that discharge to the Torus. Sump and Suppression Pool level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

ECL: Notification of Unusual Event

Initiating Condition: Loss of all but one AC power source to essential buses for 15 minutes or longer.

Operating Mode Applicability: 4, 5, Defueled

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU2.1 a. AC power capability to 1A3 and 1A4 buses is reduced to a single power source for 15 minutes or longer.

AND

b. Any additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of essential buses being fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED increase in RCS temperature.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CU3.1 UNPLANNED increase in RCS temperature to greater than 212°F.

CU3.2 Loss of **ALL** RCS temperature and RPV level indication for 15 minutes or longer.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

Basis:

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or the inability to determine RCS temperature and level, represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

EAL CU3.1 involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

EAL CU3.2 reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

ECL: Notification of Unusual Event

Initiating Condition: Loss of Vital DC power for 15 minutes or longer.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CU4.1 Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the Vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category R.

ECL: Notification of Unusual Event

Initiating Condition: Loss of all onsite or offsite communications capabilities.

Operating Mode Applicability: 4, 5, Defueled

Emergency Action Levels:

CU5.1 Loss of **ALL** of the following onsite communication methods:

- Plant Operations Radio System
- In-Plant Phone System
- Plant Paging System (Gaitronics)

CU5.2 Loss of **ALL** of the following offsite response organization communications methods:

- DAEC All-Call phone
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system
- FTS Phone system

CU5.3 Loss of **ALL** of the following NRC communications methods:

- FTS Phone system
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system

Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to offsite response organizations and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

EAL CU5.1 addresses a total loss of the communications methods used in support of routine plant operations.

EAL CU5.2 addresses a total loss of the communications methods used to notify all offsite response organizations of an emergency declaration. The offsite response organizations referred to here are the State of Iowa, Linn County, and Benton County.

EAL CU5.3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

CA1

ECL: Alert

Initiating Condition: Loss of RPV inventory.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CA1.1 Loss of RPV inventory as indicated by level less than 119.5 inches.

CA1.2 a. RPV level cannot be monitored for 15 minutes or longer

AND

b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool due to a loss of RPV inventory.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For EAL CA1.1, a lowering of water level below 119.5 inches indicates that operator actions have not been successful in restoring and maintaining RPV water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

Although related, EAL CA1.1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

For EAL CA1.2, the inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, the operators would need to determine that RCS inventory loss was occurring by observing sump and Suppression Pool level changes. The drywell floor and equipment drain sumps, reactor building equipment and floor drain sumps receive all liquid waste from floor and equipment drains inside the primary containment and reactor building. A rise in Suppression Pool water level may be indicative of valve misalignment or leakage in systems that discharge to the Torus. Sump and Suppression Pool level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1

If the RPV inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

ECL: Alert

Initiating Condition: Loss of all offsite and all onsite AC power to essential buses for 15 minutes or longer.

Operating Mode Applicability: 4, 5, Defueled

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CA2.1 Loss of **ALL** offsite and **ALL** onsite AC Power to 1A3 and 1A4 buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via IC CS1 or RS1.

ECL: Alert

Initiating Condition: Inability to maintain the plant in cold shutdown.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CA3.1 UNPLANNED increase in RCS temperature to greater than 212°F for greater than the duration specified in Table C-2.

Table C-2 RCS Heat-up Duration Thresholds		
RCS Integrity	**CONTAINMENT CLOSURE Status**	**Heat-up Duration**
Intact	Not applicable	60 minutes*
Not intact	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

CA3.2 UNPLANNED RCS pressure increase greater than 10 psig due to a loss of RCS cooling.

Definitions:

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

Basis:

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

RCS integrity is intact when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams).

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact. The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because

1) the evaporated reactor coolant may be released directly into the Containment atmosphere and subsequently to the environment, and
2) there is reduced reactor coolant inventory above the top of irradiated fuel.

EAL CA3.2 provides a pressure-based indication of RCS heat-up.

Escalation of the emergency classification level would be via IC CS1 or RS1.

ECL: Alert

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

Notes:
- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

CA6.1 a. The occurrence of **ANY** of the Table C-3 hazardous events:

Table C-3 Hazardous Events
• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

AND

b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:

- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode,

OR

- The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Definitions:

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction, or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria CA6.1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

An event affecting a single-train SAFETY SYSTEM (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under CA6 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train SAFETY SYSTEM, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC RS1.

ECL: Site Area Emergency

Initiating Condition: Loss of RPV inventory affecting core decay heat removal capability.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CS1.1 a. CONTAINMENT CLOSURE not established.

AND

b. RPV level less than +64 inches

CS1.2 a. CONTAINMENT CLOSURE established.

AND

b. RPV level less than +15 inches

CS1.3 a. RPV level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **EITHER** of the following:

- Drywell Monitor (9184A/B) reading greater than 5.0 R/hr
- UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery

Definitions:

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses a significant and prolonged loss of RPV inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If reactor vessel level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified reactor vessel levels of EALs CS1.1.b and CS1.2.b reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

In the Cold Shutdown and Refueling Modes, LT/LI-4559, 4560, and 4561 (RX VESSEL NARROW RANGE LEVEL) instruments read up to 22" high due to hot calibrations. LI-4541 (WR GEMAC, FLOODUP) should be used in these Modes for comparison to EAL thresholds since it is calibrated cold and reads accurately. If normal means of RPV level indication are not available due to plant evolutions, redundant means of RPV level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

In EAL CS1.3.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RPV.

These EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.*

Escalation of the emergency classification level would be via IC CG1 or RG1.

CG1

ECL: General Emergency

Initiating Condition: Loss of RPV inventory affecting fuel clad integrity with containment challenged.

Operating Mode Applicability: 4, 5

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CG1.1 a. RPV level less than +15 inches for 30 minutes or longer.

AND

b. **ANY** indication from the Secondary Containment Challenge Table C-1.

CG1.2 a. RPV level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **EIHER** of the following:

- Drywell Monitor (9184A/B) reading greater than 5.0 R/hr.
- UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery.

AND

c. **ANY** indication from the Secondary Containment Challenge Table C-1.

Table C-1 Secondary Containment Challenge
• CONTAINMENT CLOSURE not established*
• Drywell Hydrogen or Torus Hydrogen greater than 6% **AND** Drywell Oxygen or Torus Oxygen greater than 5%
• UNPLANNED increase in containment pressure
• Secondary containment radiation monitors above max safe operating limits (MSOL) of EOP 3, Table 6

* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute time limit, then declaration of a General Emergency is not required.

Definitions:

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the inability to restore and maintain reactor vessel level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If reactor vessel level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

In EAL CG1.2.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

For EAL CG1.2.b, the calculated radiation level on the Drywell Monitors (9184A/B) is without the reactor head in place. Calculated in radiation levels with the reactor head in place are below the normal variation in background readings of these monitors.

The inability to monitor RPV level may be caused by instrumentation and/or power failures or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RPV.

For the Containment Challenge Table, Secondary Containment max safe operating (MSOL) limits from EOP 3 are defined as the highest parameter value at which neither: (1) equipment necessary for the safe shutdown of the plant will fail nor (2) personnel access necessary for the safe shutdown of the plant will be precluded.

These EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.*

8 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) ICS/EALS

ECL: Notification of Unusual Event

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: All

Emergency Action Levels:

E-HU1.1 Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by a radiation reading greater than the values shown on Table E-1 on the spent fuel cask.

Table E-1 Cask Dose Rates	
61BT DSC	
3 feet from HSM Surface	800 mrem/hr
Outside HSM Door – Centerline of DSC	200 mrem/hr
End Shield Wall Exterior	40 mrem/hr

Definition:

CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

Basis:

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

9 FISSION PRODUCT BARRIER ICS/EALS

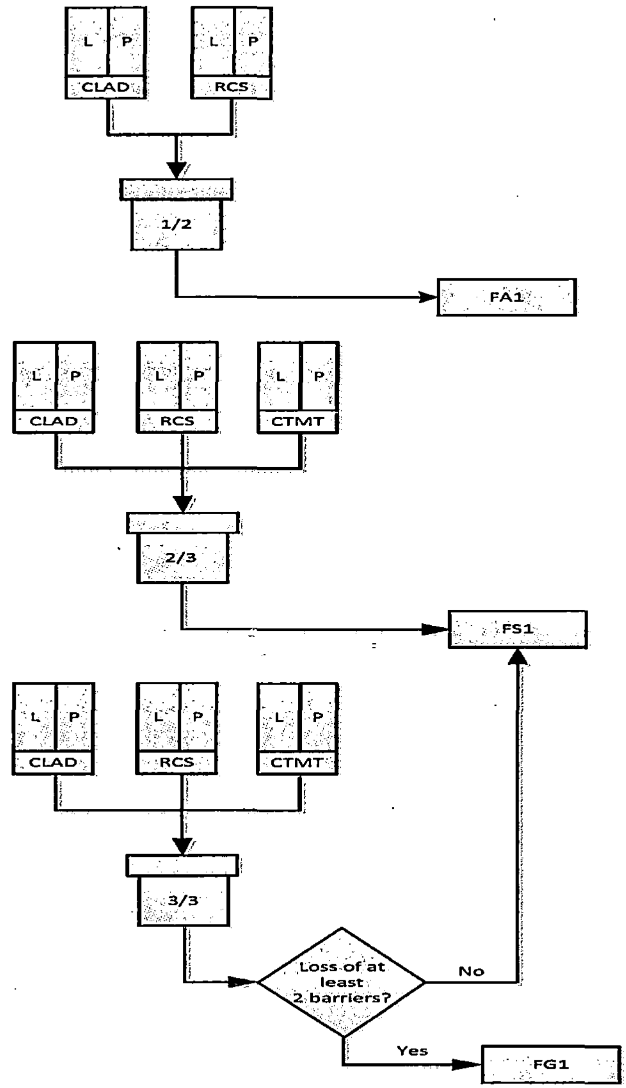

Table F-1: DAEC EAL Fission Product Barrier Table

Thresholds for LOSS or POTENTIAL LOSS of Barriers

FA1 ALERT	FS1 SITE AREA EMERGENCY	FG1 GENERAL EMERGENCY
ANY Loss **OR ANY** Potential Loss of **EITHER** the Fuel Clad **OR** RCS barrier.	Loss **OR** Potential Loss of **ANY** two barriers.	Loss of **ANY** two barriers **AND** Loss **OR** Potential Loss of the third barrier
Operating Mode Applicability: 1, 2, 3	**Operating Mode Applicability:** 1, 2, 3	**Operating Mode Applicability:** 1, 2, 3

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. RCS Activity		**1. Primary Containment Conditions**		**1. Primary Containment Conditions**	
A. Coolant activity greater than 300 μCi/gm dose equivalent I-131.	Not Applicable	A. Primary containment pressure greater than 2 psig due to RCS leakage.	Not Applicable	A. UNPLANNED rapid drop in Drywell pressure following Drywell pressure rise **OR** B. Drywell pressure response not consistent with LOCA conditions. **OR** C. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal **OR** D. Intentional primary containment venting per EOPs	A. Torus pressure greater than 53 psig **OR** B. Drywell or Torus H2 cannot be determined to be less than 6% and Drywell **OR** Torus O2 cannot be determined to be less than 5% **OR** C. HCL (Graph 4 of EOP 2) exceeded.

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
2. RPV Water Level		**2. RPV Water Level**		**2. RPV Water Level**	
A. SAG entry is required	A. RPV water level cannot be restored and maintained above +15 inches **OR** cannot be determined.	A. RPV water level cannot be restored and maintained above +15 inches **OR** cannot be determined.	Not Applicable	Not Applicable	A. SAG entry is required
3. Not Applicable		**3. RCS Leak Rate**		**3. Primary Containment Isolation Failure**	
Not Applicable	Not Applicable	A. UNISOLABLE break in Main Steam, HPCI, Feedwater, RWCU, or RCIC as indicated by the failure of both isolation valves in **ANY** one line to close **AND** **EITHER**: • High MSL flow or steam tunnel temperature annunciators **OR** • Direct report of steam release **OR** B. Emergency RPV Depressurization required.	A. UNISOLABLE primary system leakage that results in exceeding the Max Normal Operating Limit (MNOL) of EOP 3, Table 6 for **EITHER** of the following: • Temperature **OR** • Radiation Level	A. UNISOLABLE primary system leakage that results in exceeding the Max Safe Operating Limit (MSOL) of EOP 3, Table 6 for **EITHER** of the following: • Temperature **OR** • Radiation Level	Not Applicable

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
4. Primary Containment Radiation		**4. Primary Containment Radiation**		**4. Primary Containment Radiation**	
A. Drywell Monitor (9184A/B) reading greater than 1250 R/hr. **OR** B. Torus Monitor (9185A/B) reading greater than 125 R/hr	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 5 R/hr after reactor shutdown	Not Applicable	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 5000 R/hr. **OR** B. Torus Monitor (9185A/B) reading greater than 500 R/hr
5. Other Indications		**5. Other Indications**		**5. Other Indications**	
A. Fuel damage assessment indicates at least 5% fuel clad damage.	Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Fuel damage assessment indicates at least 20% fuel clad damage.
6. Emergency Director Judgment		**6. Emergency Director Judgment**		**6. Emergency Director Judgment**	
A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis Information For
DAEC EAL Fission Product Barrier Table F-1

DAEC FUEL CLAD BARRIER THRESHOLDS:

The Fuel Clad barrier consists of the zircalloy or stainless steel fuel bundle tubes that contain the fuel pellets.

1. RCS Activity

Loss 1.A

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.

There is no Potential Loss threshold associated with RCS Activity.

2. RPV Water Level

Loss 2.A

The Loss threshold represents any EOP requirement for entry into the Severe Accident Guidelines.

This is identified in the BWROG EPGs/SAGs when adequate core cooling cannot be assured.

Potential Loss 2.A

This water level corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as RCS barrier Loss threshold 2.A. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of the RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

DAEC FUEL CLAD BARRIER THRESHOLDS (cont.):

This threshold is considered to be exceeded when, as specified in the EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term "cannot be restored and maintained above" means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value *cannot* be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation below the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA6 or SS6 will dictate the need for emergency classification.

Since the loss of ability to determine if adequate core cooling is being provided presents a significant challenge to the fuel clad barrier, a potential loss of the fuel clad barrier is specified.

3. Not Applicable (included for numbering consistency between barrier tables)

DAEC FUEL CLAD BARRIER THRESHOLDS (cont.):

4. Primary Containment Radiation

Loss 4.A and Loss 4.B

The Drywell and Torus radiation monitor readings correspond to an instantaneous release of all reactor coolant mass into the Drywell or Torus, assuming that reactor coolant activity corresponds to approximately 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor readings in this threshold are higher than that specified for RCS Barrier Loss threshold 4.A since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Radiation.

5. Other Indications

Loss 5.A

Results obtained from procedure PASAP 7.2, Fuel Damage Assessment, indicate at least 5% fuel clad damage.

There is no Potential Loss threshold associated with Other Indications.

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

DAEC RCS BARRIER THRESHOLDS:

The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping up to and including the isolation valves.

1. **Primary Containment Conditions**

   Loss 1.A

   2 psig is the drywell high pressure scram setpoint which indicates a LOCA by automatically initiating ECCS.

   There is no Potential Loss threshold associated with Primary Containment Pressure.

2. **RPV Water Level**

   Loss 2.A

   +15 inches corresponds to the top of active fuel (TAF) and is used in the EOPs to indicate challenge to core cooling.

   The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold 2.A. Thus, this threshold indicates a Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier and that appropriately escalates the emergency classification level to a Site Area Emergency.

   This threshold is considered to be exceeded when, as specified in the EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

   The term, "cannot be restored and maintained above," means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value *cannot* be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel but does not permit extended operation beyond the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

DAEC RCS BARRIER THRESHOLDS (cont.):

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA5 or SS5 will dictate the need for emergency classification.

There is no RCS Potential Loss threshold associated with RPV Water Level.

3. RCS Leak Rate

<u>Loss Threshold 3.A</u>

Large high-energy lines that rupture outside primary containment can discharge significant amounts of inventory and jeopardize the pressure-retaining capability of the RCS until they are isolated. If it is determined that the ruptured line cannot be promptly isolated from the Control Room, the RCS barrier Loss threshold is met.

<u>Loss Threshold 3.B</u>

Emergency RPV Depressurization in accordance with the EOPs is indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is performed, the plant operators are directed to open safety relief valves (SRVs) and keep them open. Even though the RCS is being vented into the suppression pool, a Loss of the RCS barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

<u>Potential Loss Threshold 3.A</u>

Potential loss of RCS based on primary system leakage outside the primary containment is determined from EOP temperature or radiation Max Normal Operating values in areas such as main steam line tunnel, RCIC, HPCI, etc., which indicate a direct path from the RCS to areas outside primary containment.

A Max Normal Operating Limit (MNOL) value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage from a primary system warrant an Alert classification. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

An UNISOLABLE leak which is indicated by MNOL values escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold 3.A (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

DAEC RCS BARRIER THRESHOLDS (cont.):

4. Primary Containment Radiation

Loss 4.A

The Drywell monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold 4.A since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with Primary Containment Radiation.

5. Other Indications

There are no Loss or Potential Loss thresholds associated with Other Indications.

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

DAEC CONTAINMENT BARRIER THRESHOLDS:

The Primary Containment Barrier includes the drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

1. Primary Containment Conditions

Loss 1.A and 1.B

Rapid UNPLANNED loss of drywell pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase indicates a loss of drywell integrity. Drywell pressure should increase as a result of mass and energy release into the primary containment from a LOCA. Thus, drywell pressure not increasing under these conditions indicates a loss of primary containment integrity.

These thresholds rely on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Loss 1.C

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS).

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Following the leakage of RCS mass into primary containment and a rise in primary containment pressure, there may be minor radiological releases associated with allowable primary containment leakage through various penetrations or system components. Minor releases may also occur if a primary containment isolation valve(s) fails to close but the primary containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of primary containment but should be evaluated using the Recognition Category R ICs.

Loss 1.D

EOPs may direct primary containment isolation valve logic(s) to be intentionally bypassed, even if offsite radioactivity release rate limits will be exceeded. Under these conditions with a valid primary containment isolation signal, the containment should also be considered lost if primary containment venting is actually performed.

Intentional venting of primary containment for primary containment pressure or combustible gas control to the secondary containment and/or the environment is a Loss of the Containment. Venting for primary containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.

DAEC CONTAINMENT BARRIER THRESHOLDS (cont.):

Potential Loss 1.A

The threshold pressure is the Torus internal design pressure. Structural acceptance testing demonstrates the capability of the primary containment to resist pressures greater than the internal design pressure. A pressure of this magnitude is greater than those expected to result from any design basis accident and, thus, represent a Potential Loss of the Containment barrier.

Potential Loss 1.B

If hydrogen concentration reaches or exceeds the lower flammability limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the primary containment, loss of the Containment barrier could occur.

Potential Loss 1.C

The Heat Capacity Limit (HCL) is the highest suppression pool temperature from which Emergency RPV Depressurization will not raise:

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,
  OR
- Suppression chamber pressure above Primary Containment Pressure Limit A, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCL is a function of RPV pressure, suppression pool temperature and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

DAEC CONTAINMENT BARRIER THRESHOLDS (cont.):

2. RPV Water Level

There is no Loss threshold associated with RPV Water Level.

Potential Loss 2.A

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold 2.A. The Potential Loss requirement for Primary Containment Flooding indicates adequate core cooling cannot be restored and maintained and that core damage is possible. BWR EPGs/SAGs specify the conditions that require primary containment flooding. When primary containment flooding is required, the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to restore and maintain adequate core cooling.

PRA studies indicate that the condition of this Potential Loss threshold could be a core melt sequence which, if not corrected, could lead to RPV failure and increased potential for primary containment failure. In conjunction with the RPV water level Loss thresholds in the Fuel Clad and RCS barrier columns, this threshold results in the declaration of a General Emergency.

3. Primary Containment Isolation Failure

These thresholds address incomplete containment isolation that allows an UNISOLABLE direct release to the environment.

Loss 3.A

The Max Safe Operating Limit (MSOL) for Temperature and Radiation Level are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. EOPs utilize these temperatures and radiation levels to establish conditions under which RPV depressurization is required.

The temperatures and radiation levels should be confirmed to be caused by RCS leakage from a primary system. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

In combination with RCS potential loss 3.A this threshold would result in a Site Area Emergency.

There is no Potential Loss threshold associated with RCS Leak Rate.

DAEC CONTAINMENT BARRIER THRESHOLDS (cont.):

4. Primary Containment Radiation

There is no Loss threshold associated with Primary Containment Radiation.

Potential Loss 4.A

The drywell radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the drywell, assuming that 20% of the fuel cladding has failed. The radiation monitor reading for the torus corresponds to an instantaneous release of all reactor coolant mass directly into the torus, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the emergency classification level to a General Emergency.

5. Other Indications

There is no Loss threshold associated with Other Indications

Potential Loss 5.A

Results obtained from procedure PASAP 7.2, Fuel Damage Assessment, indicate at least 25% fuel clad damage. PASAP 7.2 only shows whether fuel damage is greater than or less than 25%, thus this indication is not likely to be declared before containment barrier potential loss 4.A which indicates 20% fuel damage. However, this potential loss threshold adds an additional layer of diversity to the scheme.

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

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10 HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

HU1

ECL: Notification of Unusual Event

Initiating Condition: Confirmed SECURITY CONDITION or threat.

Operating Mode Applicability: All

Emergency Action Levels:

HU1.1 A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by DAEC Security Shift Supervision.

HU1.2 Notification of a credible security threat directed at DAEC.

HU1.3 A validated notification from the NRC providing information of an aircraft threat.

Definitions:

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR 73.71 or 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between DAEC Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and offsite response organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

EAL HU1.1 references DAEC Security Shift Supervision because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.390 information.

EAL HU1.2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with Abnormal Operating Procedure (AOP) 914, Security Events.

EAL HU1.3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with Abnormal Operating Procedure (AOP) 914, Security Events.

Emergency plans and implementing procedures are public documents; therefore, EALs do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information is contained in the Security Plan.

Escalation of the emergency classification level would be via IC HA1.

ECL: Notification of Unusual Event

Initiating Condition: Seismic event greater than OBE levels.

Operating Mode Applicability: All

Emergency Action Levels:

HU2.1 Seismic event greater than Operating Basis Earthquake (OBE) as indicated by receipt of the Amber Operating Basis Earthquake Light and the wailing seismic alarm on 1C35.

Definitions:

DESIGN BASIS EARTHQUAKE (DBE): A DBE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.

OPERATING BASIS EARTHQUAKE (OBE): An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.

Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Design Basis Earthquake (DBE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event. The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

OBE events are detected in accordance with AOP 901. The OBE is associated with a peak horizontal acceleration of ± 0.06g.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

ECL: Notification of Unusual Event

Initiating Condition: Hazardous events

Operating Mode Applicability: All

Emergency Action Levels:

Note: EAL HU3.4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

HU3.1 A tornado strike within the PROTECTED AREA.

HU3.2 Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.

HU3.3 Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).

HU3.4 A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

Definitions:

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL HU3.1 addresses a tornado striking (touching down) within the Protected Area.

EAL HU3.2 addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

EAL HU3.3 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

EAL HU3.4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

ECL: Notification of Unusual Event

Initiating Condition: FIRE potentially degrading the level of safety of the plant.

Operating Mode Applicability: All

Emergency Action Levels:

Notes:

- The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

HU4.1 a. A FIRE is **NOT** extinguished within 15-minutes of **ANY** of the following FIRE detection indications:
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

b. The FIRE is located within **ANY** Table H-1 plant rooms or areas

HU4.2 a. Receipt of a single fire alarm with no other indications of a FIRE.

AND

b. The FIRE is located within **ANY** Table H-1 plant rooms or areas

AND

c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.

HU4.3 A FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

HU4.4 A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Table H-1 Fire Areas
• 1G31 DG and Day Tank Rooms • 1G21 DG and Day Tank Rooms • Battery Rooms • Essential Switchgear Rooms • Cable Spreading Room • Torus Room • Intake Structure • Pumphouse • Drywell • Torus • NE, NW, SE Corner Rooms • HPCI Room • RCIC Room • RHR Valve Room • North CRD Area • South CRD Area • CSTs • Control Building • Remote Shutdown Panel 1C388 Area • Panel 1C55/56 Area • SBGT Room

Definitions:

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

EAL HU4.1

The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

EAL HU4.2

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then EAL HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15 minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

EAL HU4.3

In addition to a FIRE addressed by EAL HU4.1 or EAL HU4.2, a FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA.

EAL HU4.4

If a FIRE within the plant or ISFSI PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Basis-Related Requirements from Appendix R and NFPA-805

Criterion 3 of Appendix A to 10 CFR 50 states in part that "structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

The Nuclear Safety Goal ("NSG") in NFPA 805, Section 1.3.1 states, "The nuclear safety goal is to provide reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition."

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance because a safe shutdown success path, free of fire damage, must be available to meet the nuclear safety goals, objectives and performance criteria for a fire under any plant operational mode or configuration.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). Even though DAEC has adopted the alternate approach provided by NFPA-805 in lieu of the deterministic requirements of Appendix R, the 30-minutes to verify a single alarm as used in EAL HU4.2 is considered a reasonable amount of time to determine if an actual FIRE exists without presenting a challenge to the nuclear safety performance criteria.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

ECL: Notification of Unusual Event

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE.

Operating Mode Applicability: All

Emergency Action Levels:

HU6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a NOUE.

ECL: Alert

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.

Operating Mode Applicability: All

Emergency Action Levels:

HA1.1 A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the DAEC Security Shift Supervision.

HA1.2 A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

Definitions:

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

OWNER CONTROLLED AREA: The site property owned by or otherwise under the control of the licensee.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between DAEC Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering).

The Alert declaration will also heighten the awareness of Offsite Response Organizations, allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR 73.71 or 10 CFR 50.72.

EAL HA1.1 is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against the ISFSI which is located outside the plant PROTECTED AREA.

EAL HA1.2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and offsite response organizations are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with Abnormal Operating Procedure (AOP) 914, Security Events.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.

In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information is contained in the Security Plan.

Escalation of the emergency classification level would be via IC HS1.

ECL: Alert

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations.

Operating Mode Applicability: All

Emergency Action Level:

HA5.1 An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).

Definitions:

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS5.

HA6

ECL: Alert

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.

Operating Mode Applicability: All

Emergency Action Level:

HA6.1 Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Definitions:

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for an Alert.

ECL: Site Area Emergency

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: All

Emergency Action Levels:

HS1.1 A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.

Definitions:

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between DAEC Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize offsite response organization resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to a HOSTILE ACTION directed at the ISFSI PROTECTED AREA which is located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR 73.71 or 10 CFR 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information is contained in the Security Plan.

Escalation of the emergency classification level would be via IC HG1.

ECL: Site Area Emergency

Initiating Condition: Inability to control a key safety function from outside the Control Room.

Operating Mode Applicability: All

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

HS5.1 a. An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).

AND

b. Control of **ANY** of the following key safety functions is not reestablished within 20 minutes.

- Reactivity control
- RPV water level
- RCS heat removal

Definitions:

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the Remote Shutdown Panel (1C388) is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within 20 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

AOP 915, "Shutdown Outside Control Room" provides the following CAUTION – "*For Control Room evacuation as the result of a fire, transfer of control at panels 1C388, 1C389, 1C390, 1C391, and 1C392 is required to be completed within 20 minutes.*"

Escalation of the emergency classification level would be via IC FG1 or CG1.

ECL: Site Area Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.

Operating Mode Applicability: All

Emergency Action Level:

HS6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Definitions:

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a Site Area Emergency.

ECL: General Emergency

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility.

Operating Mode Applicability: All

Emergency Action Level:

HG1.1 a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.

AND

b. **EITHER** of the following has occurred:

1. **ANY** of the following safety functions cannot be controlled or maintained.

- Reactivity control
- RPV water level
- RCS heat removal

OR

2. Damage to spent fuel has occurred or is IMMINENT.

Definitions:

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

Basis:

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

Timely and accurate communications between the DAEC Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

Emergency plans and implementing procedures are public documents; therefore, EALs do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information is contained in the Security Plan.

ECL: General Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.

Operating Mode Applicability: All

Emergency Action Level:

HG6.1 Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Definitions:

HOSTILE ACTION: An act toward DAEC or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for a General Emergency.

11 SYSTEM MALFUNCTION ICS/EALS

ECL: Notification of Unusual Event

Initiating Condition: Loss of ALL offsite AC power capability to essential buses for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

SU1.1 Loss of **ALL** offsite AC power capability to 1A3 **AND** 1A4 buses for 15 minutes or longer.

Definitions:

None

Basis:

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC essential buses. This condition represents a potential reduction in the level of safety of the plant.

The intent of this EAL is to declare a Notification of Unusual Event when offsite power has been lost and both of the emergency diesel generators have successfully started and energized their respective 4kv essential bus.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the essential buses, whether or not the buses are powered from it.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Escalation of the emergency classification level would be via IC SA1.

ECL: Notification of Unusual Event

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

SU3.1 An UNPLANNED event results in the inability to monitor one or more of the Table S-1 parameters from within the Control Room for 15 minutes or longer.

Table S-1 Safety System Parameters
• Reactor power
• RPV Water Level
• RPV Pressure
• Primary Containment Pressure
• Suppression Pool Level
• Suppression Pool Temperature

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via IC SA3.

SU4

ECL: Notification of Unusual Event

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.

Operating Mode Applicability: 1, 2, 3

Emergency Action Levels:

SU4.1 Pretreatment Offgas System (RM-4104) Hi-Hi Radiation Alarm.

SU4.2 Sample analysis indicates that reactor coolant specific activity is greater than 2.0 µCi/gm dose equivalent I-131 for 12 hours or longer.

Definitions:

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

For EAL SU4.1, RM-4104 Hi-Hi Radiation Alarm has been chosen because it is operationally significant, is readily recognizable by the Control Room Operations Staff, and is set at a level corresponding to noble gas release rate, after 30-minute delay and decay of 1 Ci/sec.

For EAL SU4.2, coolant samples exceeding the 2.0 µCi/gm dose equivalent I-131concentration require prompt action by DAEC Technical Specifications and are representative of minor fuel cladding degradation.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

ECL: Notification of Unusual Event

Initiating Condition: RCS leakage for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Levels:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

SU5.1 RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer.

SU5.2 RCS identified leakage greater than 25 gpm for 15 minutes or longer.

SU5.3 Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.

Definitions:

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

EAL SU5.1 and EAL SU5.2 are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications).

EAL SU5.3 addresses a RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These EALs thus apply to leakage into the containment, a secondary-side system) or a location outside of containment.

The leak rate values for each EAL were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). EAL SU5.1 uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. A stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category R or F.

ECL: Notification of Unusual Event

Initiating Condition: Automatic or manual scram fails to shutdown the reactor.

Operating Mode Applicability: 1, 2

Emergency Action Levels:

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

SU6.1 a. An automatic scram did not shutdown the reactor.

AND

b. ANY of the following manual actions taken at 1C05 are successful in lowering reactor power below 5% power

- Manual Scram Pushbuttons
- Mode Switch to Shutdown
- Alternate Rod Insertion (ARI)

SU6.2 a. A manual scram did not shutdown the reactor.

AND

b. **EITHER** of the following:

1. ANY of the following subsequent manual actions taken at 1C05 are successful in lowering reactor power below 5% power.
    - Manual Scram Pushbuttons
    - Mode Switch to Shutdown
    - Alternate Rod Insertion (ARI)

OR

2. A subsequent automatic scram is successful in shutting down the reactor.

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor scram, operators will promptly initiate manual actions at the reactor control console to shutdown the reactor (e.g., initiate a manual reactor scram quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control console to shutdown the reactor (e.g., initiate a manual reactor scram using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control console".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the SAFETY SYSTEMS are designed (typically 3 to 5% power).

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

ECL: Notification of Unusual Event

Initiating Condition: Loss of ALL onsite or offsite communications capabilities.

Operating Mode Applicability: 1, 2, 3

Emergency Action Levels:

SU7.1 Loss of **ALL** of the following onsite communication methods:

- Plant Operations Radio System
- In-Plant Phone System
- Plant Paging System (Gaitronics)

SU7.2 Loss of **ALL** of the following offsite response organization communications methods:

- DAEC All-Call phone
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system
- FTS Phone system

SU7.3 Loss of **ALL** of the following NRC communications methods:

- FTS Phone system
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system

Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to offsite response organizations and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

EAL SU7.1 addresses a total loss of the communications methods used in support of routine plant operations.

EAL SU7.2 addresses a total loss of the communications methods used to notify all offsite response organizations of an emergency declaration. The offsite response organizations referred to here are the State of Iowa, Linn County, and Benton County.

EAL SU7.3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

ECL: Alert

Initiating Condition: Loss of ALL but one AC power source to essential buses for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

SA1.1 a. AC power capability to 1A3 and 1A4 buses is reduced to a single power source for 15 minutes or longer.

AND

b. **ANY** additional single power source failure will result in a loss of **ALL** AC power to SAFETY SYSTEMS.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of essential buses being fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

Escalation of the emergency classification level would be via IC SS1.

ECL: Alert

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

SA3.1 a. An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for 15 minutes or longer.

Table S-1 Safety System Parameters
• Reactor power
• RPV Water Level
• RPV Pressure
• Primary Containment Pressure
• Suppression Pool Level
• Suppression Pool Temperature

AND

b. **ANY** of the Table S-2 transient events are in progress.

Table S-2 Significant Transients
• Automatic or manual runback greater than 25% thermal reactor power
• Electrical load rejection greater than 25% full electrical load
• Reactor scram
• ECCS actuation
• Thermal power oscillations greater than 10%

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via ICs FS1 or IC RS1.

ECL: Alert

Initiating Condition: Automatic or manual scram fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.

Operating Mode Applicability: 1, 2

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

Emergency Action Level:

SA6.1 a. An automatic or manual scram did not shutdown the reactor.

AND

b. **ALL** of the following manual actions taken at 1C05 are not successful in lowering reactor power below 5% power

- Manual Scram Pushbuttons
- Mode Switch to Shutdown
- Alternate Rod Insertion (ARI)

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram. This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control consoles (e.g., locally opening breakers). Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles."

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shutdown the reactor is prolonged enough to cause a challenge to the RPV water level or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the SAFETY SYSTEMS are designed (typically 3 to 5% power).

ECL: Alert

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of the SAFETY SYSTEM, then this emergency classification is not warranted.

SA8.1 a. The occurrence of **ANY** of the Table S-3 hazardous events:

Table S-3 Hazardous Events
• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

AND

b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:

- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode,

OR

- The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Definitions:

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria SA8.1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

An event affecting a single-train SAFETY SYSTEM (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under SA8 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train SAFETY SYSTEM, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC FS1 or RS1.

ECL: Site Area Emergency

Initiating Condition: Loss of ALL offsite and ALL onsite AC power to essential buses for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

SS1.1 Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

ECL: Site Area Emergency

Initiating Condition: Loss of ALL Vital DC power for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

SS2.1 Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG2.

ECL: Site Area Emergency

Initiating Condition: Inability to shutdown the reactor causing a challenge to RPV water level or RCS heat removal.

Operating Mode Applicability: 1, 2

Emergency Action Levels:

SS6.1 a. An automatic or manual scram did not shutdown the reactor.

AND

b. ALL of the following manual actions taken at 1C05 are not successful in lowering reactor power below 5% power:

- Manual Scram Pushbuttons
- Mode Switch to Shutdown
- Alternate Rod Insertion (ARI)

AND

c. **EITHER** of the following conditions exist:

- RPV level cannot be restored and maintained above -25 inches.

OR

- HCL (Graph 4 of EOP 2) exceeded.

Definitions:

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shutdown the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the SAFETY SYSTEMS are designed (typically 3 to 5% power).

Escalation of the emergency classification level would be via IC RG1 or FG1.

ECL: General Emergency

Initiating Condition: Prolonged loss of ALL offsite and ALL onsite AC power to essential buses.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

SG1.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses.

AND

b. **EITHER** of the following:

- Restoration of at least one AC essential bus in less than 4 hours is not likely.

OR

- RPV level cannot be restored and maintained above -25 inches.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a prolonged loss of all power sources to AC essential buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC essential bus by the end of the 4 hour station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one essential bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

ECL: General Emergency

Initiating Condition: Loss of ALL AC and Vital DC power sources for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3

Emergency Action Level:

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

SG2.1 a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses for 15 minutes or longer.

AND

b. Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.

Definitions:

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

Basis:

This IC addresses a concurrent and prolonged loss of both AC and Vital DC power. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of Vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both AC and DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

APPENDIX A – ACRONYMS AND ABBREVIATIONS

AC	Alternating Current
AOP	Abnormal Operating Procedure
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
CNMT	Containment
DC	Direct Current
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECL	Emergency Classification Level
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EPG	Emergency Procedure Guideline
FEMA	Federal Emergency Management Agency
GE	General Emergency
HCL	Heat Capacity Limit
HPCI	High Pressure Coolant Injection
IC	Initiating Condition
ID	Inside Diameter
ISFSI	Independent Spent Fuel Storage Installation
Keff	Effective Neutron Multiplication Factor
LCO	Limiting Condition of Operation
LOCA	Loss of Coolant Accident
mR, mRem, mrem, mREM	milli-Roentgen Equivalent Man
MW	Megawatt
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NORAD	North American Aerospace Defense Command
NOUE	Notification Of Unusual Event
NUMARC[1]	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake
OCA	Owner Controlled Area
ODAM	Offsite Dose Assessment Manual
PA	Protected Area
PAG	Protective Action Guideline
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
PSIG	Pounds per Square Inch Gauge
R	Roentgen
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
Rem, rem, REM	Roentgen Equivalent Man

[1] NUMARC was a predecessor organization of the Nuclear Energy Institute (NEI).

RPS .... Reactor Protection System
RPV .... Reactor Pressure Vessel
RWCU .... Reactor Water Cleanup
SCBA .... Self-Contained Breathing Apparatus
SPDS .... Safety Parameter Display System
TEDE .... Total Effective Dose Equivalent
TAF .... Top of Active Fuel
TSC .... Technical Support Center
UFSAR .... Updated Final Safety Analysis Report

APPENDIX B – DEFINITIONS

The following definitions are taken from Title 10, Code of Federal Regulations, and related regulatory guidance documents.

Alert: Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.

General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Notification of Unusual Event (NOUE): Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Site Area Emergency: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.

The following are key terms necessary for overall understanding the DAEC emergency classification scheme.

Emergency Action Level (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Emergency Classification Level (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are:

Notification of Unusual Event (NOUE)
Alert
Site Area Emergency (SAE)
General Emergency (GE)

Fission Product Barrier Threshold: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

Selected terms used in Initiating Condition and Emergency Action Level statements are set in all capital letters (e.g., ALL CAPS). These words are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. This corresponds to the pressure boundary for the Dry Shielded Canister (DSC) shell (including the inner bottom cover plate) base metal and associated confinement boundary welds.

CONTAINMENT CLOSURE: Procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.

DESIGN BASIS EARTHQUAKE (DBE): A DBE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION: An act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

OPERATING BASIS EARTHQUAKE (OBE): An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.

OWNER CONTROLLED AREA: This term is typically taken to mean the site property owned by or otherwise under the control of the licensee.

PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

REFUELING PATHWAY: Includes all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the Company. UFSAR Figure 1.2-1 identifies the DAEC SITE BOUNDARY.

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

ATTACHMENT 3

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATING TO

LICENSE AMENDMENT REQUEST TSCR-166

UPDATED DEVIATIONS AND DIFFERENCES MATRIX

100 pages follow

UPDATED DAEC DEVIATIONS AND DIFFERENCES MATRIX

TABLE OF CONTENTS

GENERAL COMMENTS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
GLOBAL #1	References to NEI 99-01	Replaced with DAEC	Difference	Convert generic guidance to DAEC specific.	None
GLOBAL #2	Effective date	Replaced with **TBD, 2018**	Difference	Convert generic guidance to DAEC specific.	None
GLOBAL #3	Defined terms in Appendix B; Title Case	Defined terms in Appendix B; Upper Case	Difference	All defined terms in Appendix B used in the document are in upper case (CAPs) to indicate that the terms are defined.	None
GLOBAL #4	PWR specific references	PWR references removed	Difference	DAEC is a BWR	None
GLOBAL #5	Recognition Category A-Abnormal Radiation Levels/Radiological Effluent category and Emergency Action Levels; AU, AA, AS, and AG	Recognition Category R-Abnormal Radiation Levels/Radiological Effluent category and Emergency Action Levels; RU, RA, RS, and RG	Difference	DAEC implemented the optional designation of "R" for radiological related items to maintain continuity with previous practice at DAEC.	None
GLOBAL #6	Permanently Defueled Section	Deleted references to Permanently Defueled Station	Difference	Not Applicable to DAEC	None
GLOBAL #7	Acknowledgments, Notice and Executive Summary	Deleted	Difference	Not Applicable to DAEC	None
GLOBAL #8	Parameters or indications listed in EALs	Some parameters or indications listed in EALs were placed in tables or bulletized lists.	Difference	Tables or bullets were created to present DAEC-specific information in a manner familiar to and desired by scheme users.	None
GLOBAL #9	Site specific information or indication statements	"Site specific information or indications" were replaced with DAEC-specific information or indications where applicable.	Difference	Compliance with intent of the guidance.	None
GLOBAL #10	Operating Mode Applicability lists mode names (i.e., Power Operation, Startup)	Operating Mode Applicability lists mode numbers (i.e., 1, 2, etc.)	Difference	Mode numbers used for consistency with DAEC procedures and training.	None
GLOBAL #11	Developer's Notes	Developer's Notes deleted	Difference	Developer's notes are not reflected in the implementation of the EALs.	None
GLOBAL #12	Example EAL statement	"Example" deleted from statement	Difference	In adopting the EAL, the "example" status is no longer applicable.	None
GLOBAL #13	The following terms: "all, any, or, either" are sometimes capitalized and/or bolded in ICs and EALs	Consistently capitalized and bolded the following terms: "ALL, ANY, OR, EITHER" in ICs and EALs.	Difference	Capitalized and bolded conditional terms in ICs and EALs for consistency based on user feedback.	None
GLOBAL #14	Defined terms are only listed in APPENDIX B - DEFINITIONS	Defined terms are also listed as in separate section of each IC/EAL where the terms are used.	Difference	Aid to the user to present all needed information within the same section of the Basis document.	None
GLOBAL #15	Term "emergency buses"	Replaced with "essential buses"	Difference	Changed to reflect DAEC nomenclature	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
COVER PAGE	Development of Emergency Action Levels for Non-Passive Reactors	Duane Arnold Emergency Action Level Technical Bases Document	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
Introduction	Acknowledgments, Notice and Executive Summary	Deleted	Difference	Not Applicable to DAEC	None
TOC	1. Regulatory Background	1. Basis for Emergency Action Levels	Difference	Title change	None
TOC	1.1 Operating Reactors	1.1 Regulatory Background	Difference	Title change	None
TOC	1.2 Permanently Defueled Station	Deleted section	Difference	Not Applicable to DAEC	None
TOC	1.3 Independent Spent Fuel Storage Installation (ISFSI)	1.2 Independent Spent Fuel Storage Installation (ISFSI)	Difference	Re-numbered	None
TOC	1.4 NRC Order EA-12-051	1.3 NRC Order EA-12-051	Difference	Re-numbered	None
TOC	1.5 Applicability of Advance and Small Modular Reactor Designs	Deleted section	Difference	Not Applicable to DAEC	None
TOC	3.Design of the NEI 99-01 Emergency Classification Scheme	3. Design of the DAEC Emergency Classification Scheme	Difference	Title Change	None
TOC	3.3 NSSS Design Differences	Deleted section	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
TOC	3.4 Organization and Presentation of Generic Information	Changed to 3.3 DAEC 3.4 Organization and Presentation of Generic Information	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
TOC	4.0 Site-Specific Scheme Development	4.0 DAEC Scheme Development	Difference	Title change	None
TOC	4.4; 4.5; 4.6; 4.8	Deleted sections	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
TOC	4.7 Developer and User Feedback				None
TOC	Appendix C-Permanently Defueled Station ICs/EALs	Deleted section	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
1.1	Regulatory Background	Regulatory Background	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document and removed developer information	None
1.2	Permanently Defueled Station	Section deleted	Difference	Not Applicable to DAEC	None
1.3	1.3 Independent Spent Fuel Storage Installation (ISFSI)	1.2 Independent Spent Fuel Storage Installation (ISFSI)	Difference	Re-numbered section.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
1.4	1.4 NRC Order EA-12-051	1.3 NRC Order EA-12-051	Difference	Re-numbered and removed wording to add these readings (DAEC installation completed).	None
1.5	Applicability to Advanced and Small Modular Reactor Designs	Section deleted	Difference	Not Applicable to DAEC	None
2	KEY TERMINOLOGY USED IN NEI 99-01	KEY TERMINOLOGY USED IN DAEC EAL SCHEME	Difference	Minor changes to reflect DAEC-specific implementation.	None
3	DESIGN OF THE NEI 99-01 EMERGENCY CLASSIFICATION SCHEME	DESIGN OF THE DAEC EMERGENCY CLASSIFICATION SCHEME	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document	None
3.1	Assignment of Emergency Classification Levels (ECLs)	Assignment of Emergency Classification Levels (ECLs)	Difference	Changes made to adapt the generic NEI guidance to a DAEC-specific document, removed references to PWRs, and removed developer information.	None
3.2	Types of Initiating Conditions and Emergency Action Levels	Types of Initiating Conditions and Emergency Action Levels	Verbatim		None
3.3	Text referring to NSSS design differences for various types or plants; Developer guidance	Deleted	Difference	Guidance is now DAEC specific	None
3.4	Organization and Presentation of Generic Information	DAEC-Specific Organization and Presentation of Generic Information	Difference	Renumbered to 3.3, made DAEC-specific, and deleted developer information	None
3.5	Mode of Applicability Matrix; Typical BWR Operating Modes	Deleted "Permanently Defueled" section of matrix; replaced Typical BWR Operating Modes with DAEC-specific Operating Modes	Difference	Renumbered to 3.4, removed PWR information, removed permanently defueled, and inserted DAEC Operating Modes to comply with the document intent.	V1
4	Site Specific Scheme Development Guidance	Development of the DAEC Emergency Classification Scheme	Difference	Updated to reflect DAEC specific scheme development process.	None
5	GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS	GUIDANCE ON USING THE DAEC EALS	Difference	Guidance is now DAEC specific	None
6 - 11	Recognition Category IC/EAL Matrixes	removed	Difference	Matrixes were intended for use by EAL developers. Inclusion in licensee scheme is not desired.	None

ABNORMAL RAD LEVELS / RADIOACTIVE EFFLUENT ICS/EALS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AU1	Recognition Category: AU1	RU1	Difference	Global Comment #5	None
	Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.	Release of gaseous or liquid radioactivity greater than 2 times the ODAM limits for 60 minutes or longer.	Difference	Global Comment #9	None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AU1 (cont.)	(1) Reading on ANY effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer: (site-specific monitor list and threshold values corresponding to 2 times the controlling document limits)	(1) Reading on **ANY** Table R-1 effluent radiation monitor greater than column "NOUE" for 60 minutes or longer: [inserted Table R-1 of DAEC-specific radiation monitors and threshold values]	Difference	See Global Comments #8, 9, 12, & 13. Reworded EAL statement to remove operator confusion as to whether they needed to multiply the values of the following table by 2 or if the value provided already was 2X. Wording now matches wording of RS1 and RG1 allowing for easier operator progression through the EALs.	V2
	(2) Reading on ANY effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.	(2) Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.	Difference	Global Comment #13	None
	(3) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.	(3) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODAM limits for 60 minutes or longer.	Difference	Global Comment #9	None
				Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AU2	Recognition Category: AU2	RU2	Difference	Global Comment #5 & 14	None
	Initiating Condition: UNPLANNED loss of water level above irradiated fuel.	UNPLANNED loss of water level above irradiated fuel.	Verbatim		None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None
	(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following: (site-specific level indications). **AND**	(1) a UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following: • Report to control room (visual observation) • Fuel pool level indication (LI-3413) less than 36 feet and lowering • WR GEMAC Floodup indication (LI-4541) coming on scale **AND**	Difference	Global Comment #9, 12 & 13	V3

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AU2 (cont.)	b. UNPLANNED increase in area radiation levels as indicated by ANY of the following radiation monitors. (site-specific list of area radiation monitors)	b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors. • Spent Fuel Pool Area, RI-9178 • North Refuel Floor, RI-9163 • New Fuel Vault Area, RI-9153 • South Refuel Floor, RI-9164 • NW Drywell Area Hi Range Rad Monitor, RIM-9184A • South Drywell Area Hi Range Rad Monitor, RIM-9184B	Difference	Global Comments #9 & 13 Intent and meaning of the EALs are not altered.	V4

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AA1	Recognition Category: AA1	RA1	Difference	Global Comment #5 & 14	None
	Initiating condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.	Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.	Verbatim		None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None
	(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	(1) Reading on **ANY** Table R-1 radiation monitor greater than column "Alert" for 15 minutes or longer: [inserted Table R-1 of DAEC-specific radiation monitors and threshold values]	Difference	Global Comment #8, 9, 12 & 13	V5
	(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).	(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]	Difference	Global Comment #9 Added bracketed 'Preferred' to reinforce the 4th Note of the IC	None
	(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.	(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for one hour of exposure.	Difference	Global Comment #9	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AA1 (cont.)	(4) Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.	(4) Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AA2	Recognition Category: AA2	RA2	Difference	Global Comment #5 & 14	None
	Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel.	Significant lowering of water level above, or damage to, irradiated fuel.	Verbatim		None
	Operating Mode of Applicability: All	Operating Mode of Applicability: All	Verbatim		None
	(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.	(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.	Verbatim		None
	(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by ANY of the following radiation monitors: (site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms)	(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by Hi Rad alarm for **ANY** of the following ARMs: • Spent Fuel Pool Area, RI-9178 • North Refuel Floor, RI-9163 • New Fuel Vault Area, RI-9153 • South Refuel Floor, RI-9164 **OR** Reading greater than 5 R/hr on ANY of the following radiation monitors (in Mode 5 only): • NW Drywell Area Hi Range Rad Monitor, RIM-9184A • South Drywell Area Hi Range Rad Monitor, RIM-9184B	Difference	Global Comment #8, 9, 12 & 13 Threshold values for the Drywell monitors are only applicable in Mode 5 since the calculated radiation levels from damage to irradiated fuel would be masked by the typical background levels on these monitors during plant operation, and mechanical damage to a fuel assembly in the vessel can only happen with the reactor head removed (Mode 5).	V6
	(3) Lowering of spent fuel pool level to (site-specific Level 2 value). [See Developer Notes	(3) Lowering of spent fuel pool level to 25.17 feet	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V7

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AA3	Recognition Category: AA3	RA3	Difference	Global Comment #5 & 14	None
	Initiating Condition: Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.	Radiation levels that impede access to areas necessary for normal plant operation.	Difference	Reworded IC to reflect non-applicability of EAL #2.	None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Dose rate greater than 15 mR/hr in ANY of the following areas: • Control Room • Central Alarm Station • (other site-specific areas/rooms)	(1) Dose rate greater than 15 mR/hr in **ANY** of the following areas: • Control Room ARM (RM-9162) • Central Alarm Station (by survey)	Difference	Global Comment #9, 12 & 13	None
	(2) An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified)	Not used at DAEC	Difference	EALs RA3 and HA5 are not applicable to DAEC because an evaluation has shown that there are no rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. All areas outside the Control Room that contain equipment necessary for normal plant operation, cooldown and shutdown do not require physical access to operate. Intent and meaning of the EALs are not altered.	V8

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AS1	Recognition Category: AS1	RS1	Difference	Global Comment #5 & 14	None
	Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.	Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	(1) Reading on **ANY** Table R-1 effluent radiation monitor greater than column "SAE" for 15 minutes or longer. [inserted Table R-1 of DAEC-specific radiation monitors and threshold values]	Difference	Global Comment #8, 9, 12 & 13	V9
	(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond (site-specific dose receptor point).	(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]	Difference	Global Comment #3 & 9 Added bracketed 'Preferred' to reinforce the 4th Note of the IC	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AS1 (cont.)	(3) Field survey results indicate EITHER of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.	(3) Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.	Difference	Global Comment #3, 9, & 13 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AS2	Recognition Category: AS2	RS2	Difference	Global Comment #5	None
	Initiating Condition: Spent fuel pool level at (site-specific Level 3 description).	Spent fuel pool level at 16.36 feet	Difference	Global Comment #9	V10
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Lowering of spent fuel pool level to (site-specific Level 3 value).	(1) Lowering of spent fuel pool level to 16.36 feet	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V10

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AG1	Recognition Category: AG1	RG1	Difference	Global Comment #5 & 14	None
	Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.	Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	(1) Reading on ANY Table R-1 effluent radiation monitor greater than column "GE" for 15 minutes or longer. [inserted Table R-1 of DAEC-specific radiation monitors and threshold values]	Difference	Global Comment #8, 9, 12 & 13	V9
	(2) Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond (site-specific dose receptor point).	(2) Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]	Difference	Global Comment #3 & 9 Added bracketed 'Preferred' to reinforce the 4th Note of the IC	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AG1 (cont.)	(3) Field survey results indicate EITHER of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.	(3) Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.	Difference	Global Comment #3 & 9	None
				Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
AG2	Recognition Category: AG2	RG2	Difference	Global Comment #5	None
	Initiating Condition: Spent fuel pool level cannot be restored to at least (site-specific Level 3 description) for 60 minutes or longer.	Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer.	Difference	Global Comment #9	V10
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Spent fuel pool level cannot be restored to at least (site-specific Level 3 value) for 60 minutes or longer.	(1) Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V10

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CU1	Recognition Category: CU1	CU1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: UNPLANNED loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory for 15 minutes or longer.	UNPLANNED loss of RPV inventory for 15 minutes or longer	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) UNPLANNED loss of reactor coolant results in (reactor vessel/RCS [PWR] or RPV [BWR]) level less than a required lower limit for 15 minutes or longer.	(1) UNPLANNED loss of reactor coolant results in RPV level less than a required lower limit for 15 minutes or longer.	Difference	Global Comment #4 & 12	None
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored. AND	(2) a. RPV level cannot be monitored. **AND**	Difference	Global Comment #4	None
	b. UNPLANNED increase in (site-specific sump and/or tank) levels.	b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	None
CU2	Recognition Category: CU2	CU2	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer.	Loss of all but one AC power source to essential buses for 15 minutes or longer.	Difference	Global comment #15	None
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 4, 5, Defueled	Difference	Global Comment #10	None
	(1) a. AC power capability to (site-specific emergency buses) is reduced to a	(1) a. AC power capability to 1A3 and 1A4 buses is reduced to a single power	Difference	Global Comment #9, 12, & 13	V11

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
	single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.	source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS.		Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CU3	Recognition Category: CU3	CU3	Verbatim	Global Comment #11, 14	None
	Initiating Condition: UNPLANNED increase in RCS temperature.	UNPLANNED increase in RCS temperature.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit).	(1) UNPLANNED increase in RCS temperature to greater than 212°F	Difference	Global Comment #9 & 12	V1
	(2) Loss of ALL RCS temperature and (reactor vessel/RCS [PWR] or RPV [BWR]) level indication for 15 minutes or longer.	(2) Loss of **ALL** RCS temperature and RPV level indication for 15 minutes or longer	Difference	Global Comment #4 & 13 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CU4	Recognition Category: CU4	CU4	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of Vital DC power for 15 minutes or longer.	Loss of Vital DC power for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) Indicated voltage is less than (site-specific bus voltage value) on required Vital DC buses for 15 minutes or longer.	(1) Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer	Difference	Global Comment #9, 12, 13 Intent and meaning of the EALs are not altered.	V12

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CU5	Recognition Category: CU5	CU5	Verbatim		None
	Initiating Condition: Loss of all onsite or offsite communications capabilities.	Loss of all onsite or offsite communications capabilities.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 4, 5, Defueled	Difference	Global Comment #10	None
	(1) Loss of ALL of the following onsite communication methods: (site-specific list of communications methods)	(1) Loss of **ALL** of the following onsite communication methods: • Plant Operations Radio System • In-Plant Phone System • Plant Paging System (Gaitronics)	Difference	Global Comment #9, 12 & 13	V13
	(2) Loss of ALL of the following ORO communications methods: (site-specific list of communications methods)	(2) Loss of **ALL** of the following offsite response organization communications methods: • DAEC All-Call phone • All telephone lines (PBX and commercial) • Cell Phones (including fixed cell phone system) • Control Room fixed satellite phone system • FTS Phone system	Difference	Global Comment #9 & 13	V13 V14

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CU5 (cont.)	(3) Loss of ALL of the following NRC communications methods: (site-specific list of communications methods)	(3) Loss of **ALL** of the following NRC communications methods: • FTS Phone system • All telephone lines (PBX and commercial) • Cell Phones (including fixed cell phone system) • Control Room fixed satellite phone system	Difference	Global Comment #9, 12 & 13 Intent and meaning of the EALs are not altered.	V13

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CA1	Recognition Category: CA1	CA1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory.	Loss of RPV inventory.	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory as indicated by level less than (site-specific level).	(1) Loss of RPV inventory as indicated by level less than 119.5 inches	Difference	Global Comment #4, 9 & 12	V15
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 15 minutes or longer	(2) a. RPV level cannot be monitored for 15 minutes or longer	Difference	Global Comment #4	None
	AND b. UNPLANNED increase in (site-specific sump and/or tank) levels due to a loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory.	**AND** b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool due to a loss of RPV inventory.	Difference	Global Comment #4, 9 & 13 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CA2	Recognition Category: CA2	CA2	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.	Loss of all offsite and all onsite AC power to essential buses for 15 minutes or longer.	Difference	Global Comment #15	None
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 4, 5, Defueled	Difference	Global Comment #10	None
	(1) Loss of ALL offsite and ALL onsite AC Power to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite and **ALL** onsite AC Power to 1A3 and 1A4 for 15 minutes or longer.	Difference	Global Comment #9, 12 & 13 Intent and meaning of the EALs are not altered.	V12

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CA3	Recognition Category: CA3	CA3	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Inability to maintain the plant in cold shutdown.	Inability to maintain the plant in cold shutdown.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit) for greater than the duration specified in the following table.	(1) UNPLANNED increase in RCS temperature to greater than 212°F for greater than the duration specified in Table C-2.	Difference	Global Comment #9 & 12	V1
	Table: RCS Heat-up Duration Thresholds RCS Status | Containment Closure Status Intact (but not at reduced inventory [*PWR*]) | Not applicable Not intact (or at reduced inventory [*PWR*]) | Established Not intact (or at reduced inventory [*PWR*]) | Not Established * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.	Table C-2 RCS Heat-up Duration Thresholds RCS Integrity | Containment Closure Status | Heat-up Duration Intact | Not Applicable | 60 minutes* Not intact | Established | 20 minutes* Not intact | Not Established | 0 minutes * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.	Difference	Global Comment #4 Changed "RCS Status" to "RCS Integrity" to match current site nomenclature	None
	(2) UNPLANNED RCS pressure increase greater than (site-specific pressure reading). (This EAL does not apply during water-solid plant conditions. [PWR])	(2) UNPLANNED RCS pressure increase greater than 10 psig due to a loss of RCS cooling.	Difference	Global Comment #4 & 9 Added "due to a loss of RCS cooling" to clarify the intent of the EAL Intent and meaning of the EALs are not altered.	V16

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
CA6	Recognition Category: CA6	CA6	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Verbatim		None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager	(1) a. The occurrence of ANY of the Table C-3 hazardous events: **Table C-3 Hazardous Events** • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director	Difference	Global Comment #9, 12 & 13	None

CA6 (cont.)	**AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** 1. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	**AND** b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode. **AND** 2. **EITHER** of the following: • Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or • The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.	Deviation Deviation Difference	Adopted the revised EAL wording provided in approved EAL FAQ 2016-02. Adopted the revised EAL wording provided in approved EAL FAQ 2016-02. Added the following clarification to the Basis from EALFAQ 2018-04: An event affecting a single-train SAFETY SYSTEM (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under SA8 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train SAFETY SYSTEM, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement. Intent and meaning of the EALs are not altered.	V17 V17 V18

DAEC DEVIATIONS AND DIFFERENCES MATRIX

CS1	Recognition Category: CS1	CS1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory affecting core decay heat removal capability.	Loss of reactor vessel/RCS inventory affecting core decay heat removal capability.	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) a. CONTAINMENT CLOSURE not established. **AND** b. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level).	(1) a. CONTAINMENT CLOSURE not established. **AND** b. RPV level less than +64 inches	Difference	Global Comment #9 & 12	V19
	(2) a. CONTAINMENT CLOSURE established. **AND** b. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level).	(2) a. CONTAINMENT CLOSURE established. **AND** b. RPV level less than +15 inches	Difference	Global Comment #4 & 9	V19

DAEC DEVIATIONS AND DIFFERENCES MATRIX

CS1 (cont.)	(3) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 30 minutes or longer. **AND** b. Core uncovery is indicated by **ANY** of the following: • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [PWR] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications)	(3) a. RPV level cannot be monitored for 30 minutes or longer. **AND** b. Core uncovery is indicated by **ANY** of the following: • Drywell Monitor (9184A/B) reading greater than 5.0 R/hr • UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery	Difference Difference	Global Comment #4 Global Comment #9 &13	None V6
				Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

CG1	Recognition Category: CG1	CG1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory affecting fuel clad integrity with containment challenged.	Loss of reactor vessel/RCS inventory affecting fuel clad integrity with containment challenged.	Difference	Global Comment #4	None
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 4, 5	Difference	Global Comment #10	None
	(1) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level) for 30 minutes or longer. **AND** b. ANY indication from the Containment Challenge Table (see below).	(1) a. RPV level less than +15 inches for 30 minutes or longer. **AND** b. ANY indication from the Containment Challenge Table (see below).	Difference	Global Comment #4, 9, 12 & 13	V19
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 30 minutes or longer.	(2) a. RPV level cannot be monitored for 30 minutes or longer.	Difference	Global Comment #4	None

	AND b. Core uncovery is indicated by ANY of the following: • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [PWR] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery **AND**	**AND** b. Core uncovery is indicated by **ANY** of the following: • Drywell Monitor (9184A/B) reading greater than 5.0 R/hr • Erratic source range monitor indication • UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery	Difference	Global Comment #8, 9 & 13	V6
	c. **ANY** indication from the Containment Challenge Table (see below).	**AND** c. **ANY** indication from the Secondary Containment Challenge Table C-1.	Difference	Global Comment #9	None
	Containment Challenge Table ONTAINMENT CLOSURE not established* Explosive mixture) exists inside containment NPLANNED increase in containment pressure econdary containment radiation monitor reading ite specific value) [BWR]	**Table C-1 Containment Challenge Ta** • CONTAINMENT CLOSURE not established • Drywell Hydrogen or Torus Hydrogen gre **AND** Drywell Oxygen or Torus Oxygen gre • UNPLANNED increase in containment pre • Secondary containment radiation monito safe operating limits (MSOL) of EOP 3, Tal	Difference	Global Comment #9	V20 V21
	* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.	*If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.	Verbatim	Intent and meaning of the EALs are not altered.	

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) ICS/EALS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>DAEC</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="4">E-HU1</td><td>Recognition Category: E-HU1</td><td>E-HU1</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.</td><td>Damage to a loaded cask CONFINEMENT BOUNDARY.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: All</td><td>Operating Mode Applicability: All</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than (2 times the site-specific cask specific technical specification allowable radiation level) on the surface of the spent fuel cask.</td><td>(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by a radiation reading greater than the values shown on Table E-1 on the spent fuel cask.
<table><tr><th colspan="2">Table E-1 Cask Dose Rates</th></tr><tr><td>61BT DSC</td><td>800 mrem/hr</td></tr><tr><td>3 feet from HSM Surface</td><td>200 mrem/hr</td></tr><tr><td>Outside HSM Door – Centerline of DSC</td><td>40 mrem/hr</td></tr></table></td><td>Difference</td><td>Global Comment #8, 9, 12 & 14

Intent and meaning of the EALs are not altered.</td><td>V22</td></tr>
</table>

FISSION PRODUCT BARRIER ICS/EALS

The following section is configured in a manner that is different from the Fission Product Barrier Tables in the DAEC EAL Technical Bases Document. Where the Technical Bases Document evaluates all three fission product barriers simultaneously for a specific sub-category, this matrix presents each fission product barrier individually for all sub-categories. The significance of this presentation is that where the fission product barrier table in the Technical Bases Document moves vertically through the sub-categories, this matrix moves horizontally.

DAEC DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th colspan="6">Fission Product Barrier Emergency Classifications</th><th></th></tr>
<tr><th colspan="3">NEI 99-01 Rev. 6</th><th>DAEC</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td colspan="3">Table 9-F-1: Recognition Category "F" Initiating Condition Matrix</td><td rowspan="4">Deleted</td><td rowspan="4">Difference</td><td rowspan="4">Deleted per developer note. Mode applicability carried over onto Table 9-F EAL listing.
Global Comment #11</td><td rowspan="4">None</td></tr>
<tr><td>Alert</td><td>Site Area Emergency</td><td>General Emergency</td></tr>
<tr><td>Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.</td><td>Loss or Potential Loss of any two barriers.</td><td>Loss of any two barriers and Loss or Potential Loss of the third barrier.</td></tr>
<tr><td>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</td><td>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</td><td>Op Modes: Power Operation, Hot Standby, Startup, Hot Shutdown</td></tr>
<tr><td colspan="3">Table 9-F-2: BWR EAL Fission Product Barrier Table Thresholds for LOSS or POTENTIAL LOSS of Barriers</td><td>Table F-1: DAEC EAL Fission Product Barrier Table Thresholds for LOSS or POTENTIAL LOSS of Barriers</td><td>Difference</td><td>Renumbered and re-labeled due to deletion of Tables 9-F-1 & 3.
Added Global Comment #9</td><td>None</td></tr>
<tr><td colspan="3">Table 9-F-3: PWR EAL Fission Product Barrier Table Thresholds for LOSS or POTENTIAL LOSS of Barriers</td><td>Deleted</td><td>Difference</td><td>Global Comment #4</td><td>None</td></tr>
<tr><td colspan="3">Basis Information For BWR EAL Fission Product Barrier Table 9-F Developer Notes.</td><td>Deleted Developer Notes</td><td>Difference</td><td>Transform generic NEI 99-01 guidance into DAEC-specific application.</td><td>None</td></tr>
<tr><td colspan="3">Figure 9-F-4: PWR Containment Integrity or Bypass Example</td><td>Deleted</td><td>Difference</td><td>Global Comment #4</td><td>None</td></tr>
</table>

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Fuel Clad Barrier						
Table 9-F	**NEI 99-01 Rev. 6**		**DAEC**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
1. RCS Activity	A. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).	Not Applicable	A. Coolant activity greater than 300 µCi/gm dose equivalent I-131	Not Applicable	Difference	General Comment #9
2. RPV Water Level	A. Primary containment flooding required.	A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to the top of active fuel) or cannot be determined.	A. SAG entry is required.	A. RPV water level cannot be restored and maintained above +15 inches **OR** cannot be determined.	Difference	EPFAQ 2015-004 V15 General Comment #9, 13
3. Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Verbatim	None
4. Primary Containment Radiation	A. Primary containment radiation monitor reading greater than (site-specific value).	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 1250 R/hr. **OR** B. Torus Monitor (9185A/B) reading greater than 125 R/hr	Not Applicable	Difference	V23 Global Comment #9
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	A. Fuel damage assessment indicates at least 5% fuel clad damage.	Not Applicable	Difference	Global Comment #9 Core damage assessment procedure.

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Fuel Clad Barrier						
Table 9-F	**NEI 99-01 Rev. 6**		**DAEC**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	Verbatim	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of RCS Barrier						
Table 9-F	NEI 99-01 Rev. 6		DAEC		Change	Justification
Sub-Category	Loss	Potential Loss	Loss	Potential Loss		
1. Primary Containment Pressure Renamed to **1. Primary Containment Conditions**	A. Primary containment pressure greater than (site-specific value) due to RCS leakage.	Not Applicable	A. Primary containment pressure greater than 2 psig due to RCS leakage.	Not Applicable	Difference	V24 Global Comment #9
2. RPV Water Level	A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to the top of active fuel) or cannot be determined.	Not Applicable	A. RPV water level cannot be restored and maintained above +15 inches **OR** cannot be determined.	Not Applicable	Difference	V19 Global Comment #9, 13
3. RCS Leak Rate	A. UNISOLABLE break in **ANY** of the following: (site-specific systems with potential for high-energy line breaks) **OR** B. Emergency RPV Depressurization.	A. UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following: 1. Max Normal Operating Temperature **OR** 2. Max Normal Operating Area Radiation Level.	A. UNISOLABLE break in Main Steam, HPCI, Feedwater, RWCU, or RCIC as indicated by the failure of both isolation valves in **ANY** one line to close **AND EITHER:** • High MSL flow or steam tunnel temperature annunciators **OR** • Direct report of steam release **OR** B. Emergency RPV Depressurization required.	A. UNISOLABLE primary system leakage that results in exceeding the Max Normal Operating Limit (MNOL) of EOP 3, Table 6 for **EITHER** of the following: • Temperature **OR** • Radiation Level	Difference	V25 Global Comment #9 Added site-specific indication of an unisolable steam line break which includes failure of both isolation valves to LOSS 3.A.

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of RCS Barrier						
4. Primary Containment Radiation	A. Primary containment radiation monitor reading greater than (site-specific value).	Not Applicable	A. Drywell Monitor (9184A/B) reading greater than 5 R/hr after reactor shutdown	Not Applicable	Difference	Global Comment #9 V23
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	Not Applicable	Not Applicable	Difference	Global Comment #9
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	Verbatim	None

Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier						
Table 9-F-2	NEI 99-01 Rev. 6		DAEC		Change	Justification
Sub-Category	Loss	Potential Loss	Loss	Potential Loss		
1. Primary Containment Conditions	A. UNPLANNED rapid drop in primary containment pressure following primary containment pressure rise **OR** B. Primary containment pressure response not consistent with LOCA conditions.	A. Primary containment pressure greater than (site-specific value) **OR** B. (site-specific explosive mixture) exists inside primary containment **OR** C. HCTL exceeded.	A. UNPLANNED rapid drop in Drywell pressure following Drywell pressure rise **OR** B. Drywell pressure response not consistent with LOCA conditions. **OR** C. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal **OR** D. Intentional primary containment venting per EOPs	A. Torus pressure greater than 53 psig **OR** B. Drywell or Torus H2 cannot be determined to be less than 6% and Drywell **OR** Torus O2 cannot be determined to be less than 5% **OR** C. HCL (Graph 4 of EOP 2) exceeded.	Difference	Global Comment #9 V20 V26 V27 Primary Containment Isolation Failure Loss 3.A and 3.B moved to sub-category 1 "Primary Containment Conditions" as Losses 1.C and 1.D to consolidate concepts into single sub-category
2. RPV Water Level	Not Applicable	A. Primary containment flooding required.	Not Applicable	A. SAG entry is required.	Difference	EPFAQ 2015-004

DAEC DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th colspan="7">Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier</th></tr>
<tr><th>Table 9-F-2</th><th colspan="2">NEI 99-01 Rev. 6</th><th colspan="2">DAEC</th><th>Change</th><th>Justification</th></tr>
<tr><th>Sub-Category</th><th>Loss</th><th>Potential Loss</th><th>Loss</th><th>Potential Loss</th><th></th><th></th></tr>
<tr><td>3. Primary Containment Isolation Failure</td><td>A. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal
OR
B. Intentional primary containment venting per EOPs
OR
C. UNISOLABLE primary system leakage that results in exceeding EITHER of the following:
1. Max Safe Operating Temperature.
OR
2. Max Safe Operating Area Radiation Level.</td><td>Not Applicable</td><td>A. UNISOLABLE primary system leakage that results in exceeding the Max Safe Operating Limit (MSOL) of EOP 3, Table 6 for EITHER of the following:
• Temperature
OR
• Radiation Level</td><td>Not Applicable</td><td>Difference</td><td>Global Comment #9
V28

Primary Containment Isolation Failure Loss 3.A and 3.B moved to sub-category 1 "Primary Containment Conditions" as Losses 1.C and 1.D to consolidate concepts into single sub-category</td></tr>
<tr><td>4. Primary Containment Radiation</td><td>Not Applicable</td><td>A. Primary containment radiation monitor reading greater than (site-specific value).</td><td>Not Applicable</td><td>A. Drywell Monitor (9184A/B) reading greater than 5000 R/hr.
OR
B. Torus Monitor (9185A/B) reading greater than 500 R/hr</td><td>Difference</td><td>Global Comment #9
V23</td></tr>
</table>

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**DAEC**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	Not Applicable	A. Fuel damage assessment indicates at least 20% fuel clad damage.	Difference	Global Comment #9 Core damage assessment procedure.
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	B. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	**C.** **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	D. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	Verbatim	None

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU1	Recognition Category: HU1	HU1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Confirmed SECURITY CONDITION or threat.	Confirmed SECURITY CONDITION or threat.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).	(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by DAEC Security Shift Supervision.	Difference	Global Comment #9 & 12	None
	(2) Notification of a credible security threat directed at the site.	(2) Notification of a credible security threat directed at DAEC.	Difference	Global Comment #9	None
	(3) A validated notification from the NRC providing information of an aircraft threat.	(3) A validated notification from the NRC providing information of an aircraft threat.	Verbatim	None	None
				Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU2	Recognition Category: HU2	HU2	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Seismic event greater than OBE levels.	Seismic event greater than OBE levels.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by: (site-specific indication that a seismic event met or exceeded OBE limits)	(1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by receipt of the Amber Operating Basis Earthquake Light and the wailing seismic alarm on 1C35.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V29

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU3	Recognition Category: HU3	HU3	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Hazardous event.	Hazardous event.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A tornado strike within the PROTECTED AREA.	(1) A tornado strike within the PROTECTED AREA.	Verbatim	Global Comment #12	None
	(2) Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	(2) Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	Verbatim		None
	(3) Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	(3) Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	Verbatim		None
	(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	Verbatim		None
	(5) (Site-specific list of natural or technological hazard events)		Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU4	Recognition Category: HU4	HU4	Verbatim	Global Comment #11, 14	None
	Initiating Condition: FIRE potentially degrading the level of safety of the plant.	FIRE potentially degrading the level of safety of the plant.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) a. A FIRE is NOT extinguished within 15-minutes of ANY of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND**	(1) a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND**	Difference	Global Comment #12 & 13	None
	b. The FIRE is located within ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas)	b. The FIRE is located within **ANY** Table H-1 plant rooms or areas. **Table H-1 Fire Areas** • 1G31 DG and Day Tank Rooms • 1G21 DG and Day Tank Rooms • Battery Rooms • Essential Switchgear Rooms • Cable Spreading Room • Torus Room • Intake Structure • Pumphouse • Drywell • Torus • NE, NW, SE Corner Rooms • HPCI Room • RCIC Room • RHR Valve Room • North CRD Area • South CRD Area • CSTs • Control Building • Remote Shutdown Panel 1C388 Area • Panel 1C55/56 Area • SBGT Room	Difference	Global Comment #8, 9, & 13	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU4 (cont.)	(2) a. Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND** b. The FIRE is located within ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas) **AND**	(2) a. Receipt of a single fire alarm with no other indications of a FIRE. **AND** b. The FIRE is located within **ANY** Table H-1 plant rooms or areas. **AND**	Difference	Global Comment #8, 9 & 13	None
	c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.	c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.	Verbatim	N/A	None
	(3) A FIRE within the plant or ISFSI [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.	(3) A FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60 minutes of the initial report, alarm or indication.	Difference	Global Comment #9	None
	(4) A FIRE within the plant or ISFSI [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	(4) A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	Difference	Global Comment #9	None
				Basis revised to include NFPA-805 in the discussion of Appendix R basis for the EAL thresholds. Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HU7	Recognition Category: HU7	HU7	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO) UE.	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE.	Difference	NOUE versus (NO)UE, DAEC uses the full NOUE term	None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.	Verbatim	Global Comment #3, 12, 14	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HA1	Recognition Category: HA1	HA1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision). (2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the DAEC Security Shift Supervision. (2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	Difference Verbatim	Global Comment #9, 12, 14 Intent and meaning of the EALs are not altered.	None

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HA5	Recognition Category: HA5	Not used at DAEC	Difference	EALs RA3 and HA5 are not applicable to DAEC because an evaluation has shown that there are no rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. All areas outside the Control Room that contain equipment necessary for normal plant operation, cooldown and shutdown do not require physical access to operate.	V8
	Initiating Condition: Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.	Not used at DAEC	Difference		None
	Operating Mode Applicability: All	Not used at DAEC	Difference		None
	(1) a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified) **AND** b. Entry into the room or area is prohibited or impeded.	Not used at DAEC	Difference		None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HA6	Recognition Category: HA6	HA5	Difference	Renumbered to align with other similar ICs	None
	Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations.	Control Room evacuation resulting in transfer of plant control to alternate locations.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).	(1) An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V30

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HA7	Recognition Category: HA7	HA6	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.	Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	(1) Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	Verbatim	Global Comment #12 Intent and meaning of the EALs are not altered	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HS1	Recognition Category: HS1	HS1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.	HOSTILE ACTION within the PROTECTED AREA.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).	(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HS6	Recognition Category: HS6	HS5	Difference	Renumbered to align with other similar ICs	None
	Initiating Condition: Inability to control a key safety function from outside the Control Room.	Inability to control a key safety function from outside the Control Room.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that (site specific number of) minutes has been exceeded, or will likely be exceeded.	**Note:** The Emergency Director should declare the Site Area Emergency promptly upon determining that 20 minutes has been exceeded, or will likely be exceeded.		Global Comment #9	V30
	(1) a. An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).	(1) a. An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).	Difference	Global Comment #9, 12	None
	AND b. Control of **ANY** of the following key safety functions is not reestablished within (site-specific number of minutes). • Reactivity control • Core cooling [PWR] / RPV water level [BWR] • RCS heat removal	**AND** b. Control of **ANY** of the following key safety functions is not reestablished within 20 minutes. • Reactivity control • RPV water level • RCS heat removal	Difference	Global Comment #4, 9 Intent and meaning of the EALs are not altered.	V30

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HS7	Recognition Category: HS7	Recognition Category: HS6	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: ALL	Verbatim		None
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	Verbatim	Global Comment #12 Intent and meaning of the EALs are not altered	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HG1	Recognition Category: HG1	HG1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility.	HOSTILE ACTION resulting in loss of physical control of the facility.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).	(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.	Difference	Global Comment #9, 12	None
	AND b. **EITHER** of the following has occurred: 1. **ANY** of the following safety functions cannot be controlled or maintained. • Reactivity control • Core cooling [PWR] / RPV water level [BWR] • RCS heat removal	**AND** b. **EITHER** of the following has occurred: 1. **ANY** of the following safety functions cannot be controlled or maintained. • Reactivity control • RPV water level • RCS heat removal	Difference	Global Comment #4, 9	None
	OR 2. Damage to spent fuel has occurred or is IMMINENT.	**OR** 2. Damage to spent fuel has occurred or is IMMINENT.	Verbatim	Intent and meaning of the EALs are not altered.	None

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
HG7	Recognition Category: HG7	HG6	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.	Verbatim		None
	Operating Mode Applicability: All	Operating Mode Applicability: All	Verbatim		None
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	Verbatim	Global Comment #12 Intent and meaning of the EALs are not altered	None

SYSTEM MALFUNCTION ICS/EALS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU1	Recognition Category: SU1	SU1	Verbatim		None
	Initiating Condition: Loss of all offsite AC power capability to emergency buses for 15 minutes or longer.	Loss of ALL offsite AC power capability to essential buses for 15 minutes or longer.	Difference	Global Comment #15	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) Loss of **ALL** offsite AC power capability to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite AC power capability to 1A3 AND 1A4 buses for 15 minutes or longer.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>DAEC</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="5">SU2</td><td>Recognition Category: SU2</td><td>SU3</td><td>Difference</td><td>Global Comment #11, 14
Renumbered to align with other similar ICs</td><td>None</td></tr>
<tr><td>Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.</td><td>UNPLANNED loss of Control Room indications for 15 minutes or longer.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown</td><td>Operating Mode Applicability: 1, 2, 3</td><td>Difference</td><td>Global Comment #10</td><td>None</td></tr>
<tr><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.</td><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the Table S-1 parameters from within the Control Room for 15 minutes or longer.</td><td>Difference</td><td>Global Comment #12</td><td>None</td></tr>
<tr><td><table>
<tr><th>[BWR parameter list]</th><th>[PWR parameter list]</th></tr>
<tr><td>Reactor Power</td><td>Reactor Power</td></tr>
<tr><td>RPV Water Level</td><td>RCS Level</td></tr>
<tr><td>RPV Pressure</td><td>RCS Pressure</td></tr>
<tr><td>Primary Containment Pressure</td><td>In-Core/Core Exit Temperature</td></tr>
<tr><td>Suppression Pool Level</td><td>Levels in at least (site-specific number) two steam generators</td></tr>
<tr><td>Suppression Pool Temperature</td><td>Steam Generator Auxiliary or Emergency Feed Water Flow</td></tr>
</table></td><td>Table S-1 Safety System Parameters
• Reactor Power
• RPV Water Level
• RPV Pressure
• Primary Containment Pressure
• Suppression Pool Level
• Suppression Pool Temperature</td><td>Difference</td><td>Global Comment #4, 9

Intent and meaning of the EALs are not altered.</td><td>None</td></tr>
</table>

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU3	Recognition Category: SU3	SU4	Verbatim	Global Comment #11, 14R Renumbered IC to align with other similar ICs	None
	Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.	Reactor coolant activity greater than Technical Specification allowable limits.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) (Site-specific radiation monitor) reading greater than (site-specific value).	(1) Pretreatment Offgas System (RM-4104) Hi-Hi Radiation Alarm	Difference	Global Comment #9 & 12	None
	(2) Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.	(2) Sample analysis indicates that reactor coolant specific activity is greater than 2.0 µCi/gm dose equivalent I-131 for 12 hours or longer.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V31

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU4	Recognition Category: SU4	SU5	Verbatim	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: RCS leakage for 15 minutes or longer.	RCS leakage for 15 minutes or longer.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) RCS unidentified or pressure boundary leakage greater than (site-specific value) for 15 minutes or longer.	(1) RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer.	Difference	Global Comment #9 & 12	V32
	(2) RCS identified leakage greater than (site-specific value) for 15 minutes or longer.	(2) RCS identified leakage greater than 25 gpm for 15 minutes or longer.	Difference	Global Comment #9	V32
	(3) Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.	(3) Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.	Verbatim	Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU5	Recognition Category: SU5	SU6	Verbatim	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor.	Automatic or manual scram fails to shutdown the reactor.	Difference	Global Comment #4	None
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1, 2	Difference	Global Comment #10 DAEC can be up to 12% power in STARTUP Mode, so Mode 2 applicability added	V33
	(1) a. An automatic (trip [PWR] / scram [BWR]) did not shutdown the reactor.	(1) a. An automatic scram did not shutdown the reactor.	Difference	Global Comment #4 & 12	None
	AND b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	**AND** b. **ANY** of the following manual actions taken at 1C05 are successful in lowering reactor power below 5% power • Manual Scram Pushbuttons • Mode Switch to Shutdown • Alternate Rod Insertion (ARI)	Difference	Global Comment #9	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU5 (cont.)	(2) a. A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor.	(2) a. A manual scram did not shutdown the reactor.	Difference	Global Comment #4	None
	AND b. EITHER of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	**AND** b. 1. **EITHER** of the following subsequent manual actions taken at 1C05 <u>are successful</u> in lowering reactor power below 5% power • Manual Scram Pushbuttons • Mode Switch to Shutdown • Alternate Rod Insertion (ARI)	Difference	Global Comment #9	None
	OR 2. A subsequent automatic (trip [PWR] / scram [BWR]) is successful in shutting down the reactor.	**OR** 2. A subsequent automatic scram is successful in shutting down the reactor.	Difference	Global Comment #4 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU6	Recognition Category: SU6	SU7	Verbatim	Global Comment #14 Renumbered to align with other similar ICs	None
	Initiating Condition: Loss of all onsite or offsite communications capabilities.	Loss of **ALL** onsite or offsite communications capabilities.	Difference	Global Comment #13	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) Loss of ALL of the following Onsite communication methods: (site-specific list of communications methods)	(1) Loss of **ALL** of the following Onsite communication methods: • Plant Operations Radio System • In-Plant Phone System • Plant Paging System (Gaitronics)	Difference	Global Comment #9, 12 & 13	V16
	(2) Loss of ALL of the following ORO communications methods: (site-specific list of communications methods)	(2) Loss of **ALL** of the following offsite response organization communications methods: • DAEC All-Call phone • All telephone lines (PBX and commercial) • Cell Phones (including fixed cell phone system) • Control Room fixed satellite phone system • FTS Phone system	Difference	Global Comment #9 & 13	V13, V14

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU6 (cont.)	(3) Loss of ALL of the following NRC communications methods: (site-specific list of communications methods)	(4) Loss of **ALL** of the following NRC communications methods: • FTS Phone system • All telephone lines (PBX and commercial) • Cell Phones (including fixed cell phone system) • Control Room fixed satellite phone system	Difference	Global Comment #9 & 13 Intent and meaning of the EALs are not altered.	V13, V14

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SU7	Recognition Category: SU7 Initiating Condition: Failure to isolate containment or loss of containment pressure control. [PWR] Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown (1) a. Failure of containment to isolate when required by an actuation signal. **AND** b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal. (1) a. Containment pressure greater than (site-specific pressure). **AND** b. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.	Not Applicable	Difference	Global Comment #4 This IC and EALs are only applicable to PWR plants.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SA1	Recognition Category: SA1	SA1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer.	Loss of **ALL** but one AC power source to essential buses for 15 minutes or longer.	Difference	Global Comment #15	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. AND b. Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS.	(1) a. AC power capability to 1A3 and 1A4 buses is reduced to a single power source for 15 minutes or longer. **AND** a. **ANY** additional single power source failure will result in a loss of **ALL** AC power to SAFETY SYSTEMS.	Difference Difference	Global Comment #9, 12 Global Comment #13 Intent and meaning of the EALs are not altered.	None None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

<table>
<tr><th>Section</th><th>NEI 99-01 Rev. 6</th><th>DAEC</th><th>Change</th><th>Justification</th><th>Validation #</th></tr>
<tr><td rowspan="5">SA2</td><td>Recognition Category: SA2</td><td>SA3</td><td>Difference</td><td>Global Comment #11, 14
Renumbered to align with other similar ICs</td><td>None</td></tr>
<tr><td>Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.</td><td>UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.</td><td>Verbatim</td><td></td><td>None</td></tr>
<tr><td>Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown</td><td>Operating Mode Applicability: 1, 2, 3</td><td>Difference</td><td>Global Comment #10</td><td>None</td></tr>
<tr><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.</td><td>(1) a. An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for 15 minutes or longer</td><td>Verbatim</td><td>Global Comment #12</td><td>None</td></tr>
<tr><td><table>
<tr><td>[BWR parameter list]</td><td>[PWR parameter list]</td></tr>
<tr><td>Reactor Power</td><td>Reactor Power</td></tr>
<tr><td>RPV Water Level</td><td>RCS Level</td></tr>
<tr><td>RPV Pressure</td><td>RCS Pressure</td></tr>
<tr><td>Primary Containment Pressure</td><td>In-Core/Core Exit Temperature</td></tr>
<tr><td>Suppression Pool Level</td><td>Levels in at least (site-specific number) steam generators</td></tr>
<tr><td>Suppression Pool Temperature</td><td>Steam Generator Auxiliary or Emergency Feed Water Flow</td></tr>
</table>
AND</td><td>Table S-1 Safety System Parameters
• Reactor Power
• RPV Water Level
• RPV Pressure
• Primary Containment Pressure
• Suppression Pool Level
• Suppression Pool Temperature

AND</td><td>Difference</td><td>Global Comment #4, 8</td><td>None</td></tr>
</table>

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SA2 (cont.)	b. **ANY** of the following transient events in progress. • Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load • Reactor scram [BWR] / trip [PWR] • ECCS (SI) actuation • Thermal power oscillations greater than 10% [BWR]	b. **Any** of the Table S-2 transient events are in progress **Table S-2 Significant Transients** • Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load • Reactor scram • ECCS actuation • Thermal power oscillations greater than 10%	Difference	Global Comment #4, 9 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SA5	Recognition Category: SA5	SA6	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	Automatic or manual scram fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	Difference	Global Comment #4 & 9	None
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1, 2	Difference	Global Comment #10 DAEC can be up to 12% power in STARTUP Mode, so Mode 2 applicability added	V33
	(1) a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor.	(1) a. An automatic or manual scram did not shutdown the reactor.	Difference	Global Comment #4, 9 & 12	None
	AND b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	**AND** b. **ALL** of the following manual actions taken at 1C05 are not successful in lowering reactor power below 5% power • Manual Scram Pushbuttons • Mode Switch to Shutdown • Alternate Rod Insertion (ARI)	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SA9	Recognition Category: SA9	SA8	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Verbatim		None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) a. The occurrence of ANY of the following hazardous events:	(1) a. The occurrence of ANY of the Table S-3 hazardous events:	Difference	Global Comment #12 & 13	None
	• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager	Table S-3 Hazardous Events • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director	Difference	Global Comment #8 & 9	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SA9 (cont.)	**AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.	**AND** b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode. **AND**	Deviation	Adopted the revised EAL structure and wording provided in approved EAL FAQ 2016-02.	V17
	OR 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	2. **EITHER** of the following: • Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or • The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.	Deviation Difference	Adopted the revised EAL wording provided in approved EAL FAQ 2016-02 Added the following clarification to the Basis from EALFAQ 2018-04: An event affecting a single-train SAFETY SYSTEM (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under SA8 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train SAFETY SYSTEM, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement. Intent and meaning of the EALs are not altered.	V17 V18

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SS1	Recognition Category: SS1	SS1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.	Loss of **ALL** offsite and **ALL** onsite AC power to essential buses for 15 minutes or longer.	Difference	Global Comment #13, 15	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses for 15 minutes or longer.	Difference	Global Comment #9, 12 & 13 Intent and meaning of the EALs are not altered.	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SS5	Recognition Category: SS5	SS6	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Inability to shutdown the reactor causing a challenge to (core cooling [PWR] / RPV water level [BWR]) or RCS heat removal.	Inability to shutdown the reactor causing a challenge to RPV water level or RCS heat removal.	Difference	Global Comment #4	None
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1, 2	Difference	Global Comment #10 DAEC can be up to 12% power in STARTUP Mode, so Mode 2 applicability added	V33
	(1) a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor.	(1) a. An automatic or manual scram did not shutdown the reactor.	Difference	Global Comment #4, 9 & 12	None
	AND b. All manual actions to shutdown the reactor have been unsuccessful.	**AND** b. All manual actions to shutdown the reactor have been unsuccessful.	Verbatim		None
	AND c. EITHER of the following conditions exist: • (Site-specific indication of an inability to adequately remove heat from the core) • (Site-specific indication of an inability to adequately remove heat from the RCS)	**AND** c. **EITHER** of the following conditions exist: • RPV level cannot be restored and maintained above -25 inches. **OR** • HCL (Graph 4 of EOP 2) exceeded.	Difference	Global Comment #9	V34 V27
				Intent and meaning of the EALs are not altered.	

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SS8	Recognition Category: SS8	SS2	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Loss of all Vital DC power for 15 minutes or longer.	Loss of **ALL** Vital DC power for 15 minutes or longer.	Difference	Global Comment #13	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) Indicated voltage is less than (site-specific bus voltage value) on **ALL** (site-specific Vital DC busses) for 15 minutes or longer.	(1) Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.	Difference	Global Comment #9 & 12 Intent and meaning of the EALs are not altered.	V12

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SG1	Recognition Category: SG1	SG1	Verbatim	Global Comment #11, 14	None
	Initiating Condition: Prolonged loss of all offsite and all onsite AC power to emergency buses.	Prolonged loss of **ALL** offsite and **ALL** onsite AC power to essential buses.	Difference	Global Comment #13, 15	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) a. Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses).	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses	Difference	Global Comment #9 & 13	None
	AND b. EITHER of the following: • Restoration of at least one AC emergency bus in less than (site-specific hours) is not likely.	**AND** b. **EITHER** of the following: • Restoration of at least one AC essential bus in less than 4 hours is not likely. **OR**	Difference	Global Comment #9 & 13	
	• (Site-specific indication of an inability to adequately remove heat from the core)	• RPV level cannot be restored and maintained above -25 inches.	Difference	Global Comment #9 Intent and meaning of the EALs are not altered.	V34

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
SG8	Recognition Category: SG8	SG2	Difference	Global Comment #11, 14 Renumbered to align with other similar ICs	None
	Initiating Condition: Loss of all AC and Vital DC power sources for 15 minutes or longer.	Loss of **ALL** AC and Vital DC power sources for 15 minutes or longer.	Verbatim	Global Comment #13	None
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3	Difference	Global Comment #10	None
	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses) for 15 minutes or longer. **AND** b. Indicated voltage is less than (site-specific bus voltage value) on ALL (site-specific Vital DC busses) for 15 minutes or longer.	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to 1A3 and 1A4 buses for 15 minutes or longer. **AND** b. Indicated voltage is less than 105 VDC on **BOTH** Div 1 and Div 2 125 VDC buses for 15 minutes or longer.	Difference Difference	Global Comment #9, 12, 13 Global Comment #9 & 13 Intent and meaning of the EALs are not altered.	None V12

APPENDIX A – ACRONYMS AND ABBREVIATIONS

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS	AC.......Alternating Current	AC.......Alternating Current	Verbatim		N/A
	AOP......Abnormal Operating Procedure	AOP......Abnormal Operating Procedure	Verbatim		N/A
	APRM...Average Power Range Meter		Difference	Not used	N/A
	ATWS...Anticipated Transient Without Scram	ATWS...Anticipated Transient Without Scram	Verbatim		N/A
	B&W....Babcock and Wilcox		Difference	Not used	N/A
	BIIT......Boron Injection Initiating Temperature		Difference	Not used	N/A
	BWR....Boiling Water Reactor	BWR....Boiling Water Reactor	Verbatim		N/A
	CDE......Committed Dose Equivalent	CDE......Committed Dose Equivalent	Verbatim		N/A
	CFR......Code of Federal Regulations	CFR......Code of Federal Regulations	Verbatim		N/A
	CTMT/CNMT...Containment		Difference	Not used	N/A
	CSF......Critical Safety Function		Difference	Not used	N/A
	CSFST...Critical Safety Function Status Tree		Difference	Not used	N/A
	DBA......Design Basis Accident		Difference	Not used	N/A
	DC........Direct Current	DC........Direct Current	Verbatim		N/A
	EAL.......Emergency Action Level	EAL.......Emergency Action Level	Verbatim		N/A
	ECCS....Emergency Core Cooling System	ECCS....Emergency Core Cooling System	Verbatim		N/A
	ECL.......Emergency Classification Level	ECL.......Emergency Classification Level	Verbatim		N/A
	EOF......Emergency Operations Facility	EOF......Emergency Operations Facility	Verbatim		N/A
	EOP......Emergency Operating Procedure	EOP......Emergency Operating Procedure	Verbatim		N/A
	EPA......Environmental Protection Agency	EPA......Environmental Protection Agency	Verbatim		N/A
	EPG.....Emergency Procedure Guideline	EPG.....Emergency Procedure Guideline	Verbatim		N/A
	EPIP.....Emergency Planning Implementing Procedure		Difference	Not used	N/A

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS (cont.)	EPR......Evolutionary Power Reactor		Difference	Not used	N/A
	EPRI.....Electric Power Research Institute		Difference	Not used	N/A
	ERG.....Emergency Response Guideline		Difference	Not used	N/A
	FEMA...Federal Emergency Management Agency	FEMA...Federal Emergency Management Agency	Verbatim		N/A
	FSAR....Final Safety Analysis Report		Difference	Not used	N/A
	GE........General Emergency	GE........General Emergency	Verbatim		N/A
	HCTL....Heat Capacity Temperature Limit	HCL....Heat Capacity Limit	Difference	Updated to reflect DAEC EOPs	N/A
	HPCI.....High Pressure Coolant Injection	HPCI.....High Pressure Coolant Injection	Verbatim		N/A
	HSI........Human System Interface		Difference	Not used	N/A
	IC..........Initiating Condition	IC..........Initiating Condition	Verbatim		N/A
	ID.........Inside Diameter	ID.........Inside Diameter	Verbatim		N/A
	IPEEE...Individual Plant Examination of External Events (Generic Letter 88-20)		Difference	Not used	N/A
	ISFSI....Independent Spent Fuel Storage Installation	ISFSI....Independent Spent Fuel Storage Installation	Verbatim		N/A
	Keff.....Effective Neutron Multiplication Factor	Keff.....Effective Neutron Multiplication Factor	Verbatim		N/A
	LCO.....Limited Condition of Operation	LCO.....Limited Condition of Operation	Verbatim		N/A
	LOCA...Loss of Coolant Accident	LOCA...Loss of Coolant Accident	Verbatim		N/A
	MCR....Main Control Room		Difference	Not used	N/A
	MSIV...Main Steam Isolation Valve		Difference	Not used	N/A
	MSL.....Main Stem Line		Difference	Not used	N/A
	mR, mRem, mrem, mREM....milli-Roentgen Equivalent Man	mR, mRem, mrem, mREM....milli-Roentgen Equivalent Man	Verbatim		N/A
	MW.....Megawatt	MW.....Megawatt	Verbatim		N/A
	NEI.......Nuclear Energy Institute	NEI.......Nuclear Energy Institute	Verbatim		N/A
	NPP......Nuclear Power Plant		Difference	Not used	N/A

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS (cont.)	NRC.....Nuclear Regulatory Agency	NRC.....Nuclear Regulatory Agency	Verbatim		N/A
	NSSS....Nuclear Steam Supply System		Difference	Not used	N/A
	NORAD...North American Aerospace Defense Command	NORAD...North American Aerospace Defense Command			N/A
	(NO)UE...(Notification of) Unusual Event	NOUE...Notification of Unusual Event	Difference	DAEC uses full NOUE terminology	N/A
	NUMARC....Nuclear Management and Resources Council	NUMARC....Nuclear Management and Resources Council	Verbatim		N/A
	OBE.....Operating Basis Earthquake	OBE.....Operating Basis Earthquake	Verbatim		N/A
	OCA.....Owner Controlled Area	OCA.....Owner Controlled Area	Verbatim		N/A
	ODCM/ODAM....Offsite Dose Calculation (Assessment) Manual	ODAM...Offsite Dose Assessment Manual	Difference	DAEC uses ODAM	N/A
	ORO.....Offsite Response Organization		Difference	Not used	N/A
	PA.........Protected Area	PA.........Protected Area	Verbatim		N/A
	PACS....Priority Information and Control System		Difference	Not used	N/A
	PAG......Protective Action Guideline	PAG......Protective Action Guideline	Verbatim		N/A
	PICS.....Process Information and Control System		Difference	Not used	N/A
	PRA/PSA...Probabilistic Risk Assessment/Probabilistic Safety Assessment	PRA/PSA...Probabilistic Risk Assessment/Probabilistic Safety Assessment	Verbatim		N/A
	PWR....Pressurized Water Reactor	PWR....Pressurized Water Reactor	Verbatim		N/A
	PS.........Protection System		Difference	Not used	N/A
	PSIG....Pounds per Square Inch	PSIG....Pounds per Square Inch	Verbatim		N/A
	R..........Roentgen	R..........Roentgen	Verbatim		N/A
	RCC....Reactor Control Console		Difference	Not used	N/A
	RCIC...Reactor Core Isolation Cooling	RCIC...Reactor Core Isolation Cooling	Verbatim		N/A

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX A – ACRONYMS AND ABBREVIATIONS (cont.)	RCS.....Reactor Coolant System	RCS.....Reactor Coolant System	Verbatim		N/A
	Rem, rem, REM...Roentgen Equivalent Man	Rem, rem, REM...Roentgen Equivalent Man	Verbatim		N/A
	RETS....Radiological Effluent Technical Specifications		Difference	Not used	N/A
	RPS......Reactor Protection System	RPS......Reactor Protection System	Verbatim		N/A
	RPV......Reactor Pressure Vessel	RPV......Reactor Pressure Vessel	Verbatim		N/A
	RVLIS...Reactor Vessel Level Instrumentation System		Difference	Not used	N/A
	RWCU...Reactor Water Cleanup	RWCU...Reactor Water Cleanup	Verbatim		N/A
	SAR.......Safety Analysis Report		Difference	Not used	N/A
	SAS........Safety Automation System		Difference	Not used	N/A
	SBO.......Station Blackout		Difference	Not used	N/A
	SCBA.....Self-Contained Breathing Apparatus	SCBA.....Self-Contained Breathing Apparatus	Verbatim		N/A
	SG.........Steam Generator		Difference	Not used	N/A
	SI..........Safety Injection		Difference	Not used	N/A
	SICS......Safety Information Control System		Difference	Not used	N/A
	SPDS.....Safety Parameter Display System	SPDS.....Safety Parameter Display System	Verbatim		N/A
	SRO.......Senior Reactor Operator		Difference	Not used	N/A
	TEDE.....Total Effective Dose Equivalent	TEDE.....Total Effective Dose Equivalent	Verbatim		N/A
	TOAF.....Top of Active Fuel	TAF.....Top of Active Fuel	Difference	Updated to reflect DAEC EOPs	N/A
	TSC........Technical Support System	TSC........Technical Support System	Verbatim		N/A
	-	UFSAR....Final Safety Analysis Report	Difference	Used in Section 3.1	N/A
	WOG.....Westinghouse Owners Group		Difference	Not used	N/A

APPENDIX B - DEFINITIONS

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	Alert: Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.	Alert: Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.	Verbatim		None
	General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.	General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.	Verbatim		None
	Notification of Unusual Event: Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	Unusual Event: Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.	Difference	See Global Comment #3	None

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	Site Area Emergency: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.	Site Area Emergency: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.	Verbatim		None
	Emergency Action Level (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.	Emergency Action Level (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.	Verbatim		None
	Emergency Classification Level (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: • Notification of Unusual Event (NOUE) • Alert • Site Area Emergency (SAE) • General Emergency (GE)	Emergency Classification Level (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: • Notification of Unusual Event (NOUE) • Alert • Site Area Emergency (SAE) • General Emergency (GE)	Verbatim		None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	Fission Product Barrier Threshold: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.	Fission Product Barrier Threshold: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.	Verbatim		None
	Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.	Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.	Verbatim		None
	CONFINEMENT BOUNDARY: (Insert a site-specific definition for this term.) **Developer Note** – The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.	CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. This corresponds to the pressure boundary for the Dry Shielded Canister (DSC) shell (including the inner bottom cover plate) base metal and associated confinement boundary welds.	Difference	Removed developer notes and added site-specific language.	None
	CONTAINMENT CLOSURE: (Insert a site-specific definition for this term.) **Developer Note** – The procedurally defined conditions or actions taken to secure containment (primary or secondary for BWR) and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.	CONTAINMENT CLOSURE: Site specific procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. For DAEC, this is considered to be Secondary Containment as required by Technical Specifications.	Difference	Removed developer notes and added existing definition from present EALs.	None
		DESIGN BASIS EARTHQUAKE (DBE): A DBE is vibratory ground motion for which certain (generally, safety-related) structures, systems, and components must be designed to remain functional.	Difference	Added term used in HU2 versus use of footnotes	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.	EXPLOSION: A rapid, violent, and catastrophic failure of a piece of equipment due to combustion, chemical reaction, or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.	Verbatim		None
	FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized. Developer Note – This term is applicable to PWRs only.		Difference	Term not used for BWRs	None
	FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.	FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.	Verbatim		None
	HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.	HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.	Verbatim		None

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
	HOSTILE ACTION: An act toward a NPP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).	HOSTILE ACTION: An act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power plant. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).	Difference	Spelled out 'NPP' in 2 places	None
	HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.	HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.	Verbatim		None
	IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.	IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.	Verbatim		None
	INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.	INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.	Verbatim		None

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	NORMAL LEVELS: As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.		Difference	Term not used in this EAL scheme	None
		OPERATING BASIS EARTHQUAKE (OBE): An OBE is vibratory ground motion for which those features of a nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional.	Difference	Added term used in HU2 versus use of footnotes	None
	OWNER CONTROLLED AREA: (Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the site property owned by, or otherwise under the control of, the licensee. In some cases, it may be appropriate for a licensee to define a smaller area with a perimeter closer to the plant Protected Area perimeter (e.g., a site with a large OCA where some portions of the boundary may be a significant distance from the Protected Area). In these cases, developers should consider using the boundary defined by the Restricted or Secured Owner Controlled Area (ROCA/SOCA). The area and boundary selected for scheme use must be consistent with the description of the same area and boundary contained in the Security Plan.	OWNER CONTROLLED AREA: The site property owned by or otherwise under the control of the licensee.	Difference	Definition from developer notes used. Developer Notes deleted.	None
	PROJECTILE: An object directed toward a NPP that could cause concern for its continued operability, reliability, or personnel safety.	PROJECTILE: An object directed toward a nuclear power plant that could cause concern for its continued operability, reliability, or personnel safety.	Difference	Spelled out 'NPP'	None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
APPENDIX B - DEFINITIONS	PROTECTED AREA: (Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the area under continuous access monitoring and control, and armed protection as described in the site Security Plan.	PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.	Difference	Definition from developer notes used. Developer Notes deleted.	None
	REFUELING PATHWAY: (Insert a site-specific definition for this term.) **Developer Note** – This description should include all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.	REFUELING PATHWAY: The reactor refueling cavity, spent fuel pool, and fuel transfer canal.	Difference	DAEC-specific definition supplied. Developer Notes deleted.	None
	RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection. **Developer Note** – This term is applicable to PWRs only.		Difference	Not used	None
	SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related. **Developer Note** – This term may be modified to include the attributes of "safety-related" in accordance with 10 CFR 50.2 or other site-specific terminology, if desired.	SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These systems are classified as safety-related.	Difference	Removed developer notes and clarified last sentence.	None
	SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety	SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety	Verbatim		None

DAEC DEVIATIONS AND DIFFERENCES MATRIX

IC	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
	of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.	of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.			
		SITE BOUNDARY: That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the Company. UFSAR Figure 1.2-1 identifies the DAEC SITE BOUNDARY.	Difference	Defined term from ODCM needed for several EALs	None
	UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.	UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.	Verbatim		None
APPENDIX B - DEFINITIONS	UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.	UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.	Verbatim		N/A
	VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.	VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.	Deviation	Updated to reflect wording and guidance of approved EAL FAQ 2016-02. The updated wording clarifies damage assessment meriting an ALERT declaration as used in ICs using this definition (CA6 and SA9).	V17

APPENDIX C - Permanently Defueled ICs/EALs

DAEC DEVIATIONS AND DIFFERENCES MATRIX

Section	NEI 99-01 Rev. 6	DAEC	Change	Justification	Validation #
Appendix C – Permanently Defueled ICs/EALs	Appendix C - Permanently Defueled ICs/EALs	Not used at DAEC	Difference	Not applicable to DAEC	None

ATTACHMENT 4

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATING TO

LICENSE AMENDMENT REQUEST TSCR-166

UPDATED SUPPORTING TECHNICAL INFORMATION

248 pages follow

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[(a)] or Startup/Hot Standby	NA
3	Hot Shutdown[(a)]	Shutdown	$>$ 212
4	Cold Shutdown[(a)]	Shutdown	$\leq$ 212
5	Refueling[(b)]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Development of EAL Threshold values from NEE-323-CALC-003

Calculated values are provided in Calc-003 as shown below.

Values for the RU1 Gaseous EALs were determined and are shown below.

Table 1 – Gaseous Effluent Setpoints

Location	Detector	RU1 Threshold (µCi/cc)
Offgas Stack	Kaman 10	1.97E-01
Turbine Building Vent	Kaman 2	7.74E-04
Reactor Building Vent	Kaman 4	6.00E-04
Reactor Building Vent	Kaman 6	9.60E-04
Reactor Building Vent	Kaman 8	9.60E-04
LLRPSF Building Vent	Kaman 12	1.19E-03

Values for the Liquid Effluent RU1 EALs were determined and are shown below.

Table 2 – Liquid Effluent Setpoints

Location	Equipment ID	RU1 Unusual Event Level (cps)
GSW	RE-4767	1.53E+03
RHRSW/ESW	RE-1997	8.42E+02
RHRSW Dilution Line*	RE-4268	1.06E+03

The values are rounded for ease of operator use and to provide a step-wise progression through the emergency classification levels. The resulting values used in the DAEC RU1.1 EAL are shown in the NOUE column below:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
Gaseous	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
Liquid	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
Liquid	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

ENERCON Excellence—Every project. Every day.	CALCULATION COVER SHEET	CALC NO. NEE-323-CALC-003 REV. 00 PAGE NO. 1 of 9

Title:	Documentation of the RU1 Emergency Action Levels	Client: Duane Arnold Energy Center Project Identifier: NEE-323

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions, including preliminary information, that require confirmation? (If **YES**, identify the assumptions.)	☒	☐
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, identify the design verified calculation.) **Design Verified Calculation No.** ______	☐	☒
3	Does this calculation supersede an existing Calculation? (If **YES**, identify the design verified calculation.) **Superseded Calculation No.** ______	☐	☒

Scope of Revision:
Initial Issue

Revision Impact on Results:
Initial Issue

Study Calculation ☐ **Final Calculation** ☒

Safety-Related ☐ **Non-Safety-Related** ☒

(Print Name and Sign)

Originator: Jay Bhatt	**Date:** **12/12/17**
Design Verifier[1] (Reviewer if NSR): Ryan Skaggs	**Date:** **12/12/17**
Approver: Aaron Holloway	**Date:** **12/12/17**

Note 1: For non-safety-related calculation, design verification can be substituted by review.

ENERCON Excellence—Every project. Every day.	CALCULATION REVISION STATUS SHEET	CALC NO. NEE-323-CALC-003
		REV. 00

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
00	12/12/17	Initial Issue

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
All	00		

APPENDIX/ATTACHMENT REVISION STATUS

APPENDIX NO.	NO. OF PAGES	REVISION NO.	ATTACHMENT NO.	NO. OF PAGES	REVISION NO.
			1	4	00
			2	18	00
			3	9	00

1.0 Purpose and Scope

The Duane Arnold Energy Center site is implementing the guidance of Revision 6 to the Document NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," which is the industry-developed methodology for emergency classification for the current operating fleet. Changes to the definitions of the condition for entry into the Emergency Action Level (EAL) RU1 result in the development of a new entry threshold value for this EAL.

This calculation provides calculated threshold values for the following EALs (from NEI 99-01, Rev. 6). Note that NEI 99-01 designates abnormal radiological conditions as "AU," NEE has adopted the "RU" designation permitted under the guidance.

(1) Reading on ANY effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.

(2) Reading on ANY effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

This calculation uses the latest radiation monitor setpoints to determine the resultant EAL thresholds.

2.0 Summary of Results and Conclusions

Values for the RU1 Gaseous EALs were determined and are shown below.

Table 1 – Gaseous Effluent Setpoints

Location	Detector	RU1 Threshold (µCi/cc)
Offgas Stack	Kaman 10	1.97E-01
Turbine Building Vent	Kaman 2	7.74E-04
Reactor Building Vent	Kaman 4	6.00E-04
Reactor Building Vent	Kaman 6	9.60E-04
Reactor Building Vent	Kaman 8	9.60E-04
LLRPSF Building Vent	Kaman 12	1.19E-03

Values for the Liquid Effluent RU1 EALs were determined and are shown below.

Table 2 – Liquid Effluent Setpoints

Location	Equipment ID	RU1 Unusual Event Level (cps)
GSW	RE-4767	1.53E+03
RHRSW/ESW	RE-1997	8.42E+02
RHRSW Dilution Line*	RE-4268	1.06E+03

*RE-4268 was previously known as the RHRSW Rupture Disk

3.0 References

3.1 NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors." November 2012.

3.2 DAEC Offsite Dose Assessment Manual (ODAM), Rev. 37.

3.3 Plant Chemistry Procedure PCP 8.3, Alarm Setpoints and Background Determination for KAMAN Normal Range Monitors.

3.4 Plant Chemistry Procedure PCP 8.7, Alarm Setpoints For Liquid Rad Monitors.

3.5 Technical Specifications, Section 5.5.4, Radioactive Effluent Controls Program.

3.6 DAEC Emergency Plan, Section 'I', Rev. 27

4.0 Assumptions

It is assumed that the current setpoint for the Kaman 4 monitor is 3.00E-04 µCi/cc. The latest setpoint determination received is from 3/4/2016 which exceeds the 18 month frequency specified by PCP 8.3.

5.0 Design Inputs

5.1 The setpoint determinations from Attachment 2 and Attachment 3, represent the latest responses at the associated gaseous and liquid effluent monitors. While the three most recent surveillances for each monitor are included for information, only the latest setpoint is used to determine the EAL threshold. It should be noted that the "RM" equipment designations are equivalent to the "RE" equipment IDs.

5.2 The gaseous effluent equipment ID number, monitor common name and range are taken from DAEC Emergency Plan Section "I" and ODAM Figure 3-1, and are presented in Table 3.

Table 3 – Gaseous Effluent Design Inputs

Location	Monitor Common Name	Equipment ID	Monitor Range (µCi/cc)
Offgas Stack	KAMAN 9/10	RE-4176, RE-4175	1E-07 - 1E+05
Turbine Building Vent	KAMAN 1/2	RE-5945 / RE-5946	1E-07 - 1E+05
Reactor Building Vent	KAMAN 3/4 KAMAN 5/6 KAMAN 7/8	RE-7645, RE-7644 RE-7647, RE-7646 RE-7649, RE-7648	1E-07 - 1E+05
LLRPSF Building Vent	KAMAN 12	RE-8801	1E-07 - 3E-01

5.3 The liquid effluent equipment ID number, and range are taken from ODAM Table I-2, and are presented in Table 4.

Table 4 – Liquid Effluent Design Inputs

Location	Equipment ID	Monitor Range (cps)
GSW	RE-4767	1E-01 - 1E+06
RHRSW/ESW	RE-1997	1E-01 - 1E+06
RHRSW Dilution Line	RE-4268	1E-01 - 1E+06

6.0 Methodology

The alarm setpoint of a radioactive noble gas effluent monitor is calculated on the basis of whole body dose equivalent rate offsite of 500 mrem/yr per the ODAM. The alarm setpoint for liquid radwaste effluent line provides automatic isolation when 10 times the water effluent concentration listed in 10 CFR 20 Appendix B, Table 2, is being exceeded in the unrestricted area per the ODAM. These setpoints are in accordance with Technical Specifications limits specified in 5.5.4b and 5.5.4g. This calculation considers historical setpoint determination for gaseous release (PCP 8.3) and liquid effluent (PCP 8.7). The latest three setpoints for each monitor were reviewed. Due to the high variance for some of the monitors, the latest alarm setpoint is used to determine the EAL thresholds.

7.0 Calculation

7.1 Gaseous Setpoints

Plant Chemistry Procedure PCP 8.3 is used by Chemistry Technicians to calculate setpoints for building vent KAMAN monitors at least once every 18 months. The three latest setpoint determinations for each location are shown in Attachment 2 for information. It should be noted that where the original PCP 8.3 setpoint calculation sheet is unavailable, the value is taken from the associated monitor calibration procedure.

Thresholds corresponding to the latest setpoints are calculated and presented here. For example the latest PCP 8.3 setpoint for Offgas stack is 9.84E-02 µCi/cc. This value is doubled to 1.97E-01 µCi/cc to correspond to the RU1 threshold. The remaining threshold values are shown in Table 5.

Table 5 – Gaseous Effluent Setpoints and Thresholds

Location	Detector	Latest PCP 8.3 Setpoint (µCi/cc)	RU1 Threshold (µCi/cc)
Offgas Stack	Kaman 10	9.84E-02	1.97E-01
Turbine Building Vent	Kaman 2	3.87E-04	7.74E-04
Reactor Building Vent	Kaman 4	3.00E-04	6.00E-04
Reactor Building Vent	Kaman 6	4.80E-04	9.60E-04
Reactor Building Vent	Kaman 8	4.80E-04	9.60E-04
LLRPSF Building Vent	Kaman 12	5.95E-04	1.19E-03

7.2 Liquid Setpoints

As a result of variability in the isotopic mix of reactor water, background radiation levels and detector efficiencies, the calculated liquid effluent setpoints will fluctuate over time.

Chemistry Technicians perform effluent liquid radiation monitor setpoint calculations at least once per 18 months with guidance provided by Plant

Chemistry Procedure PCP 8.7. The three latest setpoint determinations for each location are shown in Attachment 3. It should be noted that where the original PCP 8.7 setpoint calculation sheet is unavailable, the value is taken from the associated monitor calibration procedure.

Thresholds corresponding to the latest setpoints are calculated and presented here. For example the latest PCP 8.7 setpoint for the RHRSW Dilution Line is 421 cps. This value is doubled to 842 cps to correspond to the RU1 threshold. The remaining threshold values are shown in Table 6.

Table 6 – Liquid Effluent Setpoints and Thresholds

Location	Latest PCP 8.7 Setpoint (cps)	RU1 Threshold (cps)
GSW	7.65E+02	1.53E+03
RHRSW/ESW	4.21E+02	8.42E+02
RHRSW Dilution Line	5.30E+02	1.06E+03

8.0 Computer Software

None.

9.0 Impact Assessment

This calculation is based on “realistic” conditions for the purpose of declaring EALs, rather than typical conservative “bounding” type design basis analyses. The calculation documents the order of magnitude setpoints to assist Operations and Emergency Response personnel in determining an unusual event in accordance with NEI 99-01 Rev. 6.

ENERCON Excellence—Every project. Every day.	Attachment 1 CALCULATION PREPARATION CHECKLIST	CALC NO. NEE-323-CALC-003 REV. 00

CHECKLIST ITEMS[1]	YES	NO	N/A
GENERAL REQUIREMENTS			
1. If the calculation is being performed to a client procedure, is the procedure being used the latest revision?	☐	☐	☒
The calculation is being prepared to ENERCON's procedures.			
2. Are the proper forms being used and are they the latest revision?	☒	☐	☐
3. Have the appropriate client review forms/checklists been completed?	☐	☐	☒
The calculation is being prepared to ENERCON's procedures.			
4. Are all pages properly identified with a calculation number, calculation revision and page number consistent with the requirements of the client's procedure?	☒	☐	☐
5. Is all information legible and reproducible?	☒	☐	☐
6. Is the calculation presented in a logical and orderly manner?	☒	☐	☐
7. Is there an existing calculation that should be revised or voided?	☐	☒	☐
This is a new calculation to support implementing NEI 99-01 Rev. 6			
8. Is it possible to alter an existing calculation instead of preparing a new calculation for this situation?	☐	☒	☐
9. If an existing calculation is being used for design inputs, are the key design inputs, assumptions and engineering judgments used in that calculation valid and do they apply to the calculation revision being performed.	☒	☐	☐
10. Is the format of the calculation consistent with applicable procedures and expectations?	☒	☐	☐
11. Were design input/output documents properly updated to reference this calculation?	☐	☐	☒
12. Can the calculation logic, methodology and presentation be properly understood without referring back to the originator for clarification?	☒	☐	☐
OBJECTIVE AND SCOPE			
13. Does the calculation provide a clear concise statement of the problem and objective of the calculation?	☒	☐	☐
14. Does the calculation provide a clear statement of quality classification?	☒	☐	☐
15. Is the reason for performing and the end use of the calculation understood?	☒	☐	☐
16. Does the calculation provide the basis for information found in the plant's license basis?	☒	☐	☐
17. If so, is this documented in the calculation?	☐	☐	☒
18. Does the calculation provide the basis for information found in the plant's design basis documentation?	☐	☒	☐

CHECKLIST ITEMS[1]		YES	NO	N/A
19.	If so, is this documented in the calculation?	☐	☐	☒
20.	Does the calculation otherwise support information found in the plant's design basis documentation?	☐	☒	☐
21.	If so, is this documented in the calculation?	☐	☐	☒
22.	Has the appropriate design or license basis documentation been revised, or has the change notice or change request documents being prepared for submittal?	☐	☐	☒
DESIGN INPUTS				
23.	Are design inputs clearly identified?	☒	☐	☐
24.	Are design inputs retrievable or have they been added as attachments?	☒	☐	☐
25.	If Attachments are used as design inputs or assumptions are the Attachments traceable and verifiable?	☒	☐	☐
26.	Are design inputs clearly distinguished from assumptions?	☒	☐	☐
27.	Does the calculation rely on Attachments for design inputs or assumptions? If yes, are the attachments properly referenced in the calculation?	☒	☐	☐
28.	Are input sources (including industry codes and standards) appropriately selected and are they consistent with the quality classification and objective of the calculation?	☒	☐	☐
29.	Are input sources (including industry codes and standards) consistent with the plant's design and license basis?	☒	☐	☐
30.	If applicable, do design inputs adequately address actual plant conditions?	☒	☐	☐
31.	Are input values reasonable and correctly applied?	☒	☐	☐
32.	Are design input sources approved?	☒	☐	☐
33.	Does the calculation reference the latest revision of the design input source?	☒	☐	☐
34.	Were all applicable plant operating modes considered?	☒	☐	☐
ASSUMPTIONS				
35.	Are assumptions reasonable/appropriate to the objective?	☐	☐	☒
36.	Is adequate justification/basis for all assumptions provided?	☐	☐	☒
37.	Are any engineering judgments used?	☐	☒	☐
38.	Are engineering judgments clearly identified as such?	☐	☐	☒
39.	If engineering judgments are utilized as design inputs, are they reasonable and can they be quantified or substantiated by reference to site or industry standards, engineering principles, physical laws or other appropriate criteria?	☐	☐	☒

CHECKLIST ITEMS[1]		YES	NO	N/A
METHODOLOGY				
40.	Is the methodology used in the calculation described or implied in the plant's licensing basis?	☐	☒	☐
41.	If the methodology used differs from that described in the plant's licensing basis, has the appropriate license document change notice been initiated?	☐	☐	☒
42.	Is the methodology used consistent with the stated objective?	☒	☐	☐
43.	Is the methodology used appropriate when considering the quality classification of the calculation and intended use of the results?	☒	☐	☐
BODY OF CALCULATION				
44.	Are equations used in the calculation consistent with recognized engineering practice and the plant's design and license basis?	☒	☐	☐
45.	Is there reasonable justification provided for the use of equations not in common use?	☐	☐	☒
46.	Are the mathematical operations performed properly and documented in a logical fashion?	☒	☐	☐
47.	Is the math performed correctly?	☒	☐	☐
48.	Have adjustment factors, uncertainties and empirical correlations used in the analysis been correctly applied?	☒	☐	☐
49.	Has proper consideration been given to results that may be overly sensitive to very small changes in input?	☒	☐	☐
SOFTWARE/COMPUTER CODES				
50.	Are computer codes or software languages used in the preparation of the calculation?	☐	☒	☐
51.	Have the requirements of CSP 3.09 for use of computer codes or software languages, including verification of accuracy and applicability been met?	☐	☐	☒
52.	Are the codes properly identified along with source vendor, organization, and revision level?	☐	☐	☒
53.	Is the computer code applicable for the analysis being performed?	☐	☐	☒
54.	If applicable, does the computer model adequately consider actual plant conditions?	☐	☐	☒
55.	Are the inputs to the computer code clearly identified and consistent with the inputs and assumptions documented in the calculation?	☐	☐	☒
56.	Is the computer output clearly identified?	☐	☐	☒
57.	Does the computer output clearly identify the appropriate units?	☐	☐	☒

CHECKLIST ITEMS[1]		YES	NO	N/A
58.	Are the computer outputs reasonable when compared to the inputs and what was expected?	☐	☐	☒
59.	Was the computer output reviewed for ERROR or WARNING messages that could invalidate the results?	☐	☐	☒
RESULTS AND CONCLUSIONS				
60.	Is adequate acceptance criteria specified?	☒	☐	☐
61.	Are the stated acceptance criteria consistent with the purpose of the calculation, and intended use?	☒	☐	☐
62.	Are the stated acceptance criteria consistent with the plant's design basis, applicable licensing commitments and industry codes, and standards?	☒	☐	☐
63.	Do the calculation results and conclusions meet the stated acceptance criteria?	☒	☐	☐
64.	Are the results represented in the proper units with an appropriate tolerance, if applicable?	☒	☐	☐
65.	Are the calculation results and conclusions reasonable when considered against the stated inputs and objectives?	☒	☐	☐
66.	Is sufficient conservatism applied to the outputs and conclusions?	☒	☐	☐
67.	Do the calculation results and conclusions affect any other calculations?	☐	☒	☐
68.	If so, have the affected calculations been revised?	☐	☐	☒
69.	Does the calculation contain any conceptual, unconfirmed or open assumptions requiring later confirmation?	☐	☒	☐
70.	If so, are they properly identified?	☐	☐	☒
DESIGN REVIEW				
71.	Have alternate calculation methods been used to verify calculation results?	☐	☐	☒
No, a Design Review was performed.				

Note:

1. Where required, provide clarification/justification for answers to the questions in the space provided below each question. An explanation is required for any questions answered as "No' or "N/A".

Originator: Jay Bhatt 12/12/17

Print Name and Sign Date

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ATTACHMENT 2

KAMAN OFFGAS STACK GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. K10 CHARMAR 2. Sample No. 17-5560
3. Sample Date 8-30-17 4. Sample Time 0854 5. MWT 1906
6. Count Date 8-30-17 7. Count Time 0915
8. Monitor Reading (μCi/cc) ~~7.73 e-7~~ 7.70e-7 9. Process Flow Rate (CFM) ~~5.17e3~~ 10,000
10. Sample Volume (mL) 4.50e4 cc

Flow meter ID # L729 Cal Due Date 10-6-17

Isotope	11 μCi/mL k_i	12 Dose Factor Stack * mrem sec / yr - μCi	13 k_i x DFSi
Xe 133		4.09E-5	
Kr 85m		1.81E-4	
Kr 88		1.91E-3	
Xe 135		2.84E-4	
Kr 87		6.97E-4	
Xe 138		1.08E-3	
Xe 135m		3.39E-4	
Xe 133m		3.61E-5	
Ar 41		1.32E-3	
N 13	1.41e-7	1.08E-3 **	1.52e-10

14. Bkg = instrument background
Bkg = 2.15e-7 μCi/cc

* These dose factors are from ODAM: stack release at a distance of 1260 meters NNW of DAEC

** Arbitrarily set equal to Xe-138

15. $\sum k_i$ = 1.41e-7 15a. $\sum_i (k_i \bullet DFS_i)$ 1.52e-10

15b. $\frac{\sum k_i}{\sum (k_i \bullet DFS_i)} = \frac{(\#15)}{(\#15a)} = \frac{1.41e^{-7}}{1.52e^{-10}} = 9.28e^{2}$

16. Limit = $L = \frac{1.06}{F} \times \frac{\sum k_i}{\sum (k_i \bullet DFS_i)} = \frac{1.06}{(\#9)} \times$ (The Less of **#15b** OR 3436)

Limit = L = $\frac{1.06}{(10{,}000)}$ (928) = 9.84e-2

17. Hi Hi ALARM = A x (**#16**) = (1.0) (9.84e-2) = 9.84e-2 μCi/cc

The radioactive gas flow corresponding to the HI Hi setpoint:

Performed by: [signature] Date: 8-30-17

Independent Verification by: [signature] Date: 8-30-17

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ATTACHMENT 2

KAMAN OFFGAS STACK GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. K10 Char Mar
2. Sample No. 16-1390
3. Sample Date 2-26-16
4. Sample Time 1520
5. MWT 1911
6. Count Date 2-26-16
7. Count Time 1558
8. Monitor Reading (μCi/cc) 6.61e-7
9. Process Flow Rate (CFM) 10,000
10. Sample Volume (mL) 45,000 mL

Flow meter ID # L729 Cal Due Date 10-6-17

	11	12	13
Isotope	μCi/mL k_i	Dose Factor Stack * $\frac{\text{mrem sec}}{\text{yr - μCi}}$	k_i x DFSi
Xe 133	4.27e-9	4.09E-5	1.75e-13
Kr 85m	2.02e-9	1.81E-4	3.66e-13
Kr 88		1.91E-3	
Xe 135	3.35e-9	2.84E-4	9.51e-13
Kr 87	2.93e-9	6.97E-4	2.04e-12
Xe 138		1.08E-3	
Xe 135m		3.39E-4	
Xe 133m		3.61E-5	
Ar 41	3.07e-9	1.32E-3	4.05e-12
N 13	9.75e-8	1.08E-3 **	1.05e-10

14. Bkg = instrument background
Bkg = 1.08e-6 μCi/cc

* These dose factors are from ODAM: stack release at a distance of 1260 meters NNW of DAEC

** Arbitrarily set equal to Xe-138

15. Σk_i = 1.13e-7

15a. $\Sigma_i (k_i \bullet DFS_i)$ 1.13e-10

15b. $\frac{\Sigma\ k_i}{\Sigma\ (k_i \bullet DFS_i)} = \frac{(\#15)}{(\#15a)} = \frac{1.13e-7}{1.13e-10} =$ 1.00e3

16. Limit = $L = \frac{1.06}{F} \times \frac{\Sigma k_i}{\Sigma (k_i \bullet DFS_i)} = \frac{1.06}{(\#9)} \times$ (The Less of **#15b** OR 3436)

Limit = L = $\frac{1.06}{(10000)}$ (1000) = ~~0.106~~ 1.06e-1 JPH 2-26-16

17. Hi Hi ALARM = A x(**#16**) = (1.0) (1.06e-1) = 1.06e-1 μCi/cc

The radioactive gas flow corresponding to the HI Hi setpoint:

Performed by: [signature] / Rangdsaw Date: 2-26-16

Independent Verification by: [signature] Date: 2-26-16

DAEC DUANE ARNOLD ENERGY CENTER	SURVEILLANCE TEST PROCEDURE TITLE: **K10 CALIBRATION**	STP NS791013 Page 10 of 68 Rev. 17
Prerequisites	Performance Date: 18-APR-2014	INITIALS

4-21-14

6.0 PREREQUISITES

6.1 Make a copy of the EMS database display. JG (CHEM)

6.2 From the Chemistry Supervisor or designee, obtain and record the following alarm setpoints. (Values will be used to confirm AS FOUND data.)

6.2.1 HI 8.60 E-6 µCi/cc JG (CHEM)

6.2.2 HIHI 3.64 E-1 µCi/cc JG (CHEM)

6.3 From the Chemistry Supervisor or designee, obtain and record the desired New HI alarm setpoint. (Value will be used for the AS LEFT setpoint.)

6.3.1 Desired HI 8.60 E-6 µCi/cc JG (CHEM)

6.4 Verify Sr-90 0.09 µCi source (UID #687) is available for use. JG (CHEM)

6.5 Verify the KAMAN/EMS IDT time and the HPGe System Computer time are within ± 30 seconds. JG (CHEM) JM

NOTE

When Kaman point sources are decay corrected, decay is to be from the date marked on the source to the test date. JG

6.6 Decay correct permissible range (8.5E4 – 9.0E4 cpm) for UID #687 and record below. JG (CHEM)

PERMISSIBLE DECAY CORRECTED RANGE:

3.88 E4 cpm to 4.11 E4 cpm SF (IV)

40479068

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ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. K-2 Charmar
2. Sample No. 17-2746
3. Sample Date 5-5-17
4. Sample Time 1342-1410
5. MWT 1911
6. Count Date 5-5-17
7. Count Time 1415
8. Monitor Reading (μCi/cc) 2.70E-7
9. Process Flow Rate (CFM) 72000
10. Sample Volume (mL) 47,600

Isotope	11 k_i μCi/mL	12 Dose Factor Vent DFV_i mrem/yr m³/μCi	13 Product k_i x DFV_i
Xe 133		294	
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13	4.71 E-9	8.83E3 **	4.16 E-5
16. Σk_i =	4.71 E-9	**16a.** $\Sigma(k_i\ DFV_i)$ = 4.16 E-5	

Flow Meter ID# L-729

Cal Due Date: 10-06-17

14. Bkg = Instrument background
    Bkg = 1.76 E-7 μCi/cc

15. X/Q = 4.3 x 10^{-6} sec/m³
    (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \bullet DFV_i)} = \frac{(\#16)}{(\#16a)} = \frac{4.71\ E\text{-}9}{4.16\ E\text{-}5} =$ 1.13 E-4

17. Limit = L = $\frac{1.06}{(F)(X/Q)} \times \frac{\Sigma\ ki}{\Sigma\ (ki \bullet DFVi)} = \frac{1.06}{(\#9)(\#15)}$ x (The lesser of **#16b** OR 1.81E-4) =

Limit = L = $\frac{1.06}{(72000)(4.3\ E\text{-}6)}$ (1.13 E-4) = 3.87 E-4

18. Hi Hi ALARM = A x(**#17**) = (1.0)(3.87 E-4) = 3.87 E-4 μCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: JCM Date: 05-05-17

Independent Verification by: RK Date: 5-5-17

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ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. Kaman 2 2. Sample No. 16-841
3. Sample Date 2-5-16 4. Sample Time 11:13 5. MWT 1911
6. Count Date 2-5-16 7. Count Time 11:19
8. Monitor Reading (µCi/cc) 4.65e-8 9. Process Flow Rate (CFM) 72,000
10. Sample Volume (mL) 45,000

Isotope	11 k_i µCi/mL	12 Dose Factor Vent DFV_i mrem/yr m³/µCi	13 Product $k_i \times DFV_i$
Xe 133	none identified	294	none identified
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13	7.05e-9	8.83E3 **	6.23e-5
16. Σk_i =	7.05e-4	**16a.** $\Sigma(k_i\ DFV_i)$ =	6.23e-5

Flow Meter ID# L729

Cal Due Date: 10-6-17

14. Bkg = Instrument background
Bkg = 6.35e-7 µCi/cc

15. $X/Q = 4.3 \times 10^{-6}$ sec/m³ (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \cdot DFV_i)} = \frac{(\#16)}{(\#16a)} = \frac{7.05e^{-9}}{6.23e^{-5}} = 1.13e^{-4}$

17. $\text{Limit} = L = \frac{1.06}{(F)(X/Q)} \times \frac{\Sigma\ ki}{\Sigma\ (ki \bullet DFVi)} = \frac{1.06}{(\#9)(\#15)}$ x (The lesser of **#16b** OR 1.81E-4) =

$\text{Limit} = L = \frac{1.06}{(72{,}000)(4.3 \times 10^{-6})}(1.13e^{-4}) = 3.87e^{-4}$

18. Hi Hi ALARM = A x (**#17**) = (1.0) (3.87e-4) = 3.87e-4 µCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: [signature] Date: 2-5-16

Independent Verification by: [signature] Date: 2-5-16

PLANT CHEMISTRY PROCEDURES 3200 MANUAL	PCP 8.3
ALARM SETPOINTS AND BACKGROUND DETERMINATION FOR KAMAN NORMAL RANGE MONITORS	Rev. 33 Page 13 of 14

ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. Kaman 2
2. Sample No. 14-2682
3. Sample Date 5-8-14
4. Sample Time 12:37 - 13:03
5. MWT 1907
6. Count Date 5-8-14
7. Count Time 1324
8. Monitor Reading (μCi/cc) 2.18e-7
9. Process Flow Rate (CFM) 72000
10. Sample Volume (mL) 49400

Isotope	11 k_i μCi/mL	12 Dose Factor Vent DFV_i $\frac{mrem}{yr}$ $\frac{m^3}{\mu Ci}$	13 Product $k_i \times DFV_i$
Xe 133	NONE	294	NONE
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41	NONE	8.84E3	NONE
N 13	7.84e-9	8.83E3 **	6.92e-5
16. Σk_i =	7.84e-9	16a. $\Sigma(k_i DFV_i)$ =	6.92e-5

Flow Meter ID# L715

Cal Due Date: 7-22-16

14. Bkg = Instrument background
Bkg = 1.10e-6 μCi/cc

15. $X/Q = 4.3 \times 10^{-6}$ sec/m^3
(atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \cdot DFV_i)} = \frac{(\#16)}{(\#16a)} = \frac{7.84e\text{-}9}{6.92e\text{-}5} = $ 1.13e-4

17. Limit = L = $\frac{1.06}{(F)(X/Q)} \times \frac{\Sigma \ k_i}{\Sigma \ (k_i \bullet DFV_i)} = \frac{1.06}{(\ \#9\ )(\ \#15\ )}$ x (The lesser of **#16b** OR 1.81E-4) =

Limit = L = $\frac{1.06}{(\ 72000\ )(\ 4.3e\text{-}6\ )}$ (1.13e-4) = 3.87e-4 ✓

18. Hi Hi ALARM = A x (**#17**) = (1.0) (3.87e-4) = 3.87e-4 μCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: Kelly Date: 5-8-14

Independent Verification by: Tom Mori Date: 5-8-14

PLANT CHEMISTRY PROCEDURES 3200 MANUAL	PCP 8.3
ALARM SETPOINTS AND BACKGROUND DETERMINATION FOR KAMAN NORMAL RANGE MONITORS	Rev. 33 Page 13 of 14

ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. Kamen 4 2. Sample No. 16-1551
3. Sample Date 3-4-16 4. Sample Time 1147 5. MWT 1911
6. Count Date 3-4-16 7. Count Time 1156
8. Monitor Reading (µCi/cc) 4.49e-8 9. Process Flow Rate (CFM) 93,000
10. Sample Volume (mL) 45,000

Isotope	11 k_i µCi/mL	12 Dose Factor Vent DFV_i $\frac{mrem}{yr}\ \frac{m^3}{\mu Ci}$	13 Product $k_i \times DFV_i$
Xe 133	None identified	294 NONE IDENTIFIED	
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13	6.19e-9	8.83E3 **	5.47e-5
16. Σk_i = 6.19e-9		**16a.** $\Sigma(k_i\ DFV_i)$ = 5.47e-5	

Flow Meter ID# L729

Cal Due Date: 10/6/17

14. Bkg = Instrument background
Bkg = 6.19e-9 µCi/cc

15. X/Q = 4.3×10^{-6} sec/m^3 (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \cdot DFV_i)} = \frac{(\#16)}{(\#16a)} = \frac{6.19e\text{-}9}{5.47e\text{-}5} = 1.13e\text{-}4$

17. Limit = L = $\frac{1.06}{(F)(X/Q)} \times \frac{\Sigma\ ki}{\Sigma\ (ki \bullet DFVi)} = \frac{1.06}{(\ \#9\ )(\ \#15\ )}$ x (The lesser of **#16b** OR 1.81E-4) =

Limit = L = $\left(\frac{1.06}{(93000)(4.3e\text{-}6)}\right)(1.13e\text{-}4)$ = 3.00e-4

18. Hi Hi ALARM = A x (#17) = (1.0) (3.00e-4) = 3.00e-4 µCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint.

Performed by: [signature] (JM) Date: 3-4-6

Independent Verification by: [signature] Date: 3-4-6

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ALARM SETPOINTS AND BACKGROUND DETERMINATION FOR KAMAN NORMAL RANGE MONITORS	Rev. 33 Page 13 of 14

ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. KAMAN 4 CHARMAL 2. Sample No. 14-7631
3. Sample Date 12-9-14 4. Sample Time 1130 5. MWT 1036
6. Count Date 12-9-14 7. Count Time 1138 ~~1036~~ 12-9-14
8. Monitor Reading (μCi/cc) 5.47E-9 9. Process Flow Rate (CFM) ~~5.70E4~~ 93000
10. Sample Volume (mL) ~~48600~~

Isotope	11 k_i μCi/mL	12 Dose Factor Vent DFV_i $\frac{mrem}{yr}$ $\frac{m^3}{\mu Ci}$	13 Product $k_i \times DFV_i$
Xe 133	None Identified	294	None Identified
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13		8.83E3 **	
16. Σk_i = N/A		16a. $\Sigma(k_i\,DFV_i)$ = N/A	

Flow Meter ID# L-760

Cal Due Date: 7.11.17

14. Bkg = Instrument background
Bkg = 5.45E-7 μCi/cc

15. X/Q = 4.3×10^{-6} sec/m³
(atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \cdot DFV_i)} = \frac{(\#16)}{(\#16a)} = \frac{N/A}{N/A} = N/A$

17. Limit = L = $\frac{1.06}{(F)(X/Q)} \times \frac{\Sigma\ ki}{\Sigma\ (ki \bullet DFVi)} = \frac{1.06}{(\#9)(\#15)}$ x (The lesser of #16b OR 1.81E-4) =

Limit = L = $\frac{1.06}{(93000)(4.3E-6)}$ (1.81E-4) = 4.80E-4

18. Hi Hi ALARM = A x (#17) = (1.0) (4.80E-4) = 4.80E-4 μCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: ZALESKY Date: 12-9-14

Independent Verification by: Tom Morris Date: 12-9-14

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ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. K-4 ~~13-1159~~ cal
2. Sample No. 13-1159
3. Sample Date 3-1-13
4. Sample Time 1330
5. MWT 1910
6. Count Date 3-1-13
7. Count Time 1338
8. Monitor Reading (μCi/cc) 1.76 E-8
9. Process Flow Rate (CFM) ~~5795~~ 3-1-13 93,000
10. Sample Volume (mL) 4.18 E4

Isotope	11 k_i μCi/mL	12 Dose Factor Vent DFV_i $\frac{mrem}{yr}$ $\frac{m^3}{\mu Ci}$	13 Product $k_i \times DFV_i$
Xe 133	NONE	294	NONE
Kr 85m	IDENTIFIED	1.17E3	IDENTIFIED
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13		8.83E3 **	
16. Σk_i = N/A		**16a.** $\Sigma(k_i\ DFV_i)$ = N/A	

Flow Meter ID# L760

Cal Due Date: 5-7-14

14. Bkg = Instrument background
Bkg = 5.62 E-7 μCi/cc

15. X/Q = 4.3 x 10^{-6} sec/m^3 (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \cdot DFV_i)} = \frac{(\#16)}{(\#16a)} = \frac{N/A}{N/A} = N/A$

17. Limit = L = $\frac{1.06}{(F)(X/Q)} \times \frac{\Sigma\ k_i}{\Sigma\ (k_i \bullet DFV_i)} = \frac{1.06}{(\#9)(\#15)}$ x (The lesser of #16b OR 1.81E-4) =

Limit = L = $\frac{1.06}{(93000)(4.3 \times 10^{-6})}$ (1.81 E-4) = 4.80 E-4

18. Hi Hi ALARM = A x (#17) = (1.0) (4.80E-4) = 4.80 E-4 μCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

19. Q = 472 (A · #17)#9
Q = 472 (1.0) (4.80 E-4) (93000)
Q = ~~2.11 E~~ μCi/sec 2.11 E4 uCi/sec

Performed by: [signature] Date: 3-1-13

Independent Verification by: [signature] Date: 3-1-13

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ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. K-6 Charmar 2. Sample No. 16-8841
3. Sample Date 11-30-16 4. Sample Time 1514 5. MWT 1911
6. Count Date 11-30-16 7. Count Time 1537
8. Monitor Reading (µCi/cc) 2.78E-7 9. Process Flow Rate (CFM) 93000
10. Sample Volume (mL) 48000

Isotope	11 k_i µCi/mL	12 Dose Factor Vent DFV_i mrem/yr m³/µCi	13 Product $k_i \times DFV_i$
Xe 133	None Identified	294	None Identified
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13		8.83E3 **	
16. Σk_i =	N/A	16a. $\Sigma (k_i DFV_i)$ =	N/A

Flow Meter ID# L-729

Cal Due Date: 10-6-17

14. Bkg = Instrument background
Bkg = 7.67E-7 µCi/cc

15. X/Q = 4.3 x 10⁻⁶ sec/m³ (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\dfrac{\Sigma k_i}{\Sigma(k_i \bullet DFV_i)} = \dfrac{(\#16)}{(\#16a)} = ____ = $ N/A

17. $\text{Limit} = L = \dfrac{1.06}{(F)(X/Q)} \times \dfrac{\Sigma\ ki}{\Sigma\ (ki \bullet DFVi)} = \dfrac{1.06}{(\#9)(\#15)}$ x (The lesser of #16b OR 1.81E-4) =

$\text{Limit} = L = \dfrac{1.06}{(93000)(4.3E-6)}(1.81E-4) = $ 4.80E-4

18. Hi Hi ALARM = A x (#17) = (1.0) (4.80E-4) = 4.80E-4 µCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: [signature] Date: 11-30-16

Independent Verification by: [signature] Date: 11-30-16

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ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. NS791009 K-6 2. Sample No. 15-2125
3. Sample Date 4-16-15 4. Sample Time 1600 5. MWT 1911
6. Count Date 4-16-15 7. Count Time 1614
8. Monitor Reading (µCi/cc) 1.70E-7 9. Process Flow Rate (CFM) 93,000
10. Sample Volume (mL) 4.25E4

Isotope	11 k_i µCi/mL	12 Dose Factor Vent DFV_i $\frac{mrem}{yr}$ $\frac{m^3}{\mu Ci}$	13 Product k_i x DFV_i
Xe 133	None Identified	294	None Identified
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13		8.83E3 **	
16. $\sum k_i$ = N/A		**16a.** $\sum(k_i \cdot DFV_i)$ = N/A	

Flow Meter ID# L729

Cal Due Date: 10-6-17

14. Bkg = Instrument background
Bkg = ______ µCi/cc

15. X/Q = 4.3 x 10^{-6} sec/m³ (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\sum k_i}{\sum(k_i \bullet DFV_i)} = \frac{(\#16)}{(\#16a)} = ____ = $ N/A

17. Limit = L = $\frac{1.06}{(F)(X/Q)} \times \frac{\sum k_i}{\sum(k_i \bullet DFV_i)} = \frac{1.06}{(\#9)(\#15)}$ x (The lesser of **#16b** OR 1.81E-4) =

Limit = L = $\frac{1.06}{(93{,}000)(4.3E\text{-}6)}$ (1.81E-4) = 4.80E-4

18. Hi Hi ALARM = A x (**#17**) = (1.0)(4.80E-4) = 4.80E-4 µCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: [signature] Date: 4-16-15

Independent Verification by: [signature] Date: 4-16-15

DAEC DUANE ARNOLD ENERGY CENTER	SURVEILLANCE TEST PROCEDURE TITLE: K6 CALIBRATION	STP NS791009 Page 64 of 66 Rev. 14
	Performance Date: 9-6-13	INITIALS

7.15.6 Record the following AS LEFT values:

a. AS LEFT HI-HI ALARM SETPOINT (from Step 7.15.4): 8/RA

4.80E-4 ~~9.80 E-4~~ RA 9-6-13 µCi/cc

b. AS LEFT HI ALARM SETPOINT (from Prerequisite 6.3.1): RA

3.5E-6 µCi/cc

c. AS LEFT BACKGROUND (from Step 7.13.50 or 7.14.49): RA

7.84E-7 µCi/cc

7.15.7 At the Kaman EMS IDT, verify the following has been correctly entered into the EMS database:

a. HI-HI alarm setpoint (from Step 7.15.6.a) RA

b. HI alarm setpoint (from Step 7.15.6.b) RA

c. Background concentration (from Step 7.15.6.c) RA

7.15.8 Update database values on the status board and in Labstats. ✓ RA

7.15.9 Attach completed setpoint calculation documentation (Step 7.15.4) to this STP. RA

(PRINT / SIGN)

Performed by:		Date:	Time:	Init.
Larry Isaacs	[signature]	9-6-13	1309	RA
Richard Potter	[signature]	9-6-13	1312	RP

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ALARM SETPOINTS AND BACKGROUND DETERMINATION FOR KAMAN NORMAL RANGE MONITORS	Rev. 33 Page 13 of 14

ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D: K8 Char Max 2. Sample No. 17-2246
3. Sample Date 4-13-17 4. Sample Time 1249 5. MWT 1900
6. Count Date 4-13-17 7. Count Time 1254
8. Monitor Reading (μCi/cc) 8.76e-9 9. Process Flow Rate (CFM) 93000
10. Sample Volume (mL) 47500

Isotope	11 k_i μCi/mL	12 Dose Factor Vent DFV_i $\frac{mrem}{yr}$ $\frac{m^3}{\mu Ci}$	13 Product $k_i \times DFV_i$
Xe 133	none identified	294	none identified
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13		8.83E3 **	
16. Σk_i =	N/A	**16a.** $\Sigma (k_i\ DFV_i)$ =	N/A

Flow Meter ID# L-729

Cal Due Date: 10-6-17

14. Bkg = Instrument background
Bkg = 4.47e-7 μCi/cc

15. $X/Q = 4.3 \times 10^{-6}$ sec/m³ (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \cdot DFV_i)} = \frac{(\#16)}{(\#16a)} =$ N/A = N/A

17. Limit = L = $\frac{1.06}{(F)(X/Q)} \times \frac{\Sigma\ ki}{\Sigma\ (ki \bullet DFVi)} = \frac{1.06}{(\#9)(\#15)}$ x (The lesser of **#16b** OR 1.81E-4) =

Limit = L = $\frac{1.06}{(93{,}000)(4.3 \times 10^{-6})}$ ($1.81e^{-4}$) = $4.80e^{-4}$

18. Hi Hi ALARM = A x (**#17**) = (1.0) ($4.80e^{-4}$) = $4.80e^{-4}$ μCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: [signature] Date: 4-13-17

Independent Verification by: Rany Isaacs Date: 4-13-17

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ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. Kaman 8   2. Sample No. 15-5934
3. Sample Date 10-8-15   4. Sample Time 1130   5. MWT 1911
6. Count Date 10-8-15   7. Count Time 1139
8. Monitor Reading (µCi/cc) 7.59e-8   9. Process Flow Rate (CFM) 93000
10. Sample Volume (mL) 45 e 4

Isotope	11 k_i µCi/mL	12 Dose Factor Vent DFV_i mrem/yr · m³/µCi	13 Product k_i x DFV_i
Xe 133	None Ident.	294	None
Kr 85m		1.17E3	Identified
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13	4.62 e-9	8.83E3 **	4.08e-5
16. Σk_i =	4.62e-9	**16a.** $\Sigma (k_i DFV_i)$ =	4.08 e-5

Flow Meter ID# L729

Cal Due Date: 10-6-17

14. Bkg = Instrument background
Bkg = 3.33e-7 µCi/cc

15. X/Q = 4.3 x 10^{-6} sec/m³ (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

$$\textbf{16b.}\ \frac{\Sigma k_i}{\Sigma(k_i \bullet DFV_i)} = \frac{(\ \#16\ )}{(\ \#16a\ )} = \frac{4.62e^{-9}}{4.08e^{-5}} = 1.13e^{-4}$$

$$\textbf{17.}\ \text{Limit} = L = \frac{1.06}{(F)(X/Q)} \times \frac{\Sigma\ ki}{\Sigma\ (ki \bullet DFVi)} = \frac{1.06}{(\ \#9\ )(\ \#15\ )} \times \text{(The lesser of \#16b OR 1.81E-4)} =$$

$$\text{Limit} = L = \frac{1.06}{(93000\ )(4.3e^{-6}\ )}(\ 1.13e^{-4}\ ) = \cancel{2.99}\ 3.00\ e^{-4}$$

10-8-15

18. Hi Hi ALARM = A x (**#17**) = (1.0) (3.00e-4) = 3.00e-4 µCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: [signature] Date: 10-8-15

Independent Verification by: [signature] Date: 10-8-15

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ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. Kaman 8
2. Sample No. 14 - 2491
3. Sample Date 4-29-14
4. Sample Time 1103
5. MWT 1901
6. Count Date 4-29-14
7. Count Time 1110
8. Monitor Reading (µCi/cc) 1.22 E-7
9. Process Flow Rate (CFM) 93000 ✓
10. Sample Volume (mL) 48600

Isotope	11 k_i µCi/mL	12 Dose Factor Vent DFV_i $\frac{mrem}{yr}$ $\frac{m^3}{\mu Ci}$	13 Product k_i x DFV_i
Xe 133	NONE	294	NONE
Kr 85m	IDENTIFIED	1.17E3	Identified
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13		8.83E3 **	
16. Σk_i =	N/A	**16a.** $\Sigma(k_i\ DFV_i)$ =	N/A

Flow Meter ID# L 760

Cal Due Date: 5-7-14

14. Bkg = Instrument background
    Bkg = N/A µCi/cc

15. X/Q = 4.3 x 10^{-6} sec/m³ (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \bullet DFV_i)} = \frac{(\#16)}{(\#16a)} =$ N/A = N/A

17. Limit = L = $\frac{1.06}{(F)(X/Q)}$ x $\frac{\Sigma\ ki}{\Sigma\ (ki \bullet DFVi)} = \frac{1.06}{(\#9)(\#15)}$ x (The lesser of **#16b** OR 1.81E-4) =

Limit = L = $\frac{1.06}{(93000)(4.3\ E^{-6})}$ (1.81e^{-4}) = 4.80 e^{-4} ✓

18. Hi Hi ALARM = A x (**#17**) = (1.0) (4.80 e^{-4}) = 4.80 e^{-4} µCi/cc ✓
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: [signature] Date: 4-29-14

Independent Verification by: [signature] Date: 4-29-14

[signature] 4-29-14

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ALARM SETPOINTS AND BACKGROUND DETERMINATION FOR KAMAN NORMAL RANGE MONITORS	Rev. 33 Page 13 of 14

ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. K-12 Channar 2. Sample No. 17-2396
3. Sample Date 4-20-17 4. Sample Time 1322 5. MWT 1910
6. Count Date 4-20-17 7. Count Time 1345
8. Monitor Reading (µCi/cc) 2.61E-8 9. Process Flow Rate (CFM) 75000
10. Sample Volume (mL) 48,000

Isotope	11 k_i µCi/mL	12 Dose Factor Vent DFV_i $\frac{\text{mrem}}{\text{yr}}$ $\frac{m^3}{\mu Ci}$	13 Product $k_i \times DFV_i$
Xe 133	None Ident	294	None Ident
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13		8.83E3 **	
16. Σk_i =	N/A	**16a.** $\Sigma(k_i\ DFV_i)$ =	N/A

Flow Meter ID# L-729

Cal Due Date: 10-6-17

14. Bkg = Instrument background
Bkg = 4.05E-7 µCi/cc

15. X/Q = 4.3 x 10^{-6} sec/m^3 (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \bullet DFV_i)} = \frac{(\ \#16\ )}{(\ \#16a\ )} =$ N/A = N/A

17. Limit = L = $\frac{1.06}{(F)(X/Q)} \times \frac{\Sigma\ ki}{\Sigma\ (ki \bullet DFVi)} = \frac{1.06}{(\ \#9\ )(\ \#15\ )}$ x (The lesser of **#16b** OR 1.81E-4) =

Limit = L = $\frac{1.06}{(\ 75000\ )(\ 4.3E-6\ )}$ (1.81E-4) = 5.95E-4

18. Hi Hi ALARM = A x (**#17**) = (1.0) (5.95E-4) = 5.95E-4 µCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: [signature] Date: 4-20-17

Independent Verification by: [signature] Date: 4-20-17

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ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. Kaman 12
2. Sample No. 15-6930
3. Sample Date 11-19-15
4. Sample Time 13:18
5. MWT 1910
6. Count Date 11-19-15
7. Count Time 13:27
8. Monitor Reading (µCi/cc) 5.3e-7
9. Process Flow Rate (CFM) 75000
10. Sample Volume (mL) 3.99e4

Isotope	11 k_i µCi/mL	12 Dose Factor Vent DFV_i $\frac{mrem}{yr}$ $\frac{m^3}{\mu Ci}$	13 Product k_i x DFV_i
Xe 133	None Identified	294	None Identified
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13	5.44e-7	8.83E3 **	4.80e-5
16. Σk_i =	5.44e-7	**16a.** $\Sigma (k_i\ DFV_i)$ =	4.80e-5

Flow Meter ID# L715

Cal Due Date: 9-12-17

14. Bkg = Instrument background
    Bkg = 5.99e-7 µCi/cc

15. X/Q = 4.3 x 10^{-6} sec/m^3 (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \cdot DFV_i)} = \frac{(\#16)}{(\#16a)} = \frac{5.99e^{-7}}{4.80e^{-5}} = 1.13e^{-4}$

17. Limit = L = $\frac{1.06}{(F)(X/Q)} \times \frac{\Sigma\ k_i}{\Sigma\ (k_i \bullet DFV_i)} = \frac{1.06}{(\#9)(\#15)}$ x (The lesser of **#16b** OR 1.81E-4) =

Limit = L = $\frac{1.06}{(75000)(4.3\times10^{-6})}\ (1.13e^{-4}) = 3.71e^{-4}$

18. Hi Hi ALARM = A x (**#17**) = (1.0) (3.71e-4) = 3.71e-4 µCi/cc
    The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: [signature] Date: 11-19-15

Independent Verification by: [signature] Date: 11-19-15

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ATTACHMENT 1

VENT MONITORS GASEOUS DETECTOR HI HI SETPOINT

1. Sample I.D. K12 Charmar 2. Sample No. 14-438
3. Sample Date 1-22-14 4. Sample Time 5. MWT 1911
6. Count Date 1-22-14 7. Count Time 1237
8. Monitor Reading (μCi/cc) 1.44e-7 9. Process Flow Rate (CFM) 75000
10. Sample Volume (mL) 48600

Isotope	11 k_i μCi/mL	12 Dose Factor Vent DFV_i mrem/yr · m³/μCi	13 Product $k_i \times DFV_i$
Xe 133	N/A	294 N/A	N/A
Kr 85m		1.17E3	
Kr 88		1.47E4	
Xe 135		1.81E3	
Kr 87		5.92E3	
Xe 138		8.83E3	
Xe 135m		3.12E3	
Xe 133m		2.51E2	
Ar 41		8.84E3	
N 13		8.83E3 **	N/A
16. Σk_i = N/A		16a. $\Sigma(k_i\, DFV_i)$ = NA	

Flow Meter ID# L729
Cal Due Date: 10-9-14

14. Bkg = Instrument background
Bkg = 4.61 E-7 μCi/cc

15. X/Q = 4.3×10^{-6} sec/m³ (atmospheric dispersion)

** Arbitrarily set equal to Xe-138

16b. $\frac{\Sigma k_i}{\Sigma(k_i \cdot DFV_i)} = \frac{(\#16)}{(\#16a)}$ = N/A = N/A

17. $\text{Limit} = L = \frac{1.06}{(F)(X/Q)} \times \frac{\Sigma\ ki}{\Sigma\ (ki \bullet DFVi)} = \frac{1.06}{(\#9)(\#15)}$ x (The lesser of #16b OR 1.81E-4) =

Limit = L = $\frac{1.06}{(75000)(4.3\times10^{-6})}$ (1.81e-4) = ~~5.95e-5~~ pm 1-22-14 5.95e-4

18. Hi Hi ALARM = A x (#17) = (1.0) (5.95e-5) = ~~5.95e-5~~ pm 1-22-14 5.95e-4 μCi/cc
The radioactive gas flow corresponding to the Hi Hi setpoint:

Performed by: Kelley Date: 1-22-14

Independent Verification by: Sicma Date: 1-22-14

START 1159
STOP 1225
46800 — 1.80 LPM

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ATTACHMENT 1

LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 17-5499 2. Sample Date & Time 8-28-17 / 0035
3. Stream/Monitor Description GSW Rm -4767
4. Effluent Monitor Reading (cps) 10
5. Effluent Flow (gpm) 9600.
6. Average effluent flow during time represented by sample, F_1 (gpm) N/A
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) N/A
8. Monitor calibration factor, g, (cps/µCi/mL) 2.19e6
9. Previous alarm value setpoint (cps) 765
10. Fraction to apply as a safety margin, A = 0.5

$$\textbf{Setpoint} = 10\times\left[\frac{\Sigma_I K_I}{\Sigma_I(K_I \div WEC_I)}\times g\times\frac{F_2}{F_1}\times A\right]+Bkg \quad = \quad \text{Setpoint}=10\times\left[\frac{(15)(8)(7)}{(16)(6)}\times(10)\right]+(4)$$

$$\text{Setpoint} = 5\text{x}\left[\frac{(15)(8)(7)}{(16)(6)}\right]+(4)$$

$$\text{Setpoint} = 5\text{x}\left[\frac{(\ 1.06e^{-2}\ )(\ 2.19e^{6}\ )(\ N/A\ )}{(\ 219\ )(\ N/A\ )}\right]+(\ 10\ )$$

11. Setpoint = ~~1.06e-2~~ 8-28-17 540 ✓

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11)-(9)}{(9)} = \frac{(\ 540\ )-(\ 765\ )}{(\ 765\ )}$

12. Fractional Change = - 0.294 ✓

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint

13. Monitor Hi Alarm = 765 ✓

14. Radwaste Monitor Hi Alarm = .16 (11) = .16 (   ) = N/A cps ✓

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ATTACHMENT 1

LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 15 - 7198 ✓ 2. Sample Date & Time 12-1-15 / 11:03 ✓
3. Stream/Monitor Description GSW rad monitor RM-4767 ✓
4. Effluent Monitor Reading (cps) 10 ✓
5. Effluent Flow (gpm) 9600 gpm ✓
6. Average effluent flow during time represented by sample, F_1 (gpm) NR ✓
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) NR ✓
8. Monitor calibration factor, g, (cps/µCi/mL) ~~[illegible]~~ 2.19 e6 ✓
9. Previous alarm value setpoint (cps) 765 cps ✓
10. Fraction to apply as a safety margin, A = 0.5

$$\textbf{Setpoint} = 10\times\left[\frac{\Sigma_i K_i}{\Sigma_i(K_i \div WEC_i)}\times g\times\frac{F_2}{F_1}\times A\right]+Bkg \quad = \quad \text{Setpoint}=10\times\left[\frac{(15)(8)(7)}{(16)(6)}\times(10)\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(15)(8)(7)}{(16)(6)}\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(9.43e^{-3})(2.19e^{6})(\ NR\ )}{(\ 177.54\ )(\ NR\ )}\right]+(\ 10\ )$$

11. Setpoint = 592 ✓

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11)-(9)}{(9)} = \frac{(592)-(765)}{(765)}$

12. Fractional Change = -.226 ✓

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint

13. Monitor Hi Alarm = 765 cps ✓
14. Radwaste Monitor Hi Alarm = .16 (11) = .16 ( ) = N/A cps ✓

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ATTACHMENT 1

LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 14 - 1787 ✓ 2. Sample Date & Time 3-28-14/0030 ✓
3. Stream/Monitor Description GSW 4767 ✓
4. Effluent Monitor Reading (cps) 9 ✓
5. Effluent Flow (gpm) 9600 ✓
6. Average effluent flow during time represented by sample, F_1 (gpm) N/A ✓
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) N/A ✓
8. Monitor calibration factor, g, (cps/μCi/mL) 2.19E6 ✓ ✓
9. Previous alarm value setpoint (cps) 2254 ✓
10. Fraction to apply as a safety margin, A = 0.5 ✓

$$\textbf{Setpoint} = 10\times\left[\frac{\Sigma_i K_i}{\Sigma_i(K_i \div WEC_i)}\times g\times\frac{F_2}{F_1}\times A\right]+Bkg \quad = \quad \text{Setpoint}=10\times\left[\frac{(15)(8)(7)}{(16)(6)}\times(10)\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(15)(8)(7)}{(16)(6)}\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(7.69E\text{-}3)(2.19E6)(N/A)}{(111.32)(N/A)}\right]+(9)$$

11. Setpoint = 765 ✓

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11)-(9)}{(9)} = \frac{(765)-(2254)}{(2254)}$

12. Fractional Change = −0.66 ✓

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint

13. Monitor Hi Alarm = 765 ✓
14. Radwaste Monitor Hi Alarm = .16 (11) = .16 (N/A) = N/A cps ✓

DAEC DUANE ARNOLD ENERGY CENTER	SURVEILLANCE TEST PROCEDURE TITLE: RHRSW RADIATION MONITOR CALIBRATION RM-1997	STP NS790305 Page 6 of 18 Rev. 14
Prerequisites	Performance Date: 2-13-17	INITIALS

6.0 PREREQUISITES

6.1 From the Chemistry Supervisor, obtain the current UPSCALE HI alarm setpoint. Record below and in the trip column of the step indicated. ______ Chemistry

Step 7.1.10 614 cps

6.2 From the Chemistry Supervisor, obtain the current high voltage setting. Record below and in the step indicated. ______ Chemistry

Step 7.1.25 750 VDC

NOTE

Original Transfer Calibration Count Rate is the count rate of the 8 μCi source taken from the last time that the mockup was used to determine the detector efficiency. This can be found in the Effluent Monitor Alarm Setpoint book. It is then decay corrected to the date that this STP is being performed.

6.3 From the Chemistry Supervisor, obtain the following source information and record below:

6.3.1 Original Trans Cal Count Rate 2.29E4 cps ______ Chemistry

6.3.2 Source Number UID# 647 Cs-137 ______ Chemistry

6.3.3 Original Date of Cal Count Rate 8-24-15 ______ Chemistry

6.3.4 Geometry Point ______ Chemistry

6.3.5 Old Efficiency 6.62E-7 μCi/cc/cps ______ Chemistry

6.4 Decay correct the Original Transfer Calibration Count Rate. Record and transfer the value to the step indicated below: ______ Chemistry

Decay Corrected Transfer Count Rate 2.21E4 cps

(Transfer to Step 7.1.37.)

6.5 As directed by PCP 8.7, analyze a sample of unfiltered reactor water and calculate the UPSCALE HI setpoint. Record below and in the trip column of the table listed. ______ Chemistry

Step 7.1.28 421 cps

W/O 40328724

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LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 15-4896 2. Sample Date & Time 8-29-15 / 0027
3. Stream/Monitor Description Rm-1997 (RHRSW/ESW)
4. Effluent Monitor Reading (cps) 30
5. Effluent Flow (gpm) 4800
6. Average effluent flow during time represented by sample, F_1 (gpm) N/A
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) N/A
8. Monitor calibration factor, g, (cps/µCi/mL) 1.51e6
9. Previous alarm value setpoint (cps) 614
10. Fraction to apply as a safety margin, A = 0.5

$\frac{1}{6.62e^{-7}}$

$$\textbf{Setpoint} = 10 \times \left[\frac{\Sigma_i K_i}{\Sigma_i (K_i \div WEC_i)} \times g \times \frac{F_2}{F_1} \times A \right] + Bkg \quad = \quad \text{Setpoint} = 10 \times \left[\frac{(15)(8)(7)}{(16)(6)} \times (10) \right] + (4)$$

$$\text{Setpoint} = 5 \times \left[\frac{(15)(8)(7)}{(16)(6)} \right] + (4)$$

$$\text{Setpoint} = 5 \times \left[\frac{(\cancel{1.01e^{-2}}\ 1.10e^{-2})(1.51e^6)(N/A)}{(\cancel{102.078}\ 153.84)(N/A)} \right] + (30)$$

11. Setpoint = ~~735.6~~ 569.85

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11) - (9)}{(9)} = \frac{(\cancel{735.6}\ 569.9) - (614)}{(614)}$ 8-29-15

12. Fractional Change = ~~0.20~~ -0.07

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint

13. Monitor Hi Alarm = 614

14. Radwaste Monitor Hi Alarm = .16 (11) = .16 ( ) = N/A cps

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LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 14-884 2. Sample Date & Time 2-14-14 / 0021
3. Stream/Monitor Description RHRSW / ESW's Rm 1997
4. Effluent Monitor Reading (cps) 25
5. Effluent Flow (gpm) RHRSW 'A' = 4800gpm / RHRSW 'B' = 4800gpm
6. Average effluent flow during time represented by sample, F_1 (gpm) N/A
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) N/A
8. Monitor calibration factor, g, (cps/µCi/mL) 1.51E6
9. Previous alarm value setpoint (cps) 614
10. Fraction to apply as a safety margin, A = 0.5

$$\textbf{Setpoint} = 10\times\left[\frac{\Sigma_i K_i}{\Sigma_i(K_i \div WEC_i)}\times g \times \frac{F_2}{F_1}\times A\right] + Bkg \quad = \quad \text{Setpoint}=10\times\left[\frac{(15)(8)(7)}{(16)(6)}\times(10)\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(15)(8)(7)}{(16)(6)}\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(7.67E\text{-}3)(\ 1.51E6\ )(\ N/A\ )}{(\ 111.86\ )(\ N/A\ )}\right]+(\ 25\ )$$

11. Setpoint = 543 ✓

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11)-(9)}{(9)} = \frac{(\ 543\ )-(\ 614\ )}{(\ 614\ )}$

12. Fractional Change = −0.12 ✓

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint ⇒ OLD SETPOINT ✓

13. Monitor Hi Alarm = 614 ✓

14. Radwaste Monitor Hi Alarm = .16 (11) = .16 (N/A) = N/A cps

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ATTACHMENT 1

LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 17-316 2. Sample Date & Time 1-16-17 / 0016
3. Stream/Monitor Description RM-4268 RHRSW/ESW Dilution Line (Rupture)
4. Effluent Monitor Reading (cps) 10
5. Effluent Flow (gpm) 9600
6. Average effluent flow during time represented by sample, F_1 (gpm) N/A
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) N/A
8. Monitor calibration factor, g, (cps/µCi/mL) 1.92 e6
9. Previous alarm value setpoint (cps) 530
10. Fraction to apply as a safety margin, A = 0.5

$$\textbf{Setpoint} = 10\times\left[\frac{\Sigma_i K_i}{\Sigma_i(K_i \div WEC_i)}\times g\times\frac{F_2}{F_1}\times A\right]+Bkg \quad = \quad \text{Setpoint}=10\times\left[\frac{(15)(8)(7)}{(16)(6)}\times(10)\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(15)(8)(7)}{(16)(6)}\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(\ 8.34e^{-3}\ )(\ 1.92e^{6}\ )(\ N/A\ )}{(\ 139.13\ )(\ N/A\ )}\right]+(\ 10\ )$$

11. Setpoint = 585.5

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11)-(9)}{(9)} = \frac{(585.5)-(530)}{(530)}$

12. Fractional Change = 0.105

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint

13. Monitor Hi Alarm = 530

14. Radwaste Monitor Hi Alarm = .16 (11) = .16 ( ) = N/A cps

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LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 15-6175 2. Sample Date & Time 10-19-15/0019
3. Stream/Monitor Description RHRSW/ESW Rupture RM-4268
4. Effluent Monitor Reading (cps) 20
5. Effluent Flow (gpm) RHRSW A - 4800 gpm, RHRSW B 4800 gpm
6. Average effluent flow during time represented by sample, F_1 (gpm) N/A
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) N/A
8. Monitor calibration factor, g, (cps/μCi/mL) 2.29E6
9. Previous alarm value setpoint (cps) 863
10. Fraction to apply as a safety margin, A = 0.5

$$\textbf{Setpoint} = 10\times\left[\frac{\Sigma_i K_i}{\Sigma_i(K_i \div WEC_i)}\times g\times\frac{F_2}{F_1}\times A\right]+Bkg \quad = \quad \text{Setpoint}=10\times\left[\frac{(15)(8)(7)}{(16)(6)}\times(10)\right]+(4)$$

$$\text{Setpoint} = 5\text{x}\left[\frac{(15)(8)(7)}{(16)(6)}\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(\ 9.90E\text{-}3\ )(\ 2.29E6\ )(\ N/A\ )}{(\ 174.97\ )(\ N/A\ )}\right]+(\ 20\ )$$

11. Setpoint = 668 ✓

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11)-(9)}{(9)} = \frac{(668)-(863)}{(863)}$

12. Fractional Change = -0.226 ✓

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint

13. Monitor Hi Alarm = 863 ✓

14. Radwaste Monitor Hi Alarm = .16 (11) = .16 (   ) = N/A cps ✓

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ATTACHMENT 1

LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 14-884 2. Sample Date & Time 2-14-14 / 0021
3. Stream/Monitor Description RHRSW/ESW RUPTURE: RM 4268
4. Effluent Monitor Reading (cps) 20
5. Effluent Flow (gpm) RHRSW 'A' = 4800gpm, RHRSW 'B' = 4800gpm
6. Average effluent flow during time represented by sample, F_1 (gpm) N/A
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) N/A
8. Monitor calibration factor, g, (cps/µCi/mL) 2.29E6 ✓
9. Previous alarm value setpoint (cps) 863 ✓
10. Fraction to apply as a safety margin, A = 0.5

$$\textbf{Setpoint} = 10 \times \left[\frac{\Sigma_i K_i}{\Sigma_i (K_i \div WEC_i)} \times g \times \frac{F_2}{F_1} \times A \right] + Bkg \quad = \quad \text{Setpoint} = 10 \times \left[\frac{(15)(8)(7)}{(16)(6)} \times (10) \right] + (4)$$

$$\text{Setpoint} = 5 \times \left[\frac{(15)(8)(7)}{(16)(6)} \right] + (4)$$

$$\text{Setpoint} = 5 \times \left[\frac{(7.67E\text{-}3)(2.29E^6)(N/A)}{(111.86)(N/A)} \right] + (20)$$

11. Setpoint = 805 ✓

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11) - (9)}{(9)} = \frac{(805) - (863)}{(863)}$

12. Fractional Change = -0.07 ✓

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint ⇒ OLD SETPOINT ✓

13. Monitor Hi Alarm = 863 ✓

14. Radwaste Monitor Hi Alarm = .16 (11) = .16 (N/A) = N/A cps

3.7 PLANT SYSTEMS

3.7.8 Spent Fuel Storage Pool Water Level

LCO 3.7.8 The spent fuel storage pool water level shall be $\geq$ 36 ft.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1 -----------NOTE---------- LCO 3.0.3 is not applicable. ------------------------------ Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify the spent fuel storage pool water level is $\geq$ 36 ft.	In accordance with the Surveillance Frequency Control Program

PROBABLE ANNUNCIATORS

1C03A	A1	FUEL POOL EXHAUST HIGH-HIGH RADIATION
	B1	FUEL POOL EXHAUST HIGH RADIATION
1C04B	B6	NEW FUEL STORAGE AREA ARM HI RAD
1C04B	C6	SPENT FUEL STORAGE AREA ARM HI RAD
1C05B	C8	PCIS GROUP “3” ISOLATION INITIATED
1C09A	A2	NW DRYWELL RADIATION LEVEL HI-HI
	B2	NW DRYWELL RADIATION LEVEL HI
1C09B	A2	SOUTH DRYWELL RADIATION LEVEL HI-HI
	B2	SOUTH DRYWELL RADIATION LEVEL HI
1C35A	A1	REFUELING FLOOR NORTH END HI RADIATION
	A2	REFUELING FLOOR SOUTH END HI RADIATION

PROBABLE INDICATIONS

1. Lowering cavity and/or Spent Fuel Pool level on the 5th floor. Visual
2. Lowering cavity level Floodup Range on level indicator, LI-4541 (at 1C04).
3. Lowering Skimmer Surge Tank level on level indicator, LI-3412 (at 1C04). not used in EAL
4. Lowering Fuel Pool level on level indicator, LI-3413 (at 1C04).
5. Rising radiation levels on any of the following ARMs:

   Spent Fuel Pool Area, RI-9178
   North Refuel Floor, RI-9163
   New Fuel Vault Area, RI-9153
   South Refuel Floor, RI-9164

6. Rising Drywell radiation levels on either of the following (at 1C09):

   NW Drywell Area Hi Range Rad Monitor, RIM-9184A
   South Drywell Area Hi Range Rad Monitor, RIM-9184B

Development of EAL Threshold values from NEE-323-CALC-004

Calculated values are provided in Calc-004 as shown below.

Table 2 – Recommended RA1 Liquid EALs

Rad Monitor	Equip.	Modes 1,2,3	Modes 4, 5
		cps	cps
GSW	RE-4767	2.32E+4	1.04E+4
RHRSW/ESW	RE-1997	1.60E+4	7.20E+3
RHRSW Dilution Line	RE-4268	2.42E+4	1.09E+4

The following table of threshold values was developed for use in the DAEC EAL scheme by averaging the separate Mode 1-3 and Mode 4-5 thresholds from Calc-004, and then rounding the average values for ease of EAL evaluator use, as well as to provide a step-wise progression through the emergency classification.

	Monitor	GE	SAE	Alert
Liquid	GSW rad monitor (RIS-4767)	—	—	2.0E+04 cps
	RHRSW & ESW rad monitor (RM-1997)	—	—	1.0E+04 cps
	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	—	—	2.0E+04 cps

ENERCON Excellence—Every project. Every day.	CALCULATION COVER SHEET	CALC NO. NEE-323-CALC-004 REV. 00 PAGE NO. 1 of 23

Title:	Revised Liquid Radiological EALs per NEI 99-01	Client: Duane Arnold Energy Center Project Identifier: NEE-323

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions, including preliminary information, that require confirmation? (If **YES**, identify the assumptions.)	☐	☒
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, identify the design verified calculation.) **Design Verified Calculation No.** ______	☐	☒
3	Does this calculation supersede an existing Calculation? (If **YES**, identify the design verified calculation.) **Superseded Calculation No.** ______	☐	☒

Scope of Revision:
Initial Issue

Revision Impact on Results:
Initial Issue

Study Calculation ☐ **Final Calculation** ☒

Safety-Related ☐ **Non-Safety-Related** ☒

(Print Name and Sign)

Originator: Jay Bhatt	**Date:** **12/12/17**
Design Verifier[1] (Reviewer if NSR): Ryan Skaggs	**Date:** **12/12/17**
Approver: Zachary Rose	**Date:** **12/12/17**

Note 1: For non-safety-related calculation, design verification can be substituted by review.

ENERCON Excellence—Every project. Every day.	CALCULATION REVISION STATUS SHEET	CALC NO. NEE-323-CALC-004
		REV. 00

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
00	12/12/17	Initial Issue

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
All	00		

APPENDIX/ATTACHMENT REVISION STATUS

APPENDIX NO.	NO. OF PAGES	REVISION NO.	ATTACHMENT NO.	NO. OF PAGES	REVISION NO.
A	12	00	1	4	0
			2	3	0

1.0 Purpose and Scope

The Duane Arnold Energy Center is implementing the guidance of Revision 6 to NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," which is the industry-developed methodology for emergency classification for the current operating fleet. Changes to the definitions of the condition for entry into the Emergency Action Level (EAL) RA1 result in the development of a new entry threshold value for this EAL.

This calculation determines the liquid radiation monitor readings that correspond to the new EAL thresholds for the release of liquid radioactivity resulting in offsite dose greater than 10 mrem Total Effective Dose Equivalent (TEDE) or 50 mrem thyroid Committed Dose Equivalent (CDE) for one hour of exposure.

2.0 Summary of Results and Conclusions

A spreadsheet was used to calculate the monitor counts per second (cps) reading necessary to reach offsite dose of 10 mrem TEDE or 50 mrem child thyroid organ dose as described in Section 7.0. The output from that spreadsheet is seen below.

Table 1 – Monitor Response for Liquid Radiological EAL Thresholds

Rad Monitor	Equip. ID	Modes 1,2,3 2 Hour Decay cps		Modes 4, 5 36 Hour Decay cps	
		10 mrem TEDE	50 mrem Thyroid	10 mrem TEDE	50 mrem Thyroid
General Service Water (GSW)	RE-4767	23,200	49,100	10,400	14,000
Residual Heat Removal Service Water (RHRSW)/Essential Service Water (ESW)	RE-1997	16,000	33,800	7,200	9,650
RHRSW Dilution Line	RE-4268	24,200	51,300	10,900	14,650

For a given scenario, the threshold is always met for the TEDE dose before it is met for the organ dose.

The recommended RA1 Liquid EALs are:

Table 2 – Recommended RA1 Liquid EALs

Rad Monitor	Equip.	Modes 1,2,3	Modes 4, 5
		cps	cps
GSW	RE-4767	2.32E+4	1.04E+4
RHRSW/ESW	RE-1997	1.60E+4	7.20E+3
RHRSW Dilution Line	RE-4268	2.42E+4	1.09E+4

3.0 References

3.1 DAEC Offsite Dose Assessment Manual (ODAM), Rev. 37.

3.2 Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water, and Soil, 1993.

3.3 Code of Federal Regulations, 10CFR20, January 2013.

3.4 NUREG-1940, RASCAL 4: Description of Models and Methods, United States Nuclear Regulatory Commission, Office of Nuclear Security and Incident Response, 2012.

3.5 American National Standard Institute (ANSI/ANS). 1999. "Radioactive Source Term for Normal Operation of Light-Water Reactors," ANSI/ANS-18.1-1999, American Nuclear Society, La Grange Park, IL.

3.6 Plant Chemistry Procedure PCP 8.7, Alarm Setpoints for Liquid Rad. Monitors.

3.7 NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors", Rev. 6.

4.0 Assumptions

4.1 For the calculation determining the RA1 EAL for Reactor Modes 1, 2 and 3, the source is assumed to have decayed for 2 hours before reaching the receptor. This decay time is appropriate to produce best estimate results for liquid effluent thresholds for the corresponding Reactor Modes.

4.2 For the calculation determining the RA1 EAL for Reactor Modes 4 and 5, the source is assumed to have decayed for 36 hours before reaching the receptor. This decay time is appropriate to produce best estimate results for liquid effluent thresholds for the corresponding Reactor Modes.

4.3 Per the ODAM, a mixing ratio of 5 is assumed when the effluent mixes with the water in the river. This correlates to a dilution ratio of 1/5 = 0.2. While the ODAM

Section 2.3 states that a dilution factor of 10 can be used for drinking water, using a mixing ratio of 5 is appropriate for determining the EAL thresholds.

5.0 Design Inputs

5.1 Data for each of the Service Water Radiation monitors, taken from ODAM Table I-2 and Attachment 2, is presented here:

Table 3 – Service Water Radiation Monitor Design Inputs

Rad Monitor	Equip. ID	Range cps	Efficiency cps/µci/ml	Efficiency Source Document
GSW	RE-4767	$0.1\text{-}10^6$	2.19E+06	Attachment 2
RHRSW/ESW	RE-1997	$0.1\text{-}10^6$	1.51E+06	Attachment 2
RHRSW Dilution Line*	RE-4268	$0.1\text{-}10^6$	2.29E+06	Attachment 2

*RE-4268 was previously known as the RHRSW Rupture Disk

5.2 The isotopic mixture and half-lives used in this calculation are taken or developed from NUREG-1940 Table 1-2 and Table A-4.

Table 4 – Isotopic Mixture and Half-lives

Isotope	BWR Coolant Concentration (Ci/g)	Half-life (hr)	Isotope	BWR Coolant Concentration (Ci/g)	Half-life (hr)
Ag-110m	1.00E-12	6.00E+03	**Na-24**	2.00E-09	1.50E+01
Ba-140	4.00E-10	3.05E+02	**Np-239**	8.00E-09	5.66E+01
Ce-141	3.00E-11	7.80E+02	**P-32**	4.00E-11	5.83E+02
Ce-144*	3.00E-12	6.82E+03	**Rb-89**	5.00E-09	2.54E-01
Co-58	1.00E-10	1.70E+03	**Ru-103**	2.00E-11	9.43E+02
Co-60	2.00E-10	4.61E+04	**Ru-106**	3.00E-12	8.83E+03
Cr-51	3.00E-09	6.65E+02	**Sr-89**	1.00E-10	1.21E+03
Cs-134	3.00E-11	1.81E+04	**Sr-90**	7.00E-12	2.54E+05
Cs-136	2.00E-11	3.14E+02	**Sr-91**	4.00E-09	9.50E+00
Cs-137*	8.00E-11	2.64E+05	**Sr-92**	1.00E-08	2.71E+00
Cs-138	1.00E-08	5.38E-01	**Te-129m**	4.00E-11	8.06E+02
Cu-64	3.00E-09	1.27E+01	**Te-131m**	1.00E-10	3.00E+01
Fe-59	3.00E-11	1.07E+03	**Te-132**	1.00E-11	7.82E+01
I-131	2.20E-09	1.93E+02	**W-187**	3.00E-10	2.39E+01
I-132	2.20E-08	2.29E+00	**Y-91**	4.00E-11	1.40E+03
I-133	1.50E-08	2.08E+01	**Y-92**	6.00E-09	3.55E+00
I-134	4.30E-08	8.76E-01	**Y-93**	4.00E-09	1.01E+01
I-135	2.20E-08	6.60E+00	**Zn-65**	1.00E-10	5.86E+03
Mn-54	3.50E-11	7.51E+03	**Zr-95**	8.00E-12	1.54E+03
Mn-56	2.50E-08	2.57E+00	**Fe-55**	1.00E-09	2.37E+04
Mo-99	2.00E-09	6.60E+01	**H-3**	1.00E-08	1.08E+05
			Ni-63	1.00E-12	8.40E+05

5.3 Dose Coefficient from Water Immersion/ Annual Limit for Intake (ALI)

The dose coefficient for water immersion for each isotope are from Table III.2 of FGR12.

The Annual Limit for Intake (ALI), which represents the number of microcuries that would have to be ingested to cause a dose of 5 rem to an occupationally exposed worker is taken from 10CFR20, Appendix B, Table 1 Column One.

Table 5 – FGR 12 and ALI

Isotope	FGR 12	ALI	Isotope	FGR 12	ALI
	Sv m³ / Bq s	µCi		Sv m³ / Bq s	µCi
Ag-110m	2.94E-16	5.00E+02	**Na-24**	4.73E-16	4.00E+03
Ba-140	1.87E-17	6.00E+02	**Np-239**	1.70E-17	2.00E+03
Ce-141	7.62E-18	2.00E+03	**P-32**	1.90E-19	6.00E+02
Ce-144	1.91E-18	3.00E+02	**Rb-89**	2.30E-16	6.00E+04
Co-58	1.03E-16	2.00E+03	**Ru-103**	4.89E-17	2.00E+03
Co-60	2.74E-16	2.00E+02	**Ru-106**	2.24E-17	2.00E+02
Cr-51	3.30E-18	4.00E+04	**Sr-89**	1.49E-19	5.00E+02
Cs-134	1.64E-16	7.00E+01	**Sr-90**	1.46E-20	4.00E+01
Cs-136	2.31E-16	4.00E+02	**Sr-91**	7.48E-17	2.00E+03
Cs-137	1.49E-20	1.00E+02	**Sr-92**	1.47E-16	3.00E+03
Cs-138	2.62E-16	3.00E+04	**Te-129m**	3.39E-18	5.00E+02
Cu-64	1.98E-17	1.00E+04	**Te-131m**	1.52E-16	6.00E+02
Fe-59	1.29E-16	8.00E+02	**Te-132**	2.28E-17	7.00E+02
I-131	3.98E-17	9.00E+01	**W-187**	4.97E-17	2.00E+03
I-132	2.43E-16	9.00E+03	**Y-91**	5.44E-19	6.00E+02
I-133	6.39E-17	5.00E+02	**Y-92**	2.81E-17	3.00E+03
I-134	2.82E-16	3.00E+04	**Y-93**	1.03E-17	1.00E+03
I-135	1.73E-16	3.00E+03	**Zn-65**	6.29E-17	4.00E+02
Mn-54	8.88E-17	2.00E+03	**Zr-95**	7.82E-17	1.00E+03
Mn-56	1.86E-16	5.00E+03	**Fe-55**	0.00E+00	9.00E+03
Mo-99	1.58E-17	1.00E+03	**H-3**	0.00E+00	8.00E+04
			Ni-63	0.00E+00	9.00E+03

5.4 Dose transfer factors for radionuclides in effluent water for a child through the potable water pathway are taken from ODAM Appendix C, and shown in Table 9.

6.0 Methodology

6.1 General Approach

With a given mixture of radionuclides, the dose received by an individual offsite is a function of the gross activity present in the liquid mixture.

The resultant dose received by an offsite receptor is dependent not only on the gross radioactivity levels of the effluent but also upon the isotopic mixture present in the liquid. This calculation predicts the relative contribution of each radionuclide to the gross radiation monitored by the liquid effluent monitor.

For a liquid release, the only phenomenon affecting the mixture is radioactive decay.

With the mixture known, a given gross output reading (cps) from a liquid effluent radiation monitor can be scaled to determine the concentration of each isotope present in the liquid effluent.

The calculation then uses liquid dilution factors as described in the Offsite Dose Assessment Manual to determine the resultant concentration of radionuclides to which an individual offsite would be exposed.

Dose conversion factors are used to determine the dose (mrem) to an individual offsite due to their exposure to the liquid mixture of radionuclides.

With the given radionuclide mixture and dilution factors understood, an iterative process can be used to relate the liquid effluent monitor reading to a target offsite dose.

Two types of radiation dose are calculated:

- Thyroid CDE or Committed Dose Equivalent is the radiation dose to the thyroid due to an uptake of radioactive material. In this case, the uptake is limited to ingestion of radioactive material present in river water.
- TEDE or Total Effective Dose Equivalent is the summation of the Effective Dose Equivalent (EDE) and the Committed Effective Dose Equivalent (CEDE): TEDE = EDE + CEDE.
- EDE is the dose due to an individual being directly exposed (by submersion) to the radiation present in the liquid release. For this scenario, the individual is not actually immersed in the liquid, but boating above it so a correction factor is applied.
- CEDE is the sum of the CDE for each organ of the body with weighting factors applied for each organ.

6.2 Scenario

The ODAM described dose pathways focus on long term ingestion of radionuclides through various food pathways. This is in sharp contrast to the NEI thresholds which limit the exposure to one hour. To meet the prescribed one hour exposure scenario, the following scenario will be used:

- An adult and a child are fishing from a boat on the Cedar River downstream of the facility. While they are there, a radioactive liquid release from the facility occurs. The release lasts one hour.
- During that time frame each of the individuals ingests 500 milliliters of river water.
- The individuals leave the area one hour after the start of the exposure when they heed the announcement from the ERO siren system.

The pathways thus indicated are drinking water and boating.

6.3 Radioactive Source

The gross radioactive concentration is converted from cps to µCi/ml using the monitor efficiencies from Design Input 5.1. For example, the gross radioactive concentration in the GSW system with an indication of 10000 cps from the GSW Rad. Monitor is calculated as follows:

$$\frac{10000\ \cancel{cps} \times 1\ \mu Ci/ml}{2.19E{+}06\ \cancel{cps}} = 4.57E{-}03\ \mu Ci/ml$$

With the total concentration of the effluent known, given a mixture of isotopes where the relative amount of each isotope is known, the isotopic mixture in the effluent can be determined. The isotopic mixture and half-lives used in this calculation come from Design Input 5.2.

In order to determine the radiation dose to the receptor, the concentrations affecting the receptor must be known.

For Plant Modes 1, 2, and 3, a source decay time of two hours is assumed to account for transit time (Assumption 4.1). For Plant Modes 4 and 5, a source decay time of 36 hours is assumed (Assumption 4.2).

The NUREG-1940 source is decayed using Design Input 5.2. The Ag-110m computation is displayed (indicated as row 7) as an example:

Table 6 – Source Decay

Isotope	NUREG-1940 Concentration Ci/g	Half-life hr	lambda hrs-1	Conc. 2 Hr Decay	Conc. 36 Hr Decay
A	B	C	D	E	F
Ag-110m	1.00E-12	6.00E+03	=(LN(2))/C7	= B7*(EXP(-D7*2))	= B7*(EXP(-D7*36))
Ag-110m	1.00E-12	6.00E+03	1.16E-04	1.00E-12	9.96E-13
Ba-140	4.00E-10	3.05E+02	2.27E-03	3.98E-10	3.69E-10
Ce-141	3.00E-11	7.80E+02	8.89E-04	2.99E-11	2.91E-11
Ce-144*	3.00E-12	6.82E+03	1.02E-04	3.00E-12	2.99E-12
Co-58	1.00E-10	1.70E+03	4.08E-04	9.99E-11	9.85E-11
Co-60	2.00E-10	4.61E+04	1.50E-05	2.00E-10	2.00E-10
Cr-51	3.00E-09	6.65E+02	1.04E-03	2.99E-09	2.89E-09
Cs-134	3.00E-11	1.81E+04	3.84E-05	3.00E-11	3.00E-11
Cs-136	2.00E-11	3.14E+02	2.20E-03	1.99E-11	1.85E-11
Cs-137*	8.00E-11	2.64E+05	2.63E-06	8.00E-11	8.00E-11
Cs-138	1.00E-08	5.38E-01	1.29E+00	7.59E-10	6.95E-29
Cu-64	3.00E-09	1.27E+01	5.46E-02	2.69E-09	4.20E-10
Fe-59	3.00E-11	1.07E+03	6.49E-04	3.00E-11	2.93E-11
I-131	2.20E-09	1.93E+02	3.59E-03	2.18E-09	1.93E-09
I-132	2.20E-08	2.29E+00	3.02E-01	1.20E-08	4.16E-13
I-133	1.50E-08	2.08E+01	3.33E-02	1.40E-08	4.52E-09
I-134	4.30E-08	8.76E-01	7.91E-01	8.83E-09	1.83E-20
I-135	2.20E-08	6.60E+00	1.05E-01	1.78E-08	5.02E-10
Mn-54	3.50E-11	7.51E+03	9.23E-05	3.50E-11	3.49E-11
Mn-56	2.50E-08	2.57E+00	2.70E-01	1.46E-08	1.51E-12
Mo-99	2.00E-09	6.60E+01	1.05E-02	1.96E-09	1.37E-09
Na-24	2.00E-09	1.50E+01	4.62E-02	1.82E-09	3.79E-10
Np-239	8.00E-09	5.66E+01	1.22E-02	7.81E-09	5.15E-09
P-32	4.00E-11	5.83E+02	1.19E-03	3.99E-11	3.83E-11
Rb-89	5.00E-09	2.54E-01	2.72E+00	2.15E-11	1.26E-51
Ru-103	2.00E-11	9.43E+02	7.35E-04	2.00E-11	1.95E-11
Ru-106	3.00E-12	8.83E+03	7.85E-05	3.00E-12	2.99E-12
Sr-89	1.00E-10	1.21E+03	5.72E-04	9.99E-11	9.80E-11
Sr-90	7.00E-12	2.54E+05	2.72E-06	7.00E-12	7.00E-12
Sr-91	4.00E-09	9.50E+00	7.29E-02	3.46E-09	2.90E-10
Sr-92	1.00E-08	2.71E+00	2.56E-01	6.00E-09	1.01E-12
Te-129m	4.00E-11	8.06E+02	8.60E-04	3.99E-11	3.88E-11
Te-131m	1.00E-10	3.00E+01	2.31E-02	9.55E-11	4.35E-11
Te-132	1.00E-11	7.82E+01	8.86E-03	9.82E-12	7.27E-12
W-187	3.00E-10	2.39E+01	2.90E-02	2.83E-10	1.06E-10
Y-91	4.00E-11	1.40E+03	4.94E-04	4.00E-11	3.93E-11
Y-92	6.00E-09	3.55E+00	1.95E-01	4.06E-09	5.34E-12
Y-93	4.00E-09	1.01E+01	6.86E-02	3.49E-09	3.38E-10
Zn-65	1.00E-10	5.86E+03	1.18E-04	1.00E-10	9.96E-11
Zr-95	8.00E-12	1.54E+03	4.51E-04	7.99E-12	7.87E-12
Fe-55	1.00E-09	2.37E+04	2.93E-05	1.00E-09	9.99E-10
H-3	1.00E-08	1.08E+05	6.40E-06	1.00E-08	1.00E-08
Ni-63	1.00E-12	8.40E+05	8.25E-07	1.00E-12	1.00E-12

6.4 TEDE Dose

To determine the TEDE dose for a one hour exposure, two components are considered.

1. Direct exposure to the radioactive water present in the river is considered, commonly described as Effective Dose Equivalent (**EDE**).
2. Committed Effective Dose Equivalent (**CEDE**) is considered, which is the dose commitment due to the ingestion or inhalation of a mixture of radioactive material.

6.4.1 EDE

Immersion dose is calculated with guidance provided in Federal Guidance Report 12 (FGR12).

With the isotopic concentration at the receptor known (Table 6), the dose (mrem) at the receptor can be calculated:

$$Dose = \sum_i (x_{ir} * hT_i)$$

Where

$x_{ir} =$ concentration of radionuclide ***i*** present in the water at the receptor (µCi/ ml)

$i =$ each isotope present in the liquid release

$hT_i =$ factor from FGR12 for converting the liquid concentration to effective dose equivalent from Design Input 5.3 (mrem ml / sec µCi)

The dose coefficients in Design Input 5.3 from FGR12 relate the radioactive concentration of a liquid to the dose received by a person who is immersed in the liquid. FGR 12 also provides this statement about the relationship between immersion dose and dose received while boating:

> **Exposure during boating activities**
>
> The dose coefficients for immersion in contaminated water in Table III.2 assume immersion in an infinite pool and, thus, are appropriate for exposure while swimming. External exposure to contaminated water can also occur during boating activities. For photon exposure, a dose-reduction factor of 0.5 during boating activities is a reasonable value that is unlikely to underestimate external dose equivalents.

6.4.2 CEDE

To calculate the Committed Effective Dose Equivalent due to the consumption of contaminated water, the ALI values from Design Input 5.3 are used. The first column of the table lists the ALI for each isotope. The ALI value is the Annual Limit for Intake and represents the number of microcuries that would have to be ingested to cause a dose of 5 rem to an occupationally exposed worker.

Following the conversion of the ALI values to units of "*mrem per μCi*" the following equation can be used to determine radiation dose due to ingestion of a liquid mixture of radioisotopes.

$$Dose = \sum_i (x_{ir} * v * hE_i)$$

Where

$x_{ir} =$ concentration of the radionuclide ***i*** present in the water at the receptor (μCi/ml)

$i =$ each isotope present in the gaseous release

$hE_i =$ factor converting the gas concentration to effective dose equivalent (mrem/μCi)

$v =$ volume of water ingested (ml)

10CFR20 includes a statement regarding the use of Table 1 Column One values for members of the public:

> ...a factor of 2 to adjust the occupational values (derived for adults) so that they are applicable to other age groups.

Spreadsheets are used in section 7.1 to calculate EDE and CEDE from all of the isotopes in the mixture.

6.5 Organ dose

Methods to calculate Organ Dose are taken from ODAM section 2.6 titled "Accumulated Personal Maximum Dose". The guidance is provided here:

$$\Delta D_{ank} = 3.785 \bullet x10^{-3} \sum_i C_{ik} \bullet \Delta T_k \sum_e \frac{F1_k}{F2_{ek}} \bullet A_{eani}$$

$$D_{an} = \sum_k \Delta D_{ank}$$

where

ΔD_{ank} = the dose commitment (mrem) to organ *n* of age group *a* due to the isotopes identified in analysis *k*, where

the analyses are those required by Table 7.1-2. Thus the contribution to the dose from gamma emitters become available on a batch basis for batch releases and on a weekly basis for continuous releases. Similarly the contributions from H-3 is available on a monthly basis and the contributions from Fe-55, Sr-89, and Sr-90 become available on a quarterly basis.

D_{an} = the dose commitment during the quarter-to-date to organ *n*, including whole body, of the maximally exposed person in age group *a* (mrem)

A_{eani} = transfer factor relating a unit release of radionuclide *i* (Ci) in a unit stream flow (gal/min) to dose commitment to organ *n*, or whole body, of an exposed person in age group *a* $\left[\frac{mrem\ gal}{Ci\ \min}\right]$ via environmental pathway *e*.

C_{ik} = the concentration of radionuclide *i* in the undiluted liquid waste represented by sample *k* to be discharged (µCi/mL).

Δt_k = duration of radioactive release represented by sample *k* which occurs within time boundaries TB and TE and during which concentration C_{ik} and flows $F1_k$ and $F2_k$ exist. (min.)

$3.785 \bullet 10^{-3}$ = conversion constant (3785 mL/gal • 10^{-6} Ci/µCi)

$F1_k$ = flow in the radioactive waste release line (gal/min)* represented by sample *k*.

$F2_{ke}$ = flow into which radioactive release represented by sample *k* is mixed in the river at the point of exposure or withdrawal of water for use (same units as $F1_k$)*

For this calculation,

- Δ_{tk} is set to 60 minutes
- $F_{1k/2ke}$ is 0.2 per Assumption 4.3.

- A_{eani} values are taken from ODAM Appendix C (Design Input 5.4)
- Based on the scenario, only the child thyroid organ is considered, as this bounds the adult, and is used in Iowa as the basis for Protective Action Guidelines.
- Based on the scenario, only the drinking water pathway is considered.

Spreadsheets are used in section 7.2 to calculate CDE-Thyroid from all of the isotopes in the mixture.

7.0 Calculation

7.1 TEDE Dose

A Microsoft Excel spreadsheet uses an iterative process to determine the cps output readings from each of the three monitors that correspond to a TEDE Dose of 10 mrem.

Sections of the spreadsheet are presented here. This is the spreadsheet for the RHRSW/ESW Monitor.

Ingestion Dose:	8.24	mrem
+ Boating Dose:	1.79	mrem
Resultant Total mRem:	10.0	mrem

Variables

Monitor: RHRSW/ESW **RM1997**

FGR12 Units Conversion Factor:	3.70E+15		FGR12 Boating/Immersion Reduction Factor:	0.50
Monitor cps:	16,000	cps		
Monitor Efficiency:	1.51E+6	cps / uCi/ml	10CFR20 Ingestion Age Consideration Factor:	2
			Decay Hrs:	2
Volume Consumed:	500	mL	Dilution Factor:	0.20

Resultant River Gamma Concentration for the Given CPS Reading :

16000	~~cps~~		µCi/ml	0.20	=	2.12E-03	µCi/ml
		1.51E+06	~~cps~~				

The mrem values seen above are calculated in the spreadsheet on the following pages. As can be seen above, the dilution factor **(0.20)** is included in the efficiency equation to account for the fact that the concentration in the river will be only 20% of the concentration seen by the effluent radiation monitor per Assumption 4.3.

7.1.1 Boating Dose

The concentrations for the individual isotopes are scaled to the gross concentration determined above. In this case, the value is 2.12E-03 µCi/ml. The value of 2.12E-03 is calculated based on the monitor cps reading entered by the spreadsheet user. Through an iterative process, the user enters the monitor cps necessary to determine the desired resultant total dose (in mrem).

The dose coefficient for water immersion for each isotope in column B in Table 7 is taken from Design Input 5.3. FGR 12 displays dose factors in the SI units of SV m^3/ bq sec. Traditional units of mrem mL/sec µCi sec are desired.

FGR 12:

1	~~SV~~	~~m3~~	1E+05	mRem	1	~~bq~~	1E+06	mL		~~Ci~~	=	3.70E+15	mRem	mL
	~~bq~~	sec		~~SV~~	2.7E-11	~~Ci~~		~~m3~~	1E+06	µCi			sec	µCi

The conversion factor from SV m^3/bq sec to mrem cm^3/µCi sec is 3.70E+15.

The decayed mixture from column E is taken from Table 6. Note that the concentration values present in the (starting) mixture do not affect the result. It is the ratios of the isotopes to the gross concentration (Section 7.1 - Column G) in the mixture that are needed. Per Section 6.4.1, a 0.5 dose-reduction factor is applied in Column I. For illustrative purposes, for the isotope AG-110m, the cell formulas are displayed.

Table 7 – Boating Dose

	A	B	C	D	E	F	G	H	I
1	Nuclide	FGR 12: SV m3 bq sec	FGR12 Units Conv.	mrem mL µCi sec	Decayed Mix µCi mL	Fraction	River µCi/mL	Immersion mrem Sec	Boating mrem Hr
2	Ag-110m	2.94E-16	3.70E+15	1.09E+0	Table 6	=E2/ 1.17E-7	=F2*2.12E-3	=D2*G2	=H2*3600 * 0.5
	Ag-110m	2.94E-16	3.70E+15	1.09E+0	1.00E-12	0.00%	1.81E-8	1.97E-8	3.54E-5
	Ba-140	1.87E-17	3.70E+15	6.92E-2	3.98E-10	0.34%	7.20E-6	4.98E-7	8.97E-4
	Ce-141	7.62E-18	3.70E+15	2.82E-2	2.99E-11	0.03%	5.41E-7	1.53E-8	2.75E-5
	Ce-144*	1.91E-18	3.70E+15	7.07E-3	3.00E-12	0.00%	5.42E-8	3.83E-10	6.90E-7
	Co-58	1.03E-16	3.70E+15	3.81E-1	9.99E-11	0.09%	1.81E-6	6.88E-7	1.24E-3
	Co-60	2.74E-16	3.70E+15	1.01E+0	2.00E-10	0.17%	3.62E-6	3.67E-6	6.60E-3
	Cr-51	3.30E-18	3.70E+15	1.22E-2	2.99E-09	2.55%	5.41E-5	6.61E-7	1.19E-3
	Cs-134	1.64E-16	3.70E+15	6.07E-1	3.00E-11	0.03%	5.42E-7	3.29E-7	5.92E-4
	Cs-136	2.31E-16	3.70E+15	8.55E-1	1.99E-11	0.02%	3.60E-7	3.08E-7	5.54E-4
	Cs-137*	1.49E-20	3.70E+15	5.51E-5	8.00E-11	0.07%	1.45E-6	7.97E-11	1.44E-7
	Cs-138	2.62E-16	3.70E+15	9.69E-1	7.59E-10	0.65%	1.37E-5	1.33E-5	2.39E-2
	Cu-64	1.98E-17	3.70E+15	7.33E-2	2.69E-09	2.29%	4.86E-5	3.56E-6	6.41E-3
	Fe-59	1.29E-16	3.70E+15	4.77E-1	3.00E-11	0.03%	5.42E-7	2.59E-7	4.65E-4
	I-131	3.98E-17	3.70E+15	1.47E-1	2.18E-09	1.86%	3.95E-5	5.82E-6	1.05E-2
	I-132	2.43E-16	3.70E+15	8.99E-1	1.20E-08	10.26%	2.17E-4	1.95E-4	3.52E-1
	I-133	6.39E-17	3.70E+15	2.36E-1	1.40E-08	11.97%	2.54E-4	6.00E-5	1.08E-1
	I-134	2.82E-16	3.70E+15	1.04E+0	8.83E-09	7.54%	1.60E-4	1.67E-4	3.00E-1
	I-135	1.73E-16	3.70E+15	6.40E-1	1.78E-08	15.21%	3.22E-4	2.06E-4	3.71E-1
	Mn-54	8.88E-17	3.70E+15	3.29E-1	3.50E-11	0.03%	6.33E-7	2.08E-7	3.74E-4

A	B	C	D	E	F	G	H	I
Nuclide	FGR 12: SV m3 / bq sec	FGR12 Units Conv.	mrem mL / µCi sec	Decayed Mix µCi / mL	Fraction	River µCi/mL	Immersion mrem / Sec	Boating mrem / Hr
Mn-56	1.86E-16	3.70E+15	6.88E-1	1.46E-08	12.43%	2.63E-4	1.81E-4	3.26E-1
Mo-99	1.58E-17	3.70E+15	5.85E-2	1.96E-09	1.67%	3.54E-5	2.07E-6	3.73E-3
Na-24	4.73E-16	3.70E+15	1.75E+0	1.82E-09	1.56%	3.30E-5	5.77E-5	1.04E-1
Np-239	1.70E-17	3.70E+15	6.29E-2	7.81E-09	6.66%	1.41E-4	8.88E-6	1.60E-2
P-32	1.90E-19	3.70E+15	7.03E-4	3.99E-11	0.03%	7.22E-7	5.07E-10	9.13E-7
Rb-89	2.30E-16	3.70E+15	8.51E-1	2.15E-11	0.02%	3.89E-7	3.31E-7	5.95E-4
Ru-103	4.89E-17	3.70E+15	1.81E-1	2.00E-11	0.02%	3.61E-7	6.53E-8	1.18E-4
Ru-106	2.24E-17	3.70E+15	8.29E-2	3.00E-12	0.00%	5.42E-8	4.49E-9	8.09E-6
Sr-89	1.49E-19	3.70E+15	5.51E-4	9.99E-11	0.09%	1.81E-6	9.96E-10	1.79E-6
Sr-90	1.46E-20	3.70E+15	5.40E-5	7.00E-12	0.01%	1.27E-7	6.84E-12	1.23E-8
Sr-91	7.48E-17	3.70E+15	2.77E-1	3.46E-09	2.95%	6.25E-5	1.73E-5	3.11E-2
Sr-92	1.47E-16	3.70E+15	5.44E-1	6.00E-09	5.12%	1.08E-4	5.90E-5	1.06E-1
Te-129m	3.39E-18	3.70E+15	1.25E-2	3.99E-11	0.03%	7.22E-7	9.06E-9	1.63E-5
Te-131m	1.52E-16	3.70E+15	5.62E-1	9.55E-11	0.08%	1.73E-6	9.71E-7	1.75E-3
Te-132	2.28E-17	3.70E+15	8.44E-2	9.82E-12	0.01%	1.78E-7	1.50E-8	2.70E-5
W-187	4.97E-17	3.70E+15	1.84E-1	2.83E-10	0.24%	5.12E-6	9.41E-7	1.69E-3
Y-91	5.44E-19	3.70E+15	2.01E-3	4.00E-11	0.03%	7.23E-7	1.45E-9	2.62E-6
Y-92	2.81E-17	3.70E+15	1.04E-1	4.06E-09	3.46%	7.34E-5	7.63E-6	1.37E-2
Y-93	1.03E-17	3.70E+15	3.81E-2	3.49E-09	2.98%	6.31E-5	2.40E-6	4.33E-3
Zn-65	6.29E-17	3.70E+15	2.33E-1	1.00E-10	0.09%	1.81E-6	4.21E-7	7.57E-4
Zr-95	7.82E-17	3.70E+15	2.89E-1	7.99E-12	0.01%	1.45E-7	4.18E-8	7.53E-5
Fe-55	0.00E+00	3.70E+15	0.00E+0	1.00E-09	0.85%	1.81E-5	0.00E+0	0.00E+0
H-3	0.00E+00	3.70E+15	0.00E+0	1.00E-08	8.53%	1.81E-4	0.00E+0	0.00E+0
Ni-63	0.00E+00	3.70E+15	0.00E+0	1.00E-12	0.00%	1.81E-8	0.00E+0	0.00E+0
				1.17E-7	100.00%	2.12E-3	9.97E-4	**1.79**
						2.12E-3		mrem

7.1.2 Ingestion Dose

Ingestion dose is calculated in the section of the spreadsheet seen below. ALIs (column D) from Design Input 5.3 are converted to mrem/µCi factors (column E) as shown in the example below for Co-60 which has an ALI of 200 µCi:

$$\frac{5\ \cancel{\text{rem}}}{2E+02\ \mu\text{Ci}} \left| \frac{1000\ \text{mrem}}{1\ \cancel{\text{Rem}}} \right. = \frac{25\ \text{mRem}}{\mu\text{Ci}}$$

The resultant dose (in mrem) caused by ingesting 500 ml of the liquid is calculated per Section 6.4.2. The decayed mixture from column F is taken from Table 6.

An **example** demonstrating dose caused by drinking 500 ml of water containing cobalt-60 with a concentration of <u>0.001 µCi/ml</u>:

$$\frac{25\ \text{mrem}}{\cancel{\mu Ci}} \times \frac{0.001\ \cancel{\mu Ci}}{\cancel{ml}} \times 500\ \cancel{ml} = 12.5\ \text{mrem}$$

For an occupationally exposed worker, drinking those 500 milliliters of water contaminated with Co-60 would result in a dose of 12.5 mrem. Per Section 6.4.2, this is multiplied by 2, generating 25 mrem.

Note that there are three hard-to-detect isotopes (HTDs) present in the mixture: Fe-55, H-3 and Ni-63. Because they do not emit gamma rays, they are not detected by the service water radiation monitor. Therefore, they are effectively removed from the gross gamma calculation calibrating the monitor response (Columns H, I, and J). The HTDs are then scaled back into the calculation of the applied dose.

For illustrative purposes, for the isotope Ag-110m, the cell formulas are displayed. The decayed mixture from column F is taken from Table 6. Column M contains the multiplier for the 500 ml volume consumed and the 2x multiplier factor for members of the public.

Table 8 – Ingestion Dose

B	C	D	E	F	G	H	I	J	K	L	M
1		10CFR20 ALI	mrem / µCi	Mix Ci / ml	Mix Fraction	Mix Gamma Ci / ml	Mix Gamma Fraction	River Gamma µCi / ml	River µCi / ml	River Fraction	mrem
2	Ag-110m	5.00E+02	=1/ (D2/5000)	Table 6	=F2*1.17E-7	=F2	=H2/ 1.06E-7	=I2*2.12E-3	=J2	=K2/ 2.32E-3	=K2*E2 *2*500
	Ag-110m	5.00E+02	1.00E+1	1.00E-12	0.0%	1.00E-12	0.0%	1.99E-8	1.99E-8	0.0%	1.99E-04
	Ba-140	6.00E+02	8.33E+0	3.98E-10	0.3%	3.98E-10	0.4%	7.95E-6	7.95E-6	0.3%	6.62E-02
	Ce-141	2.00E+03	2.50E+0	2.99E-11	0.0%	2.99E-11	0.0%	5.98E-7	5.98E-7	0.0%	1.49E-03
	Ce-144*	3.00E+02	1.67E+1	3.00E-12	0.0%	3.00E-12	0.0%	5.98E-8	5.98E-8	0.0%	9.97E-04
	Co-58	1.00E+03	5.00E+0	9.99E-11	0.1%	9.99E-11	0.1%	1.99E-6	1.99E-6	0.1%	9.97E-03
	Co-60	2.00E+02	2.50E+1	2.00E-10	0.2%	2.00E-10	0.2%	3.99E-6	3.99E-6	0.2%	9.98E-02
	Cr-51	4.00E+04	1.25E-1	2.99E-09	2.6%	2.99E-9	2.8%	5.97E-5	5.97E-5	2.6%	7.47E-03
	Cs-134	7.00E+01	7.14E+1	3.00E-11	0.0%	3.00E-11	0.0%	5.99E-7	5.99E-7	0.0%	4.28E-02
	Cs-136	4.00E+02	1.25E+1	1.99E-11	0.0%	1.99E-11	0.0%	3.97E-7	3.97E-7	0.0%	4.97E-03
	Cs-137*	1.00E+02	5.00E+1	8.00E-11	0.1%	8.00E-11	0.1%	1.60E-6	1.60E-6	0.1%	7.98E-02
	Cs-138	3.00E+04	1.67E-1	7.59E-10	0.6%	7.59E-10	0.7%	1.51E-5	1.51E-5	0.7%	2.52E-03
	Cu-64	1.00E+04	5.00E-1	2.69E-09	2.3%	2.69E-9	2.5%	5.37E-5	5.37E-5	2.3%	2.68E-02
	Fe-59	8.00E+02	6.25E+0	3.00E-11	0.0%	3.00E-11	0.0%	5.98E-7	5.98E-7	0.0%	3.74E-03
	I-131	9.00E+01	5.56E+1	2.18E-09	1.9%	2.18E-9	2.1%	4.36E-5	4.36E-5	1.9%	2.42E+00
	I-132	9.00E+03	5.56E-1	1.20E-08	10.3%	1.20E-8	11.3%	2.40E-4	2.40E-4	10.3%	1.33E-01
	I-133	5.00E+02	1.00E+1	1.40E-08	12.0%	1.40E-8	13.2%	2.80E-4	2.80E-4	12.1%	2.80E+00
	I-134	3.00E+04	1.67E-1	8.83E-09	7.5%	8.83E-9	8.3%	1.76E-4	1.76E-4	7.6%	2.94E-02
	I-135	3.00E+03	1.67E+0	1.78E-08	15.2%	1.78E-8	16.8%	3.56E-4	3.56E-4	15.3%	5.93E-01
	Mn-54	2.00E+03	2.50E+0	3.50E-11	0.0%	3.50E-11	0.0%	6.98E-7	6.98E-7	0.0%	1.75E-03
	Mn-56	5.00E+03	1.00E+0	1.46E-08	12.4%	1.46E-8	13.7%	2.91E-4	2.91E-4	12.5%	2.91E-01
	Mo-99	1.00E+03	5.00E+0	1.96E-09	1.7%	1.96E-9	1.8%	3.91E-5	3.91E-5	1.7%	1.95E-01
	Na-24	4.00E+03	1.25E+0	1.82E-09	1.6%	1.82E-9	1.7%	3.64E-5	3.64E-5	1.6%	4.55E-02
	Np-239	2.00E+03	2.50E+0	7.81E-09	6.7%	7.81E-9	7.4%	1.56E-4	1.56E-4	6.7%	3.89E-01
	P-32	6.00E+02	8.33E+0	3.99E-11	0.0%	3.99E-11	0.0%	7.96E-7	7.96E-7	0.0%	6.64E-03
	Rb-89	6.00E+04	8.33E-2	2.15E-11	0.0%	2.15E-11	0.0%	4.29E-7	4.29E-7	0.0%	3.57E-05

B	C	D	E	F	G	H	I	J	K	L	M
1		10CFR20 ALI	mrem / µCi	Mix Ci / ml	Mix Fraction	Mix Gamma Ci / ml	Mix Gamma Fraction	River Gamma µCi / ml	River µCi / ml	River Fraction	mrem
	Ru-103	2.00E+03	2.50E+0	2.00E-11	0.0%	2.00E-11	0.0%	3.98E-7	3.98E-7	0.0%	9.96E-04
	Ru-106	2.00E+02	2.50E+1	3.00E-12	0.0%	3.00E-12	0.0%	5.99E-8	5.99E-8	0.0%	1.50E-03
	Sr-89	5.00E+02	1.00E+1	9.99E-11	0.1%	9.99E-11	0.1%	1.99E-6	1.99E-6	0.1%	1.99E-02
	Sr-90	4.00E+01	1.25E+2	7.00E-12	0.0%	7.00E-12	0.0%	1.40E-7	1.40E-7	0.0%	1.75E-02
	Sr-91	2.00E+03	2.50E+0	3.46E-09	2.9%	3.46E-9	3.3%	6.90E-5	6.90E-5	3.0%	1.72E-01
	Sr-92	3.00E+03	1.67E+0	6.00E-09	5.1%	6.00E-9	5.6%	1.20E-4	1.20E-4	5.2%	1.99E-01
	Te-129m	5.00E+02	1.00E+1	3.99E-11	0.0%	3.99E-11	0.0%	7.97E-7	7.97E-7	0.0%	7.97E-03
	Te-131m	6.00E+02	8.33E+0	9.55E-11	0.1%	9.55E-11	0.1%	1.91E-6	1.91E-6	0.1%	1.59E-02
	Te-132	7.00E+02	7.14E+0	9.82E-12	0.0%	9.82E-12	0.0%	1.96E-7	1.96E-7	0.0%	1.40E-03
	W-187	2.00E+03	2.50E+0	2.83E-10	0.2%	2.83E-10	0.3%	5.65E-6	5.65E-6	0.2%	1.41E-02
	Y-91	6.00E+02	8.33E+0	4.00E-11	0.0%	4.00E-11	0.0%	7.97E-7	7.97E-7	0.0%	6.64E-03
	Y-92	3.00E+03	1.67E+0	4.06E-09	3.5%	4.06E-9	3.8%	8.10E-5	8.10E-5	3.5%	1.35E-01
	Y-93	1.00E+03	5.00E+0	3.49E-09	3.0%	3.49E-9	3.3%	6.96E-5	6.96E-5	3.0%	3.48E-01
	Zn-65	4.00E+02	1.25E+1	1.00E-10	0.1%	1.00E-10	0.1%	1.99E-6	1.99E-6	0.1%	2.49E-02
	Zr-95	1.00E+03	5.00E+0	7.99E-12	0.0%	7.99E-12	0.0%	1.59E-7	1.59E-7	0.0%	7.97E-04
	Fe-55	9.00E+03	5.56E-1	1.00E-09	0.9%				1.81E-5	0.8%	1.00E-02
	H-3	8.00E+04	6.25E-2	1.00E-08	8.5%				1.81E-4	7.8%	1.13E-02
	Ni-63	9.00E+03	5.56E-1	1.00E-12	0.0%				1.81E-8	0.0%	1.00E-05
				1.17E-7	100%	1.06E-07	100.0%	2.12E-3 **2.12E-3**	2.32E-3	100.0%	**8.24 mrem**

The initial HTD value is scaled to the total value of the gammas present. Using those ratios, the new HTD concentrations are determined by multiplying the ratios by the revised sum of the gamma emitters.

Hard-to-Detect Determination

HTD	Conc1 Ci / ml	∑gammas Ci / ml	Ratio	HTD	∑gammas µCi / ml	Conc2 µCi / ml
Fe-55	1.00E-9	1.17E-07	8.53E-3	Fe-55	2.12E-3	1.81E-5
H-3	1.00E-8	1.17E-07	8.53E-2	H-3	2.12E-3	1.81E-4
Ni-63	1.00E-12	1.17E-07	8.53E-6	Ni-63	2.12E-3	1.81E-8

7.2 Organ Dose

The organ dose calculation is similar to the TEDE calculation above in that it uses a spreadsheet to determine the monitor cps reading necessary to reach the EAL threshold. The calculation is also similar in that it uses liquid concentrations and dose conversion factors to determine dose. The first part of the spreadsheet is presented here showing the gross concentration developed from section 6.5.

Child Thyroid: 50.0 mRem

Variables

Monitor: RHRSW/ESW RM1997

Decay Hours	2	ODAM Conversion Factor:	3.785E-03
Monitor CPS:	33,800 cps		
Monitor Efficiency:	1.51E+6 cps/uCi/ml	Exposure Time (Mins.):	60
Dilution Factor:	0.20	Combining Factor:	2.271E-01

Resultant Gamma Concentration for the Given CPS Reading :

33800	~~cps~~		uCi/ml	0.20	=	4.48E-03	uCi/ml
		1.51E+06	~~cps~~				

The dose value seen above is calculated in the spreadsheet on the following pages. As can be seen above, the dilution factor **(0.20)** is included in the efficiency equation to account for the fact that the concentration in the river will be only 20% of the concentration seen by the effluent rad monitor per Assumption 4.3.

The concentrations for the individual isotopes are scaled to the gross concentration determined above. In this case, the value is 4.48E-03 µCi/ml. This value is calculated based on the monitor cps reading entered by the spreadsheet user. Through an iterative process, the user enters the monitor cps necessary to determine the desired resultant total dose (in mrem).

Resultant river concentrations are presented in the section of the spreadsheet seen on the next page.

In this spreadsheet, all of the isotopes present in Appendix C of the ODAM (Design Input 5.4) are included. In many instances, there is no corresponding isotope available from the NUREG-1940 reference. The entire list was included to simplify the spreadsheet calculation.

The decayed mixture is taken from Table 6. Note that the concentration values present in the (starting) mixture do not affect the result. It is the ratios of the isotopes to the gross concentration in the mixture that are needed. These fractions are calculated in the same way as section 7.1. As in section 7.1.2, HTDs present in the mixture (Fe-55, H-3, and Ni-63) are removed from the gross gamma calculation and then scaled back into the calculation of the applied dose (see below for scaling).

The ODAM Appendix C dose transfer factors for the thyroid (Design Input 5.4) are displayed on the second to last column from the right.

Using ODAM calculation methods described in section 6.5, the H-3 dose component of the child thyroid pathway is calculated individually here as an example:

3.82E-04 µCi/mL * 2.97E+01 mrem gal/ Ci min * 3.785E-03 mL Ci/gal µCi * 60 min = 2.58E-03 mrem

Where

3.785E-03 is the conversion factor from the ODAM.

2.97E+01 is the H-3 dose transfer factor for the thyroid (Design Input 5.4).

3.82E-04 is the concentration of H-3 in the river corresponding to the monitor reading developed from Table 6 (see below for scaling).

AND the dispersion term (0.20) is not included here because it has already been included above in the concentration calculation.

Table 9 – Thyroid Dose

ODAM Isotopes	Decayed Mix Ci/gm	Mix Fraction	Mix Gamma Ci/gm	Mix Gamma Fraction	River Gamma µCi/ml	River Water Gamma Fraction	River µCi/ml	Dose Transfer Factor Thyroid (mrem gal)/(Ci min)	Thyroid Dose mrem
H 3	1.00E-08	8.532%		0.000%		0.00%	3.82E-4	2.97E+1	2.58E-3
C 14		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	6.19E+2	0.00E+0
NA 24	1.82E-09	1.556%	1.82E-09	1.717%	7.69E-05	1.72%	7.69E-5	4.91E+2	8.57E-3
P 32	3.99E-11	0.034%	3.99E-11	0.038%	1.68E-06	0.04%	1.68E-6	0.00E+0	0.00E+0
CR 51	2.99E-09	2.554%	2.99E-09	2.819%	1.26E-04	2.82%	1.26E-4	1.23E+0	3.52E-5
MN 54	3.50E-11	0.030%	3.50E-11	0.033%	1.48E-06	0.03%	1.48E-6	0.00E+0	0.00E+0
MN 56	1.46E-08	12.432%	1.46E-08	13.720%	6.14E-04	13.72%	6.14E-4	0.00E+0	0.00E+0
FE 55	1.00E-09	0.853%		0.000%		0.00%	3.82E-5	0.00E+0	0.00E+0
FE 59	3.00E-11	0.026%	3.00E-11	0.028%	1.26E-06	0.03%	1.26E-6	0.00E+0	0.00E+0
CO 58	9.99E-11	0.085%	9.99E-11	0.094%	4.21E-06	0.09%	4.21E-6	0.00E+0	0.00E+0
CO 60	2.00E-10	0.171%	2.00E-10	0.188%	8.43E-06	0.19%	8.43E-6	0.00E+0	0.00E+0
NI 63	1.00E-12	0.001%		0.000%		0.00%	3.82E-8	0.00E+0	0.00E+0
NI 65		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
CU 64	2.69E-09	2.295%	2.69E-09	2.532%	1.13E-04	2.53%	1.13E-4	0.00E+0	0.00E+0
ZN 65	1.00E-10	0.085%	1.00E-10	0.094%	4.21E-06	0.09%	4.21E-6	0.00E+0	0.00E+0
ZN 69		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
BR 83		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
BR 84		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
BR 85		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
RB 86		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
RB 88		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
RB 89	2.15E-11	0.018%	2.15E-11	0.020%	9.06E-07	0.02%	9.06E-7	0.00E+0	0.00E+0
SR 89	9.99E-11	0.085%	9.99E-11	0.094%	4.21E-06	0.09%	4.21E-6	0.00E+0	0.00E+0
SR 90	7.00E-12	0.006%	7.00E-12	0.007%	2.95E-07	0.01%	2.95E-7	0.00E+0	0.00E+0
SR 91	3.46E-09	2.950%	3.46E-09	3.255%	1.46E-04	3.26%	1.46E-4	0.00E+0	0.00E+0
SR 92	6.00E-09	5.117%	6.00E-09	5.647%	2.53E-04	5.65%	2.53E-4	0.00E+0	0.00E+0
Y 90		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
Y 91M		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
Y 91	4.00E-11	0.034%	4.00E-11	0.038%	1.68E-06	0.04%	1.68E-6	0.00E+0	0.00E+0

ODAM Isotopes	Decayed Mix Ci/gm	Mix Fraction	Mix Gamma Ci/gm	Mix Gamma Fraction	River Gamma µCi/ml	River Water Gamma Fraction	River µCi/ml	Dose Transfer Factor Thyroid (mrem gal)/(Ci min)	Thyroid Dose mrem
Y 92	4.06E-09	3.465%	4.06E-09	3.824%	1.71E-04	3.82%	1.71E-4	0.00E+0	0.00E+0
Y 93	3.49E-09	2.975%	3.49E-09	3.283%	1.47E-04	3.28%	1.47E-4	0.00E+0	0.00E+0
ZR 95	7.99E-12	0.007%	7.99E-12	0.008%	3.37E-07	0.01%	3.37E-7	0.00E+0	0.00E+0
ZR 97		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
NB 95		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
MO 99	1.96E-09	1.671%	1.96E-09	1.844%	8.26E-05	1.84%	8.26E-5	0.00E+0	0.00E+0
TC 99M		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
TC101		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
RU103	2.00E-11	0.017%	2.00E-11	0.019%	8.42E-07	0.02%	8.42E-7	0.00E+0	0.00E+0
RU105		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
RU106	3.00E-12	0.003%	3.00E-12	0.003%	1.26E-07	0.00%	1.26E-7	0.00E+0	0.00E+0
AG110M	1.00E-12	0.001%	1.00E-12	0.001%	4.21E-08	0.00%	4.21E-8	0.00E+0	0.00E+0
TE125M		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	8.09E+2	0.00E+0
TE127M		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	1.76E+3	0.00E+0
TE127		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	1.41E+1	0.00E+0
TE129M	3.99E-11	0.034%	3.99E-11	0.038%	1.68E-06	0.04%	1.68E-6	3.94E+3	1.51E-3
TE129		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	1.44E-5	0.00E+0
TE131M	9.55E-11	0.081%	9.55E-11	0.090%	4.02E-06	0.09%	4.02E-6	7.52E+2	6.87E-4
TE131		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
TE132	9.82E-12	0.008%	9.82E-12	0.009%	4.14E-07	0.01%	4.14E-7	1.35E+3	1.27E-4
I 130		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	4.32E+4	0.00E+0
I 131	2.18E-09	1.864%	2.18E-09	2.057%	9.21E-05	2.06%	9.21E-5	1.34E+6	2.80E+1
I 132	1.20E-08	10.258%	1.20E-08	11.321%	5.07E-04	11.32%	5.07E-4	1.26E+1	1.45E-3
I 133	1.40E-08	11.973%	1.40E-08	13.213%	5.92E-04	13.21%	5.92E-4	1.56E+5	2.10E+1
I 134	8.83E-09	7.538%	8.83E-09	8.318%	3.72E-04	8.32%	3.72E-4	2.55E-5	2.16E-9
I 135	1.78E-08	15.214%	1.78E-08	16.790%	7.52E-04	16.79%	7.52E-4	5.73E+3	9.78E-1
CS134	3.00E-11	0.026%	3.00E-11	0.028%	1.26E-06	0.03%	1.26E-6	0.00E+0	0.00E+0
CS136	1.99E-11	0.017%	1.99E-11	0.019%	8.39E-07	0.02%	8.39E-7	0.00E+0	0.00E+0
CS137	8.00E-11	0.068%	8.00E-11	0.075%	3.37E-06	0.08%	3.37E-6	0.00E+0	0.00E+0
CS138	7.59E-10	0.647%	7.59E-10	0.714%	3.20E-05	0.71%	3.20E-5	0.00E+0	0.00E+0
BA139		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
BA140	3.98E-10	0.340%	3.98E-10	0.375%	1.68E-05	0.37%	1.68E-5	0.00E+0	0.00E+0
BA141		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
BA142		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
LA140		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
LA142		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
CE141	2.99E-11	0.026%	2.99E-11	0.028%	1.26E-06	0.03%	1.26E-6	0.00E+0	0.00E+0
CE143		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
CE144	3.00E-12	0.003%	3.00E-12	0.003%	1.26E-07	0.00%	1.26E-7	0.00E+0	0.00E+0
PR143		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
PR144		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
ND147		0.000%	0.00E+00	0.000%	0.00E+00	0.00%	0.00E+0	0.00E+0	0.00E+0
W 187	2.83E-10	0.242%	2.83E-10	0.267%	1.19E-05	0.27%	1.19E-5	0.00E+0	0.00E+0
NP239	7.81E-09	6.660%	7.81E-09	7.350%	3.29E-04	7.35%	3.29E-4	0.00E+0	0.00E+0
	1.17E-07	100.0%	1.06E-07	100%	4.48E-03 **4.48E-3**	100.00%	4.90E-3	1.55E+6	5.00E+1

The HTD isotopes displayed in this spreadsheet were scaled into the results as they were in the TEDE spreadsheet.

Hard to Detect Determination						
HTD	Conc1	∑gammas	Ratio	HTD	∑gammas	Conc2
Fe-55	1.00E-9	1.17E-07	8.53E-3	Fe-55	4.48E-3	3.82E-5
H-3	1.00E-8	1.17E-07	8.53E-2	H-3	4.48E-3	3.82E-4
Ni-63	1.00E-12	1.17E-07	8.53E-6	Ni-63	4.48E-3	3.82E-8

Spreadsheet cases are run for all three monitors at decay times of 2 hours and 36 hours in consideration of EAL entry thresholds that are mode dependent.

See Section 2.0 for results.

8.0 Computer Software

No computer software was used in this calculation.

9.0 Impact Assessment

This calculation is based on “realistic” assumptions for the purpose of declaring EALs, rather than typical conservative “bounding” type design basis analyses. The calculation results are intended to provide order of magnitude setpoints to assist Operations and Emergency Response personnel in determining the state of the three fission product barriers in accordance with NEI 99-01 Rev. 6.

Ingestion Dose:	8.24	mrem
+ Boating Dose:	1.79	mrem
Resultant Total mRem:	10.0	mrem

Monitor: **GSW** **RIS4767**

Variables

FGR12 Units Conversion Factor:	3.70E+15		FGR12 Boating/Immersion Reduction Factor:	0.50
Monitor c[s:	23,200	cps		
Monitor Efficiency:	2.19E+6	cps / uCi/ml	10CFR20 Ingestion Age Consideration Factor:	2
			Decay Hrs:	2
Volume Consumed:	500	mL	Dilution Factor:	0.20

Resultant River Gamma Concentration for the Given CPS Reading :

23200	~~cps~~		µCi/ml	0.20	=	2.12E-03	µCi/ml
		2.19E+06	~~cps~~				

Child Thyroid: 50.0 **mRem**

Variables

Monitor: GSW RIS4767

Decay Hours	**2**	**ODAM Conversion Factor:**	**3.785E-03**
Monitor CPS:	**49,100** cps		
Monitor Efficiency:	**2.19E+6** cps uCi/ml	**Exposure Time (Mins.):**	**60**
Dilution Factor:	**0.20**	**Combining Factor:**	2.271E-01

Resultant Gamma Concentration for the Given CPS Reading :

$$\frac{49100 \ \cancel{cps}}{} \times \frac{uCi/ml}{2.19E+06 \ \cancel{cps}} \times 0.20 = 4.48E-03 \ uCi/ml$$

Ingestion Dose:	9.78	mrem
+ Boating Dose:	0.19	mrem
Resultant Total mRem:	10.0	mrem

Variables

Monitor: GSW RIS4767

FGR12 Units Conversion Factor:	3.70E+15		FGR12 Boating/Immersion Reduction Factor:	0.50
Monitor c[s:	10,400	cps		
Monitor Efficiency:	2.19E+6	cps / uCi/ml	10CFR20 Ingestion Age Consideration Factor:	2
			Decay Hrs:	36
Volume Consumed:	500	mL	Dilution Factor:	0.20

Resultant River Gamma Concentration for the Given CPS Reading :

$$\frac{10400 \ \cancel{cps}}{} \left| \frac{\mu Ci/ml}{2.19E+06 \ \cancel{cps}} \right| \frac{0.20}{} = \frac{9.50E-04 \ \mu Ci/ml}{}$$

Child Thyroid: 49.9 **mRem**

Variables

Monitor: GSW RIS4767

Decay Hours 36

ODAM Conversion Factor: 3.785E-03

Monitor CPS: 14,000 cps

Monitor Efficiency: 2.19E+6 cps uCi/ml

Exposure Time (Mins.): 60

Dilution Factor: 0.20

Combining Factor: 2.271E-01

Resultant Gamma Concentration for the Given CPS Reading :

14000	~~cps~~		uCi/ml	0.20	=	1.28E-03	uCi/ml
		2.19E+06	~~cps~~				

Ingestion Dose:	**8.24**	**mrem**
+ Boating Dose:	**1.79**	**mrem**
Resultant Total mRem:	**10.0**	**mrem**

Variables

Monitor: RHRSW/ESW RM1997

FGR12 Units Conversion Factor:	**3.70E+15**	
FGR12 Boating/Immersion Reduction Factor:	**0.50**	
Monitor c[s:	**16,000**	cps
10CFR20 Ingestion Age Consideration Factor:	**2**	
Monitor Efficiency:	**1.51E+6**	cps / uCi/ml
Decay Hrs:	**2**	
Volume Consumed:	**500**	mL
Dilution Factor:	**0.20**	

Resultant River Gamma Concentration for the Given CPS Reading :

16000	~~cps~~		µCi/ml	0.20	=	2.12E-03	µCi/ml
		1.51E+06	~~cps~~				

Child Thyroid: 50.0 **mRem**

Monitor: RHRSW/ESW RM1997

Variables

Decay Hours 2

ODAM Conversion Factor: 3.785E-03

Monitor CPS: 33,800 cps

Monitor Efficiency: 1.51E+6 cps uCi/ml

Exposure Time (Mins.): 60

Dilution Factor: 0.20

Combining Factor: 2.271E-01

Resultant Gamma Concentration for the Given CPS Reading :

$$\frac{33800 \; \cancel{cps}}{} \cdot \frac{uCi/ml}{1.51E{+}06 \; \cancel{cps}} \cdot 0.20 = 4.48E{-}03 \; uCi/ml$$

Ingestion Dose:	9.82	mrem
+ Boating Dose:	0.19	mrem
Resultant Total mRem:	10.0	mrem

Variables

Monitor: RHRSW/ESW RM1997

FGR12 Units Conversion Factor:	3.70E+15		FGR12 Boating/Immersion Reduction Factor:	0.50
Monitor c[s:	7,200	cps		
Monitor Efficiency:	1.51E+6	cps / uCi/ml	10CFR20 Ingestion Age Consideration Factor:	2
			Decay Hrs:	36
Volume Consumed:	500	mL		
			Dilution Factor:	0.20

Resultant River Gamma Concentration for the Given CPS Reading :

$$\frac{7200\ \cancel{cps}}{} \cdot \frac{\mu Ci/ml}{1.51E+06\ \cancel{cps}} \cdot \frac{0.20}{} = 9.54E\text{-}04\ \mu Ci/ml$$

Child Thyroid: 49.9 **mRem**

Variables

Monitor: **RHRSW/ESW** **RM1997**

Decay Hours	**36**	**ODAM Conversion Factor:**	**3.785E-03**
Monitor CPS:	**9,650** cps		
Monitor Efficiency:	**1.51E+6** cps uCi/ml	**Exposure Time (Mins.):**	**60**
Dilution Factor:	**0.20**	**Combining Factor:**	2.271E-01

Resultant Gamma Concentration for the Given CPS Reading :

$$\frac{9650\ \cancel{cps}}{} \cdot \frac{uCi/ml}{1.51E{+}06\ \cancel{cps}} \cdot \frac{0.20}{} = 1.28E{-}03\ uCi/ml$$

Ingestion Dose:	**8.22**	**mrem**
+ Boating Dose:	**1.79**	**mrem**
Resultant Total mRem:	**10.0**	**mrem**

Variables

Monitor: RHRSW Dilution Line RM4268

FGR12 Units Conversion Factor:	3.70E+15		**FGR12 Boating/Immersion Reduction Factor:**	**0.50**
Monitor c[s:	**24,200**	cps		
Monitor Efficiency:	2.29E+6	cps / uCi/ml	**10CFR20 Ingestion Age Consideration Factor:**	**2**
			Decay Hrs:	**2**
Volume Consumed:	**500**	mL	**Dilution Factor:**	**0.20**

Resultant River Gamma Concentration for the Given CPS Reading :

24200	~~cps~~		μCi/ml	0.20	=	**2.11E-03**	μCi/ml
		2.29E+06	~~cps~~				

Child Thyroid: 50.0 **mRem**

Monitor: RHRSW Dilution Line RM4268

Variables

Decay Hours	**2**	**ODAM Conversion Factor:**	**3.785E-03**
Monitor CPS:	**51,300** cps		
Monitor Efficiency:	**2.29E+6** cps uCi/ml	**Exposure Time (Mins.):**	**60**
Dilution Factor:	**0.20**	**Combining Factor:**	2.271E-01

Resultant Gamma Concentration for the Given CPS Reading :

$$\frac{51300\ \cancel{cps}}{} \cdot \frac{uCi/ml}{2.29E+06\ \cancel{cps}} \cdot \frac{0.20}{} = \mathbf{4.48E\text{-}03}\ uCi/ml$$

Ingestion Dose:	**9.80**	**mrem**
+ Boating Dose:	**0.19**	**mrem**
Resultant Total mRem:	**10.0**	**mrem**

Variables

Monitor: RHRSW Dilution Line RM4268

FGR12 Units Conversion Factor:	**3.70E+15**	
Monitor c[s:	**10,900**	cps
Monitor Efficiency:	**2.29E+6**	cps / uCi/ml
Volume Consumed:	**500**	mL
FGR12 Boating/Immersion Reduction Factor:	**0.50**	
10CFR20 Ingestion Age Consideration Factor:	**2**	
Decay Hrs:	**36**	
Dilution Factor:	**0.20**	

Resultant River Gamma Concentration for the Given CPS Reading :

$$\frac{10900\ \cancel{cps}}{} \times \frac{\mu Ci/ml}{2.29E+06\ \cancel{cps}} \times 0.20 = 9.52E-04\ \mu Ci/ml$$

Child Thyroid: 49.9 **mRem**

Variables

Monitor: RHRSW Dilution Line RM4268

Decay Hours 36

ODAM Conversion Factor: 3.785E-03

Monitor CPS: 14,650 cps

Exposure Time (Mins.): 60

Monitor Efficiency: 2.29E+6 cps uCi/ml

Dilution Factor: 0.20

Combining Factor: 2.271E-01

Resultant Gamma Concentration for the Given CPS Reading :

$$\frac{14650 \ \cancel{cps}}{} \cdot \frac{\text{uCi/ml}}{2.29E{+}06 \ \cancel{cps}} \cdot 0.20 = 1.28E{-}03 \ \text{uCi/ml}$$

ENERCON *Excellence—Every project. Every day.*	**Attachment 1** CALCULATION PREPARATION CHECKLIST	CALC NO. NEE-323-CALC-004 REV. 00

CHECKLIST ITEMS[1]	YES	NO	N/A
GENERAL REQUIREMENTS			
1. If the calculation is being performed to a client procedure, is the procedure being used the latest revision?	☐	☐	☒
The calculation is being prepared to ENERCON's procedures.			
2. Are the proper forms being used and are they the latest revision?	☒	☐	☐
3. Have the appropriate client review forms/checklists been completed?	☐	☐	☒
The calculation is being prepared to ENERCON's procedures.			
4. Are all pages properly identified with a calculation number, calculation revision and page number consistent with the requirements of the client's procedure?	☒	☐	☐
5. Is all information legible and reproducible?	☒	☐	☐
6. Is the calculation presented in a logical and orderly manner?	☒	☐	☐
7. Is there an existing calculation that should be revised or voided?	☐	☒	☐
This is a new calculation to support implementing NEI 99-01 Rev. 6			
8. Is it possible to alter an existing calculation instead of preparing a new calculation for this situation?	☐	☒	☐
9. If an existing calculation is being used for design inputs, are the key design inputs, assumptions and engineering judgments used in that calculation valid and do they apply to the calculation revision being performed.	☐	☐	☒
10. Is the format of the calculation consistent with applicable procedures and expectations?	☒	☐	☐
11. Were design input/output documents properly updated to reference this calculation?	☐	☐	☒
12. Can the calculation logic, methodology and presentation be properly understood without referring back to the originator for clarification?	☒	☐	☐
OBJECTIVE AND SCOPE			
13. Does the calculation provide a clear concise statement of the problem and objective of the calculation?	☒	☐	☐
14. Does the calculation provide a clear statement of quality classification?	☒	☐	☐
15. Is the reason for performing and the end use of the calculation understood?	☒	☐	☐
16. Does the calculation provide the basis for information found in the plant's license basis?	☐	☒	☐
17. If so, is this documented in the calculation?	☐	☐	☒
18. Does the calculation provide the basis for information found in the plant's design basis documentation?	☐	☒	☐

CHECKLIST ITEMS[1]		YES	NO	N/A
19.	If so, is this documented in the calculation?	☐	☐	☒
20.	Does the calculation otherwise support information found in the plant's design basis documentation?	☐	☒	☐
21.	If so, is this documented in the calculation?	☐	☐	☒
22.	Has the appropriate design or license basis documentation been revised, or has the change notice or change request documents being prepared for submittal?	☐	☐	☒
DESIGN INPUTS				
23.	Are design inputs clearly identified?	☒	☐	☐
24.	Are design inputs retrievable or have they been added as attachments?	☒	☐	☐
25.	If Attachments are used as design inputs or assumptions are the Attachments traceable and verifiable?	☒	☐	☐
26.	Are design inputs clearly distinguished from assumptions?	☒	☐	☐
27.	Does the calculation rely on Attachments for design inputs or assumptions? If yes, are the attachments properly referenced in the calculation?	☒	☐	☐
28.	Are input sources (including industry codes and standards) appropriately selected and are they consistent with the quality classification and objective of the calculation?	☒	☐	☐
29.	Are input sources (including industry codes and standards) consistent with the plant's design and license basis?	☒	☐	☐
30.	If applicable, do design inputs adequately address actual plant conditions?	☒	☐	☐
31.	Are input values reasonable and correctly applied?	☒	☐	☐
32.	Are design input sources approved?	☒	☐	☐
33.	Does the calculation reference the latest revision of the design input source?	☒	☐	☐
34.	Were all applicable plant operating modes considered?	☒	☐	☐
ASSUMPTIONS				
35.	Are assumptions reasonable/appropriate to the objective?	☒	☐	☐
36.	Is adequate justification/basis for all assumptions provided?	☒	☐	☐
37.	Are any engineering judgments used?	☐	☒	☐
38.	Are engineering judgments clearly identified as such?	☐	☐	☒
39.	If engineering judgments are utilized as design inputs, are they reasonable and can they be quantified or substantiated by reference to site or industry standards, engineering principles, physical laws or other appropriate criteria?	☐	☐	☒

CHECKLIST ITEMS[1]		YES	NO	N/A
METHODOLOGY				
40.	Is the methodology used in the calculation described or implied in the plant's licensing basis?	☐	☒	☐
41.	If the methodology used differs from that described in the plant's licensing basis, has the appropriate license document change notice been initiated?	☐	☐	☒
42.	Is the methodology used consistent with the stated objective?	☒	☐	☐
43.	Is the methodology used appropriate when considering the quality classification of the calculation and intended use of the results?	☒	☐	☐
BODY OF CALCULATION				
44.	Are equations used in the calculation consistent with recognized engineering practice and the plant's design and license basis?	☒	☐	☐
45.	Is there reasonable justification provided for the use of equations not in common use?	☐	☐	☒
46.	Are the mathematical operations performed properly and documented in a logical fashion?	☒	☐	☐
47.	Is the math performed correctly?	☒	☐	☐
48.	Have adjustment factors, uncertainties and empirical correlations used in the analysis been correctly applied?	☒	☐	☐
49.	Has proper consideration been given to results that may be overly sensitive to very small changes in input?	☒	☐	☐
SOFTWARE/COMPUTER CODES				
50.	Are computer codes or software languages used in the preparation of the calculation?	☐	☒	☐
51.	Have the requirements of CSP 3.09 for use of computer codes or software languages, including verification of accuracy and applicability been met?	☐	☐	☒
52.	Are the codes properly identified along with source vendor, organization, and revision level?	☐	☐	☒
53.	Is the computer code applicable for the analysis being performed?	☐	☐	☒
54.	If applicable, does the computer model adequately consider actual plant conditions?	☐	☐	☒
55.	Are the inputs to the computer code clearly identified and consistent with the inputs and assumptions documented in the calculation?	☐	☐	☒
56.	Is the computer output clearly identified?	☐	☐	☒
57.	Does the computer output clearly identify the appropriate units?	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
58. Are the computer outputs reasonable when compared to the inputs and what was expected?	☐	☐	☒
59. Was the computer output reviewed for ERROR or WARNING messages that could invalidate the results?	☐	☐	☒
RESULTS AND CONCLUSIONS			
60. Is adequate acceptance criteria specified?	☒	☐	☐
61. Are the stated acceptance criteria consistent with the purpose of the calculation, and intended use?	☒	☐	☐
62. Are the stated acceptance criteria consistent with the plant's design basis, applicable licensing commitments and industry codes, and standards?	☒	☐	☐
63. Do the calculation results and conclusions meet the stated acceptance criteria?	☒	☐	☐
64. Are the results represented in the proper units with an appropriate tolerance, if applicable?	☒	☐	☐
65. Are the calculation results and conclusions reasonable when considered against the stated inputs and objectives?	☒	☐	☐
66. Is sufficient conservatism applied to the outputs and conclusions?	☒	☐	☐
67. Do the calculation results and conclusions affect any other calculations?	☐	☒	☐
68. If so, have the affected calculations been revised?	☐	☐	☒
69. Does the calculation contain any conceptual, unconfirmed or open assumptions requiring later confirmation?	☐	☒	☐
70. If so, are they properly identified?	☐	☐	☒
DESIGN REVIEW			
71. Have alternate calculation methods been used to verify calculation results?	☐	☐	☒
No, a Design Review was performed.			

Note:

1. Where required, provide clarification/justification for answers to the questions in the space provided below each question. An explanation is required for any questions answered as "No' or "N/A".

Originator: Jay Bhatt — 12/12/17

Print Name and Sign — Date

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ALARM SETPOINTS FOR LIQUID RAD MONITORS	**Rev. 17** Page 9 of 11

ATTACHMENT 1

LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 15 - 7198 ✓ 2. Sample Date & Time 12-1-15 / 11:03 ✓
3. Stream/Monitor Description GSW rad monitor RM-4767 ✓
4. Effluent Monitor Reading (cps) 10 ✓
5. Effluent Flow (gpm) 9600 gpm ✓
6. Average effluent flow during time represented by sample, F_1 (gpm) NR ✓
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) NR ✓
8. Monitor calibration factor, g, (cps/µCi/mL) ~~[illegible]~~ 2.19 e6 ✓
9. Previous alarm value setpoint (cps) 765 cps ✓
10. Fraction to apply as a safety margin, A = 0.5

$$\textbf{Setpoint} = 10\times\left[\frac{\Sigma_i K_i}{\Sigma_i (K_i \div WEC_i)}\times g\times\frac{F_2}{F_1}\times A\right]+Bkg \quad = \quad \text{Setpoint}=10\times\left[\frac{(15)(8)(7)}{(16)(6)}\times(10)\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(15)(8)(7)}{(16)(6)}\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(9.43e^{-3})(2.19e^{6})(\text{NR})}{(177.54)(\text{NR})}\right]+(10)$$

11. Setpoint = 592 ✓

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11)-(9)}{(9)} = \frac{(592)-(765)}{(765)}$

12. Fractional Change = -.226 ✓

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint

13. Monitor Hi Alarm = 765 cps ✓

14. Radwaste Monitor Hi Alarm = .16 (11) = .16 ( ) = N/A cps ✓

w/o 40328724

PLANT CHEMISTRY PROCEDURES 3200 MANUAL	PCP 8.7
ALARM SETPOINTS FOR LIQUID RAD MONITORS	Rev. 17 Page 9 of 11

ATTACHMENT 1

LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 15-4896 2. Sample Date & Time 8-24-15 / 0027
3. Stream/Monitor Description Rm-1997 (RHRSW/ESW)
4. Effluent Monitor Reading (cps) 30
5. Effluent Flow (gpm) 4800
6. Average effluent flow during time represented by sample, F_1 (gpm) N/A
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) N/A
8. Monitor calibration factor, g, (cps/µCi/mL) 1.51e6 $\frac{1}{6.62e^{-7}}$
9. Previous alarm value setpoint (cps) 614
10. Fraction to apply as a safety margin, A = 0.5

$$\textbf{Setpoint} = 10\times\left[\frac{\Sigma_i K_i}{\Sigma_i(K_i \div WEC_i)} \times g \times \frac{F_2}{F_1} \times A\right] + Bkg \quad = \quad \text{Setpoint} = 10\times\left[\frac{(15)(8)(7)}{(16)(6)}\times(10)\right] + (4)$$

$$\text{Setpoint} = 5\text{x}\left[\frac{(15)(8)(7)}{(16)(6)}\right] + (4)$$

$$\text{Setpoint} = 5\text{x}\left[\frac{(\ \cancel{1.01e^{-8}}\ 1.10e^{-2})(\ 1.51e^{6}\ )(\ N/A\ )}{(\ \cancel{102.078}\ 153.84\ )(\ N/A\ )}\right] + (\ 30\ )$$

11. Setpoint = ~~735.6~~ 569.85

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11)-(9)}{(9)} = \frac{(\ \cancel{735.6}\ 569.9\ )-(\ 614\ )}{(\ 614\ )}$ 8-24-15

12. Fractional Change = ~~0.20~~ −0.07

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint

13. Monitor Hi Alarm = 614

14. Radwaste Monitor Hi Alarm = .16 (11) = .16 (   ) = N/A cps

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ALARM SETPOINTS FOR LIQUID RAD MONITORS	Rev. 17 Page 9 of 11

ATTACHMENT 1

Page 1 of 3

LIQUID EFFLUENT RADIOACTIVITY MONITOR SETPOINT

1. Sample No. 14-884 2. Sample Date & Time 2-14-14 / 0021
3. Stream/Monitor Description RHRSW/ESW RUPTURE: Rm 4268
4. Effluent Monitor Reading (cps) 20
5. Effluent Flow (gpm) RHRSW 'A' = 4800gpm , RHRSW 'B' - 4800gpm
6. Average effluent flow during time represented by sample, F_1 (gpm) N/A
7. Average dilution (discharge canal) flow during time represented by sample, F_2 (gpm) N/A
8. Monitor calibration factor, g, (cps/µCi/mL) 2.29E6 ✓
9. Previous alarm value setpoint (cps) 863 ✓
10. Fraction to apply as a safety margin, A = 0.5

$$\textbf{Setpoint} = 10\times\left[\frac{\Sigma_i K_i}{\Sigma_i(K_i \div WEC_i)}\times g\times\frac{F_2}{F_1}\times A\right]+Bkg \quad = \quad \text{Setpoint}=10\times\left[\frac{(15)(8)(7)}{(16)(6)}\times(10)\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(15)(8)(7)}{(16)(6)}\right]+(4)$$

$$\text{Setpoint} = 5\times\left[\frac{(7.67E\text{-}3)(2.29E^{6})(N/A)}{(111.86)(N/A)}\right]+(20)$$

11. Setpoint = 805 ✓

Fractional Change = $\frac{\text{New value - Previous Value}}{\text{Previous Value}} = \frac{(11)-(9)}{(9)} = \frac{(805)-(863)}{(863)}$

12. Fractional Change = -0.07 ✓

If fractional change is greater than ±0.3, adopt a new monitor alarm setting.

Continuous Monitor Hi Alarm = Setpoint ⇒ OLD SETPOINT ✓

13. Monitor Hi Alarm = 863 ✓

14. Radwaste Monitor Hi Alarm = .16 (11) = .16 (N/A) = N/A cps

Development of EAL Threshold values from NEE-323-CALC-002

Due to elevated background radiation levels on these monitors during plant operation (10-12 R/hr), the calculated threshold value was rounded to 5 (minimum serviceable threshold value accounting for scale of monitor) for ease of use by the EAL evaluator, and the "in Mode 5 only" caveat is added to the EAL usage.

The resultant EALs are:

RA2.2 Reading greater than 5 R/hr on ANY of the following radiation monitors (in Mode 5 only):

- NW Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

CS1/CG1 Core uncovery is indicated by ANY of the following:

- Drywell Monitor (9184A/B) reading greater than 5.0 R/hr

ENERCON Excellence—Every project. Every day.	CALCULATION COVER SHEET	CALC NO. NEE-323-CALC-002
		REV. 00
		PAGE NO. 1 of 28

Title:	Dose Rate Evaluation of Reactor Vessel Water Levels During Refueling for EAL Thresholds	Client: Duane Arnold Energy Center
		Project Identifier: NEE-323

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions, including preliminary information, that require confirmation? (If **YES**, identify the assumptions.)	☐	☒
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, identify the design verified calculation.) **Design Verified Calculation No.** ______	☐	☒
3	Does this calculation supersede an existing Calculation? (If **YES**, identify the design verified calculation.) **Superseded Calculation No.** ______	☐	☒

Scope of Revision:
Initial Issue

Revision Impact on Results:
Initial Issue

Study Calculation ☐ **Final Calculation** ☒

Safety-Related ☐ **Non-Safety-Related** ☒

(Print Name and Sign)

Originator: Jay Bhatt	**Date:** **12/12/17**
Design Verifier[1] (Reviewer if NSR): Caleb Trainor	**Date:** **12/12/17**
Approver: Aaron Holloway	**Date:** **12/12/17**

Note 1: For non-safety-related calculation, design verification can be substituted by review.

ENERCON Excellence—Every project. Every day.	CALCULATION REVISION STATUS SHEET	CALC NO. NEE-323-CALC-002
		REV. 00

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
00	12/12/17	Initial Issue

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
All	00		

APPENDIX/ATTACHMENT REVISION STATUS

APPENDIX NO.	NO. OF PAGES	REVISION NO.	ATTACHMENT NO.	NO. OF PAGES	REVISION NO.
A	1	00	1	5	00
B	2	00			
C	1	00			

1.0 Purpose and Scope

The purpose of this calculation is to evaluate dose rates with water at the top of active fuel in the reactor vessel during cold shutdown or refueling operations in order to set Emergency Action Level (EAL) thresholds (RA2, CS1, CG1) per NEI 99-01 [Reference 3.5]. The dose rates are calculated at the locations of the drywell monitors 9184A/B so that dose rate measurements by these devices can be correlated to the water level in the core, upon failure of other water level detection systems. This calculation is nonsafety-related as the results of the calculation do not affect the design basis or safety-related systems structures or components. These results are best estimates based on as-built conditions and provide information to operators with respect to classifying an emergency, therefore no acceptance criteria is required.

2.0 Summary of Results and Conclusions

The dose rates just prior to the core being uncovered (i.e. water at the top of the active fuel) are shown in the table below. Note that the results presented below are calculated dose rates and do not account for background radiation or any installed detector check sources.

Table 1 – Dose Rate at Top of Active Fuel

Model Description	Drywell Monitor 9184A Reading (R/hr)	Drywell Monitor 9184B Reading (R/hr)	Drywell Monitor (9184A/B) Range (R/hr)
Head Off	1.81	1.68	1 to 1E+7
Head On	1.11	7.41E-01[1]	1 to 1E+7

[1] This value is off scale low.

3.0 References

3.1 "Standard Composition Library," ORNL/NUREG/CSD-2/V1/R6, Volume 3, Section M8, March 2000

3.2 CGDG-SCALE-6.1.2, Rev 00, Commercial Grade Dedication SCALE Version 6.1.2

3.3 CGDG-MCNP6-V1.0, Rev 00, Commercial Grade Dedication MCNP6 Version 1.0

3.4 ANSI/ANS 6.1.1-1977, Neutron and Gamma Flux-To-Dose Conversion Factors

3.5 NEI 99-01, Rev. 6, "Development of Emergency Action Levels for Non-Passive Reactors"

3.6 I.RIM-V115-01, Rev. 10, "Victoreen Model 876A Containment Radiation Monitor Calibration"

3.7 NUREG 1940, "RASCAL 4: Descriptions of Models and Methods"

3.8 CAL-R00-PUP-008, Rev. 03, "Non-LOCA Radiological Consequence Dose with Alternate Source Term"

3.9 RFP 110, Rev. 45, "Refueling Procedure- Reactor Pressure Vessel Disassembly"

3.10 Technical Specifications, Section 1.1

3.11 Technical Specifications, Section 4.2.1

3.12 NUREG 1754, "A New Comparative Analysis of LWR Fuel Designs"

3.13 BECH-M009, Rev. 14, "Equipment Locations Reactor Building Section-GG"

3.14 BECH-C405, Rev. 14, "Reactor Building Floor Plan @ El. 757'-6""

3.15 NG-17-0156, Proprietary Data Transmittal to ENERCON

3.16 BECH-M405, Sh 04, Rev. 24, "Instrument Points and Rack Locations Diagram Plans at Elevs 812'-0" & 833'-6""

3.17 NG-88-0966, "G.E. Fuel Damage Documentation/Dose Rate Calculations"

3.18 C003-029, Rev. 0, "Drywell Cylindrical Shell & Cone"

3.19 VS-01-06, Rev. 4, "Top Head Assembly"

3.20 BECH-C511, Rev. 5, "Reactor Building RPV Ped Dev. Elev. & Sect's"

3.21 BECH-C514, Rev. 1, "Drywell Interior Biological Shield Wall Reinforcing Sections"

3.22 BECH-C-516, Rev. 6, "Drywell Interior Biological Shield Wall Plans El. 816'-3 ¼" to El 779'-1 ½""

3.23 BECH-M405, Sh 02, Rev. 71, "Instrument Points & Lines Diagrams Plan at Elev 757'-6""

3.24 APED-B-31-2816-001, Rev. 5, "Outline Reactor Recirculating Pump"

3.25 FSAR Section 4.3.2.1, and Section 9.1

3.26 CAL-M98-058, Rev. 1, "ADS Accumulator Size Verification"

4.0 Assumptions

4.1 The core is homogenized based on the typical 10x10 fuel assembly dimensions, taking into account the fuel rods and space between. Any small variations in fuel parameters will have a negligible effect on containment dose rates. The cladding is modeled as Zircaloy 4 in lieu of ZIRLO; this is acceptable due to the similarity of the materials.

4.2 Any non-fuel hardware, including rod end plugs, is ignored in the active fuel region. This is acceptable since the primary self-shielding occurs in the fuel itself, and there may be some unknown streaming effects through the non-fuel hardware. This homogenization takes into account the presence of water when calculating the isotopic weight fraction and homogenized density. For the case with the reactor vessel head in place, the region between the head and the active fuel region is homogenized based on the actual mass of the upper internals over the entire region. Homogenization of source regions and shields is acceptable due to the insignificant effects on the detector response given the model geometry.

4.3 The composition of the containment structure and components are based on the values in the SCALE standard composition library [Reference 3.1]. These material properties are commonly used in shielding applications, and are acceptable for modelling the structures and components used to determine the best estimate response at the detector locations.

4.4 The minimum period of decay after reactor shutdown before moving fuel is 60 hours [Reference 3.8, Section 4.3.8]. This calculation assumes a decay time of 50 hours to allow EAL thresholds to be determined for reactor vessel conditions that exist prior to the commencement of fuel movement which is representative of the applicable operating modes (cold shutdown, refueling). This decay time is appropriate to produce best estimate results for both the head on and head off configurations.

4.5 The hardware in the upper internals region between the active fuel region, reactor recirculating pumps and reactor vessel head is assumed to be stainless steel type 304. While the actual composition of the hardware may vary slightly, small variations in the material will have a negligible effect on the dose rate response at the detectors.

4.6 It is assumed that the water below the active fuel region is liquid at a constant temperature. Using a density of 0.9982 g/cm^3 is common in shielding applications. Any water above this region would be steam with little shielding value.

4.7 The source term is generated shortly after shutdown, therefore, the fuel gamma source term will predominate and the neutron-gamma and hardware activation can be neglected.

4.8 The high range detectors read out in roentgen per hour (R/h) which is a measurement of exposure rate, while the MCNP output is provided in mrem/h which is a measurement of the equivalent dose rate that represents the biological effects of ionizing radiation. It is assumed that 1 R is approximately 1000 mrem. This is acceptable as only the gamma source term is considered.

4.9 The roof of the Reactor Building is modeled as 0.5 inches of stainless steel. This will account for any scattering interactions that may contribute to the response at the detector. The magnitude of the detector response due to scattering off of the roof will be small due to the geometry and amount of shielding in the model, and is therefore acceptable.

4.10 Automatic Depressurization System Accumulators 1R003A/B/C located on the 775'-11 ½" elevation are not included in the model. The size of the accumulators are 200 gallons [Reference 3.26]. This is relatively small compared to the geometry of the model, and the corresponding scatter interactions will not have a significant impact on the detector response.

5.0 Design Inputs

5.1 Fuel Assembly Parameters

The following fuel assembly parameters are used to homogenize the core in the MCNP model. They are based on typical fuel assembly values for 10x10 fuel.

Table 2 – Design Input Fuel Assembly Parameters

Parameters	Value	Unit	Reference
Fuel type	10x10		3.25
# of Assemblies in Core	368		3.11
# Fuel rods per assembly	92		3.12
Pitch	0.51	[in]	3.12
Density (% of theoretical)	95		3.12
Fuel pellet OD	0.336	[in]	3.12
Fuel rod OD	0.395	[in]	3.12
Clad thickness	0.026	[in]	3.12
Active length	144	[in]	3.12

5.2 Model Dimensions

The following elevations and dimensions are based on the associated drawings or other reference. Some parameters are estimated using drawing scales when exact dimensions are not provided.

Table 3 – Design Input Dimensions

Dimension	ft	in	cm	Reference
Pedestal inner radius	8		243.84	3.20
Pedestal outer radius	12		365.76	3.20
Reactor vessel inner diameter		185.375	470.85	3.15
Reactro vessel thickness		5	12.70	3.15
Drywell spherical portion radius	31.5		960.12	3.17 Figure 2
Concrete around drywell spherical portion(x and y directions radius)	36	9	1120.14	3.14
Drywell cylindrical portion radius	17		518.16	3.16
Drywell liner thickness		0.75	1.91	3.18
Concrete around drywell cylindrical portion (x and y directions)	22	9	693.42	3.16
Reactor Building (x and y directions)	140		4267.20	3.14
Reactor Building Roof Thickness		0.5	1.27	Assumption 4.9
Height of active fuel		144	365.76	3.12
Vessel Height		704.5	1789.43	3.15

Dimension	ft	in	cm	Reference
Reactor vessel head thickness		3.9375	10.00	3.19
Distance from vessel 0 to bottom of active fuel		200.94	510.39	3.15, 3.12
Bio shield inner radius	9	6.25	290.20	3.21
Bio shield outer radius	11	8.25	356.24	3.21
Reactor recirculating pump height	17	2	523.24	3.24
Reactor recirculating pump radius	2	9	83.82	3.24
Detector RE-9184A distance from origin (x plane)	-4		-121.92	3.23 [Scaled]
Detector RE-9184A distance from origin (y plane)	13.33		406.29	3.23 [Scaled]
Detector RE-9184B distance from origin (x plane)	6		182.88	3.23 [Scaled]
Detector RE-9184B distance from origin (y plane)	-12		-365.76	3.23 [Scaled]
Reactor Recirculating Pump IP-201A distance from origin (x plane)	12		365.76	3.23 [Scaled]
Reactor Recirculating Pump IP-201A distance from origin (y plane)	12		365.76	3.23 [Scaled]
Reactor Recirculating Pump IP-201B distance from origin (x plane)	-12		-365.76	3.23 [Scaled]
Reactor Recirculating Pump IP-201B distance from origin (y plane)	-12		-365.76	3.23 [Scaled]

Table 4 – Design Input Elevations[2]

Dimension:	ft.	in	cm	Reference
Drywell Equator	766	0.5	0.00	3.13
Vessel 0	772	5.5	195.58	3.15
Bottom of pedestal elevation	742	9	-709.93	3.13
Top of cylindrical portion of drywell concrete	855		2711.45	3.13
Top of Reactor Building	897	6	4006.85	3.13
Detector elevation	760		-184.15	3.17
Top of pedestal/ bottom of bio shield	770	10.5	147.32	3.20
Top of bio shield	816	3.25	1530.99	3.22
Reactor recirculating pump bottom	748	8.5	-528.32	3.13

[2] All elevations listed in centimeters are relative to the equator of the drywell elevation of 766' 0.5" [Reference 3.13].

5.3 Core Isotopic Inventory

Core isotopic activities in Ci/MWt are taken from Reference 3.7 Table 1-1. A table of the input values is shown in Table 5, below. The activities in Ci are determined by multiplying by the rated thermal power of 1912 MWt taken from Reference 3.10.

Table 5 – Core Source Term

Isotope	Ci/MWt	Ci	Isotope	Ci/MWt	Ci
Ba-139	4.74E+04	9.06E+07	Rh-105	2.81E+04	5.37E+07
Ba-140	4.76E+04	9.10E+07	Ru-103	4.34E+04	8.30E+07
Ce-141	4.39E+04	8.39E+07	Ru-105	3.06E+04	5.85E+07
Ce-143	4.00E+04	7.65E+07	Ru-106	1.55E+04	2.96E+07
Ce-144	3.54E+04	6.77E+07	Sb-127	2.39E+03	4.57E+06
Cm-242	1.12E+03	2.14E+06	Sb-129	8.68E+03	1.66E+07
Cs-134	4.70E+03	8.99E+06	Sr-89	2.41E+04	4.61E+07
Cs-136	1.49E+03	2.85E+06	Sr-90	2.39E+03	4.57E+06
Cs-137	3.25E+03	6.21E+06	Sr-91	3.01E+04	5.76E+07
I-131	2.67E+04	5.11E+07	Sr-92	3.24E+04	6.19E+07
I-132	3.88E+04	7.42E+07	Tc-99m	4.37E+04	8.36E+07
I-133	5.42E+04	1.04E+08	Te-127	2.36E+03	4.51E+06
I-134	5.98E+04	1.14E+08	Te-127m	3.97E+02	7.59E+05
I-135	5.18E+04	9.90E+07	Te-129	8.26E+03	1.58E+07
Kr-83m	3.05E+03	5.83E+06	Te-129m	1.68E+03	3.21E+06
Kr-85	2.78E+02	5.32E+05	Te-131m	5.41E+03	1.03E+07
Kr-85m	6.17E+03	1.18E+07	Te-132	3.81E+04	7.28E+07
Kr-87	1.23E+04	2.35E+07	Xe-131m	3.65E+02	6.98E+05
Kr-88	1.70E+04	3.25E+07	Xe-133	5.43E+04	1.04E+08
La-140	4.91E+04	9.39E+07	Xe-133m	1.72E+03	3.29E+06
La-141	4.33E+04	8.28E+07	Xe-135	1.42E+04	2.72E+07
La-142	4.21E+04	8.05E+07	Xe-135m	1.15E+04	2.20E+07
Mo-99	5.30E+04	1.01E+08	Xe-138	4.56E+04	8.72E+07
Nb-95	4.50E+04	8.60E+07	Y-90	2.45E+03	4.68E+06
Nd-147	1.75E+04	3.35E+07	Y-91	3.17E+04	6.06E+07
Np-239	5.69E+05	1.09E+09	Y-92	3.26E+04	6.23E+07
Pr-143	3.96E+04	7.57E+07	Y-93	2.52E+04	4.82E+07
Pu-241	4.26E+03	8.15E+06	Zr-95	4.44E+04	8.49E+07
Rb-86	5.29E+01	1.01E+05	Zr-97	4.23E+04	8.09E+07

5.4 Material Compositions

The following compositions used in the MCNP model are taken or developed from the SCALE standard composition library [Reference 3.1] and are shown in Table 6.

Table 6 – Scale Standard Compositions used in MCNP Model

Material	Isotope	Weight Fraction
Zry- 4	Zr	0.9823
(6.56 g/cm^3)	Sn	0.0145
	Cr	0.0010
	Fe	0.0021
	Hf	0.0001
UO_2	U-235	0.0348
(10.412 g/cm^3)	U-238	0.8466
	O	0.1186
Air	C	0.0001
(1.21E-03 g/cm^3)	N	0.7651
	O	0.2348
Water	H	0.1111
(0.9982 g/cm^3)	O	0.8889
SS-304	Fe	0.6838
(7.94 g/cm^3)	Cr	0.1900
	Ni	0.0950
	Mn	0.0200
	Si	0.0100
	C	0.0008
	P	0.0004
Concrete	O	0.5320
(2.30 g/cm^3)	Si	0.3370
[KENO Regular	Ca	0.0440
Concrete Standard	Al	0.0340
Mix]	Na	0.0290
	Fe	0.0140
	H	0.0100
Carbon Steel	C	0.0100
(7.82 g/cm^3)	Fe	0.9900

5.5 Upper Internals

The following weights are used in the MCNP model for the region between the active fuel and the reactor vessel head [Reference 3.9, Appendix 8.9]:

- The weight of stainless steel for the moisture separator is 83,000 lbs.
- The weight of stainless steel for the steam dryer is 50,000 lbs.

5.6 The drywell (9184 A/B) and torus (9185 A/B) radiation monitor ranges (1 to 10^7 R/hr) are taken from Reference 3.6.

5.7 ANSI/ANS-1977 Flux to Dose Factors

Flux to dose conversion factors are taken from ANSI/ANS-6.1.1-1977 [Reference 3.4] and are shown in Table 7.

Table 7 – ANSI/ANS-6.1.1-1977 Flux to Dose Factors

MeV	mrem/hr/(γ/cm²/s)	MeV	mrem/hr/(γ/cm²/s)
0.01	3.96E-03	0.8	1.68E-03
0.03	5.82E-04	1	1.98E-03
0.05	2.90E-04	2.2	3.42E-03
0.07	2.58E-04	2.6	3.82E-03

6.0 Methodology

The reactor source terms are decayed to 50 hours with ORIGEN-S of the SCALE 6.1 code package, Reference 3.2. The results are used to bin design input isotope specific activities into energy dependent photon bins. These energy specific photon emission bins are used as input for the energy distribution described by the MCNP source definitions.

The MCNP6, Reference 3.3, Monte Carlo transport code is used to determine the dose rates via the flux to dose conversion factors in Table 7, while accounting for shielding and particle transport.

The detailed engineering drawings are converted into MCNP surface and cell cards in the dimensions shown in Table 3 and Table 4. The radiation monitors of interest are modeled as point detectors to determine the expected dose rate for those detectors. The dose rates are calculated for two reactor refueling conditions:

1. With Head – the reactor is modeled with a 3.9375 inch carbon steel plate as indicated in Table 3, which is additional attenuation between the source and detector. The mass of the moisture separator and steam dryer is homogenized between the active fuel region and the vessel head.
2. Without head – the reactor is modeled with air between the active fuel zone and containment.
3. A sensitivity case is run with a mirror surface at the top of the drywell to ensure the modeling of the drywell cap would not significantly affect the response at the detector locations due to scattering.

Variance reduction is accomplished with a geometric importance map that is imposed on the homogenized core. In addition, cell based importance weighting and source biasing (see Section 7.5) are utilized to improve the variance reduction of the simple geometric scheme. A superimposed weight window mesh is utilized where necessary to improve variance. The weight windows are iteratively generated using the MCNP weight windows generator card. All final dose rates presented in this calculation include weight windows variance reduction.

7.0 Calculation

7.1 Source Terms

The ORIGEN-S input deck, *DAECEAL.inp*, is provided in Appendix C. This input produces a simple case where the isotopic composition from Table 5 is decayed. The isotope is specified in the 73$$ card using the special identifier described in Section F7.6.2 of the ORIGEN-S manual, and the activity in curies is specified in the 74** card. The time steps for the decay are given on the 60** card in hours. Although multiple time steps are calculated, the source term with 50 hours decay time is used in this calculation to model the core shortly after shutdown. The output of the decay is given in terms of photons/s/Energy-Group, which is automatically normalized in the MCNP input. The results of this calculation are summarized below in Table 8. These values are used in the MCNP input source definition.

Table 8 – Binned Total Core Source Term

Energy Group	Energy Boundaries (MeV)	Photons/sec
1	0.01-0.05	2.028E+19
2	0.05-0.1	6.572E+18
3	0.1-0.2	1.557E+19
4	0.2-0.3	9.672E+18
5	0.3-0.4	3.582E+18
6	0.4-0.6	7.837E+18
7	0.6-0.8	1.373E+19
8	0.8-1	2.132E+18
9	1-1.33	4.942E+17
10	1.33-1.66	3.579E+18
11	1.66-2	6.576E+16
12	2-2.5	7.518E+16
13	2.5-3	1.110E+17
14	3-4	8.689E+14
15	4-5	1.553E+10
16	5-6.5	2.568E+08
17	6.5-8	3.792E+07
18	8-10	8.041E+06
19	10-11	4.352E+05
totals		8.37E+19

7.2 MCNP Model Core Homogenization

The source term is given for the entire core, therefore, the self-shielding from the assemblies is an important part of the dose rate response. For simplicity, the core is modeled as a three dimensional cylinder with a uniformly distributed spatial particle distribution. The calculations for determining the mass of fuel, cladding and water for the core and the resulting density are shown below. The inputs are based on the dimensions in Table 2.

$$Assembly\ Width = (Array\ Size - 1) \times pitch + Rod\ OD = (10-1)(0.51in) + 0.395in = 4.985\ in$$

$$Active\ Fuel\ Region\ Area = (Assembly\ Width)^2 \times Number\ of\ Assemblies\ in\ Core = (4.985in)^2 \times 368 = 9144.883\ in^2$$

$$Active\ Fuel\ Equivalent\ Radius = \sqrt{Active\ Fuel\ Region\ Area/_{\pi}} = \sqrt{9144.883\ in^2/_{\pi}} = 53.953\ in$$

$$Rod\ Volume_{UO_2} = \pi(Pellet\ Radius)^2 \times Active\ Length = \pi(0.168\ in)^2(144\ in) = 12.768\ in^3$$

$$Rod\ Mass_{UO_2} = \rho \times V = \left(10.412\frac{g}{cc}\right)(12.7682\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 2178.54\ g$$

$$Assembly\ Mass_{UO_2} = Rod\ Mass \times \frac{Number\ of\ Fuel\ Rods}{Assembly} = (2178.54\ g)(92) = 200.43\ kg$$

$$Clad\ Volume = \pi\left(\frac{OD^2}{4} - \frac{ID^2}{4}\right) \times Active\ Length = (\pi)\left[\frac{(0.395\ in)^2}{4} - \frac{(0.343\ in)^2}{4}\right](144\ in) = 4.34\ in^3$$

$$Rod\ Mass_{Zry-4} = \rho \times V = \left(6.56\ \frac{g}{cc}\right)(4.34\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 466.5\ g$$

$$Assembly\ Mass_{Zry-4} = Rod\ Mass \times \frac{Number\ of\ Fuel\ Rods}{Assembly} = (466.5g)(92) = 42.92\ kg$$

$$\begin{aligned} Assembly\ \ H_2O\ Volume \\ &= [(Assembly\ Width)^2 \\ &- \pi(Rod\ Radius)^2 \times Number\ of\ Fuel\ Rods] \times Active\ Length \\ &= [(4.985\ in)^2 - (\pi)(0.1975\ in)^2(92)](144\ in) = \ 1955\ in^3 \end{aligned}$$

$$Assembly\ Mass_{H_2O} = \rho \times V = \left(0.9982\frac{g}{cc}\right)\left(\frac{1955}{in^3}\right)\left(2.54\frac{cm}{in}\right)^3 = 31.98\ kg$$

$$\begin{aligned} Assembly\ Volume = Active\ Length\ \times (Assembly\ Width)^2 = (144\ in)(4.985\ in)^2 \\ = 3578.4\ in^3 \end{aligned}$$

$$Density = \frac{Total\ Mass}{Volume} = \frac{1000g/kg(200.43 + 42.92 + 31.98)\ kg}{3578.4\ in^3\left(2.54\frac{cm}{in}\right)^3} = 4.70\ g/cc$$

The corresponding isotopic composition for the homogenized active fuel region is calculated based on the compositions in Table 6. An example calculation for the mass fraction of U-235 is included below.

$$\begin{aligned} Mass\ Fraction\ U235 &= \frac{Assembly\ Mass_{UO_2}}{Total\ Mass} \times weight\ fraction\ U235 \\ &= \frac{200.43\ kg}{(200.43 + 42.92 + 31.98)\ kg} \times 0.0348 = 0.0253 \end{aligned}$$

The remaining calculations for the homogenization are done in the worksheet Compositions of the EXCEL workbook DAEAL.xlsx and are shown in Appendix B. The isotopic compositions are calculated with the water level above the top of the fuel. Note that the EXCEL workbook uses additional significant figures.

Table 9 – Homogenization of Active Fuel Region

ZAID Number	Atom	Mass Fraction Active Fuel Region Homogenized
92235	U-235	0.0253
92238	U-238	0.6163
8016	O	0.1896
40000	Zr	0.1531
50000	Sn	0.0023
24000	Cr	0.0002
26000	Fe	0.0003
72000	Hf	0.0000
1001	H	0.0129

7.3 MCNP Model Upper Internals Homogenization

For the case with the reactor vessel head in place, the steam dryer and moisture separator region are modeled as a discrete cylinder with a uniformly distributed homogenized material to account for the mass of stainless steel between the active fuel height and reactor vessel head. The homogenization accounts for the mass of metal from Section 5.5 (assumed stainless steel type 304 per Assumption 4.5) distributed evenly across the volume between the active fuel height (Z=1071.73 cm) and the head (Z=1985.01 cm).

$$Mass\ Upper\ Internals = (83000\ lb + 50000\ lb)\left(453.59\frac{g}{lb}\right) = 6.033 \times 10^7\ g$$

The mass is divided by the volume of the region between the active fuel height and the reactor vessel head to determine the density.

$$Density\ Upper\ Internals = Mass\ Upper\ Internals \div V$$
$$= 6.033 \times 10^7\ g \div (913.28cm \times (\pi(235.43cm)^2)) = 0.379\frac{g}{cc}$$

7.4 MCNP Model Geometry

The following MCNP model geometry is based on the containment dimensions summarized in Table 3 and Table 4. The model only focuses on the primary systems and components that provide shielding or reflection from the core to the radiation monitors. These components include the reactor vessel, recirculation pumps, pedestal, biological shield and drywell. VISED plots of the model geometry are provided in Figures 1-3. The MCNP surface cards with the model dimensions (cm) are shown in Figure 4, and the cell cards are shown in Figure 5 for the cases with no reactor vessel head. A VISED plot of the model with the reactor vessel head is shown in Figure 6. Areas that are not of interest are given an importance of zero (white areas) so MCNP will not track particles in locations that will not contribute to the detector response.

Figure 1 X-Z VISED Plot of Reactor Vessel (No Head)

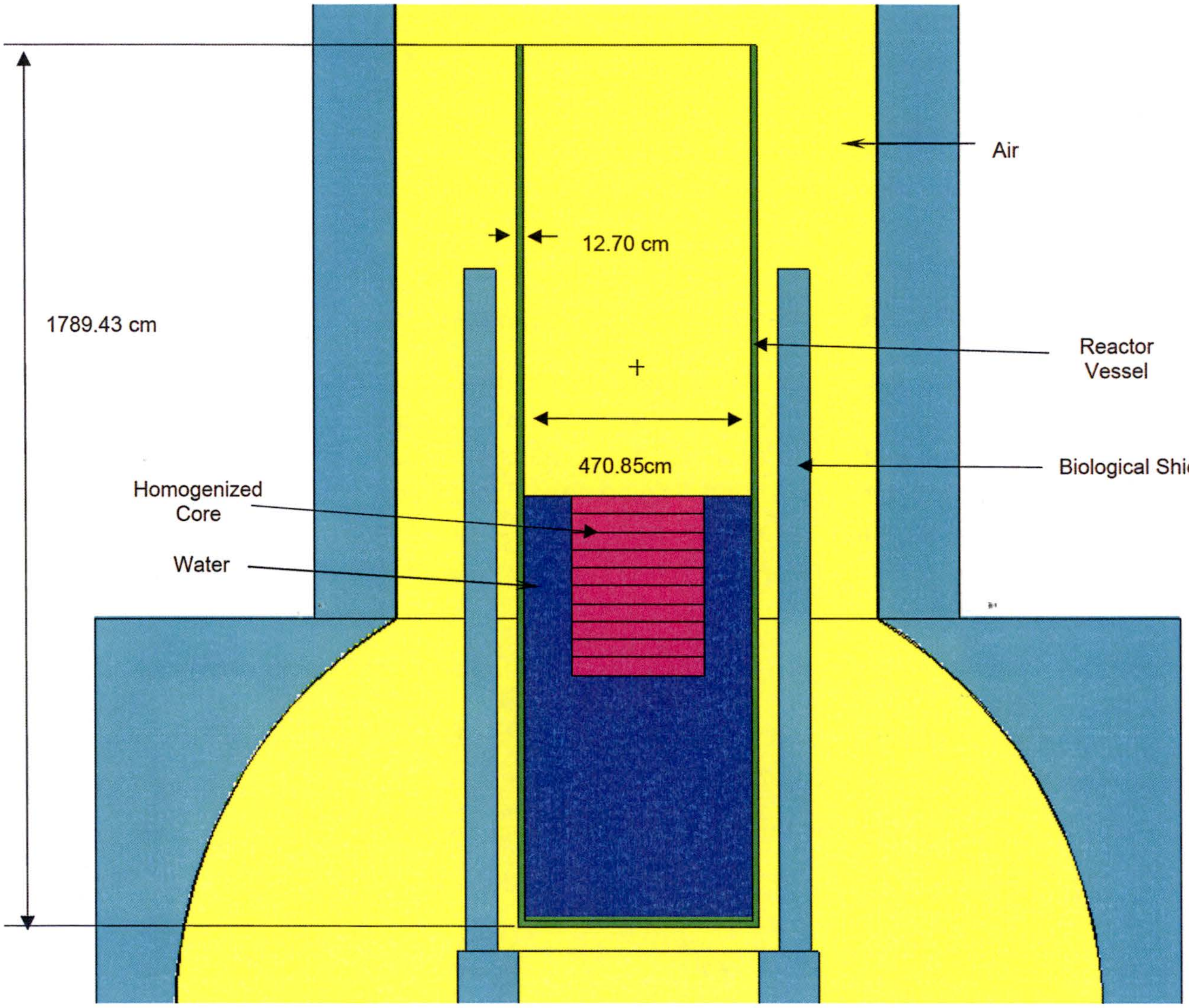
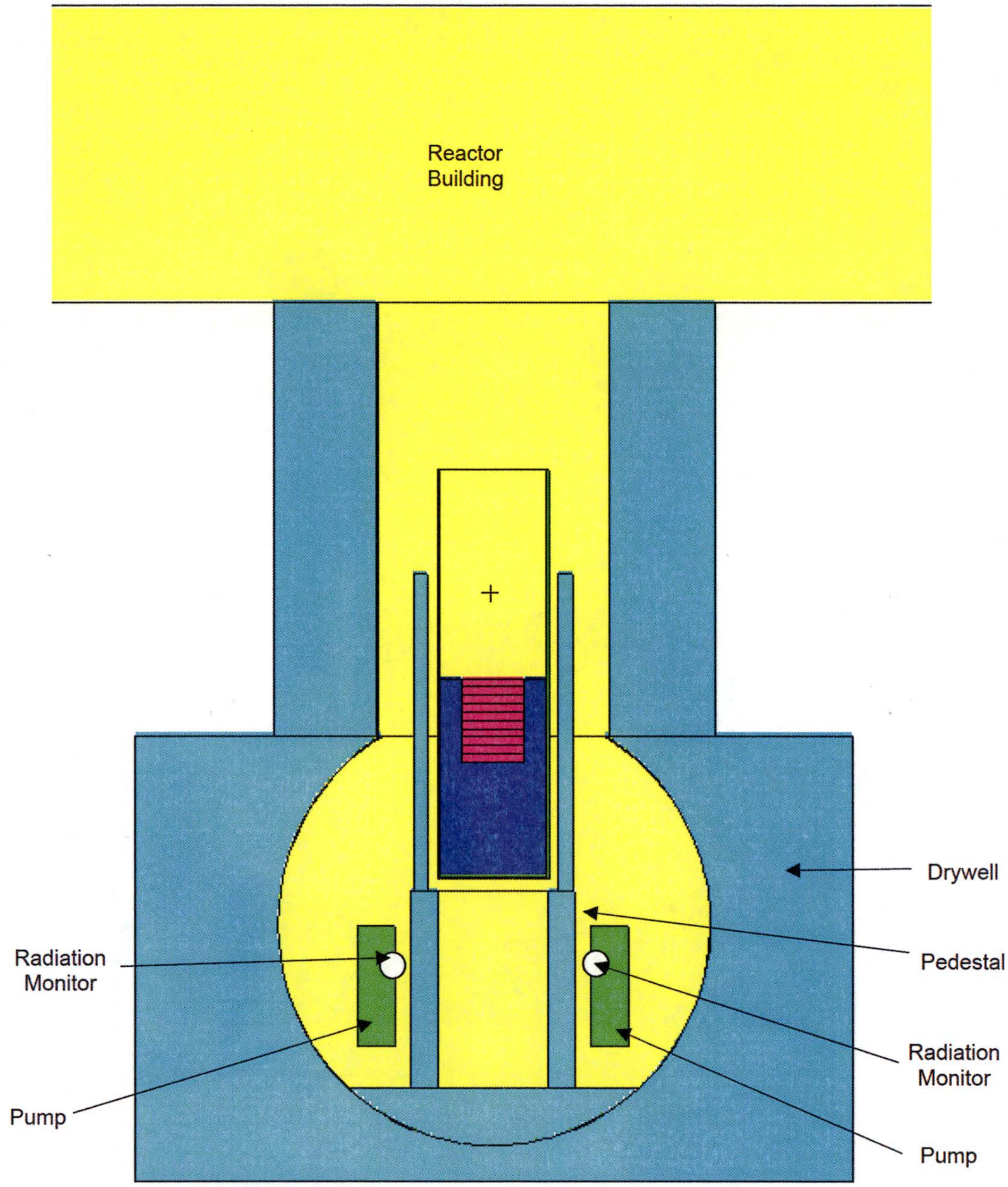
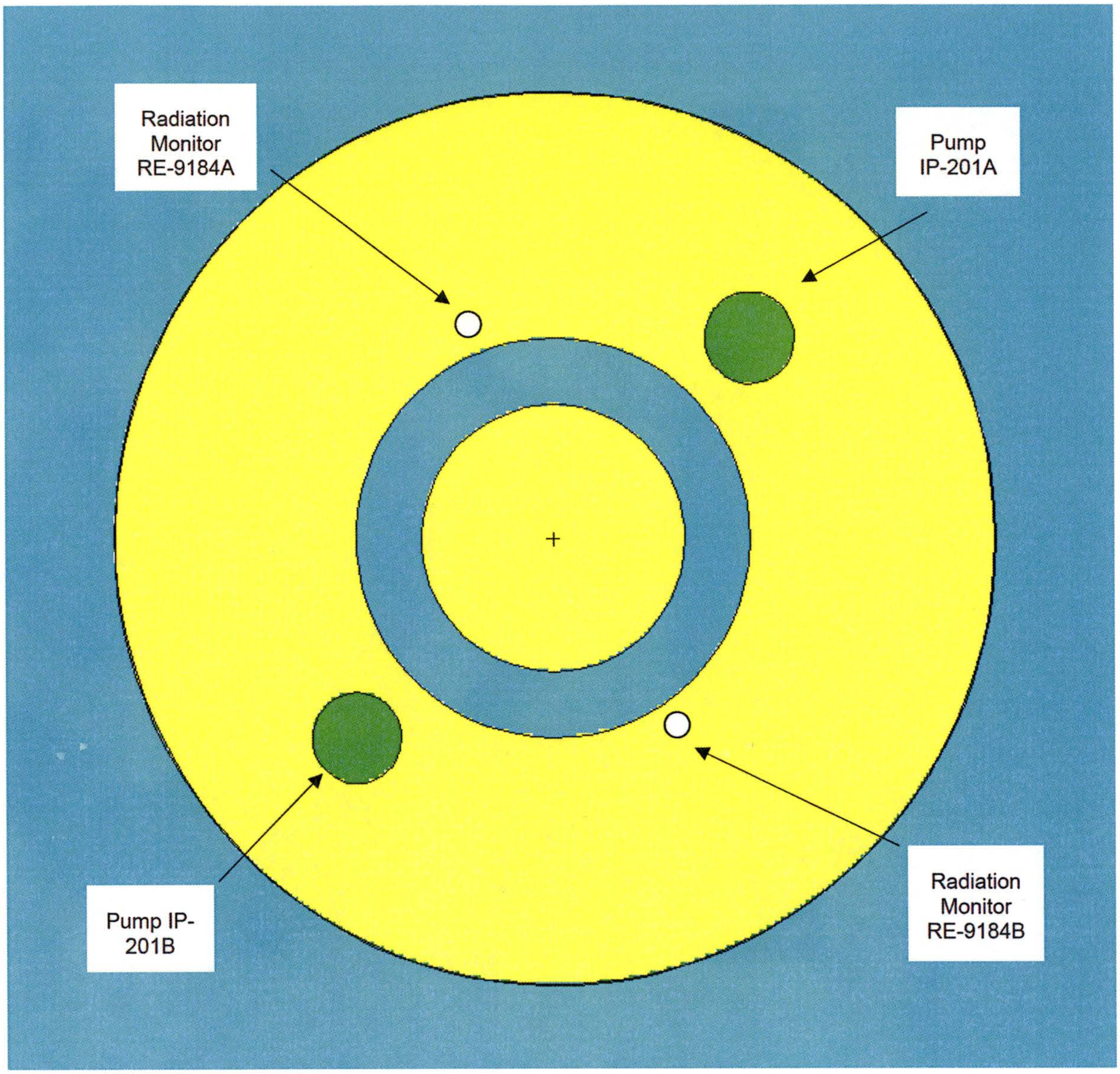


Figure 2 Vised Plot of Drywell and Reactor Building[3]



[3] Radiation monitors are not on the same plane shown above. They are included for visualization purposes only. The VISED Plot was rotated around the Z axis until the Recirculating Pumps were visible.

Figure 3 X-Y Vised Plot of Detectors and Reactor Recirculating Pumps at Elevation 760'-0" [4]



[4] Detectors are included for visualization purposes only.

Figure 4 MCNP Model Surface Cards[5]

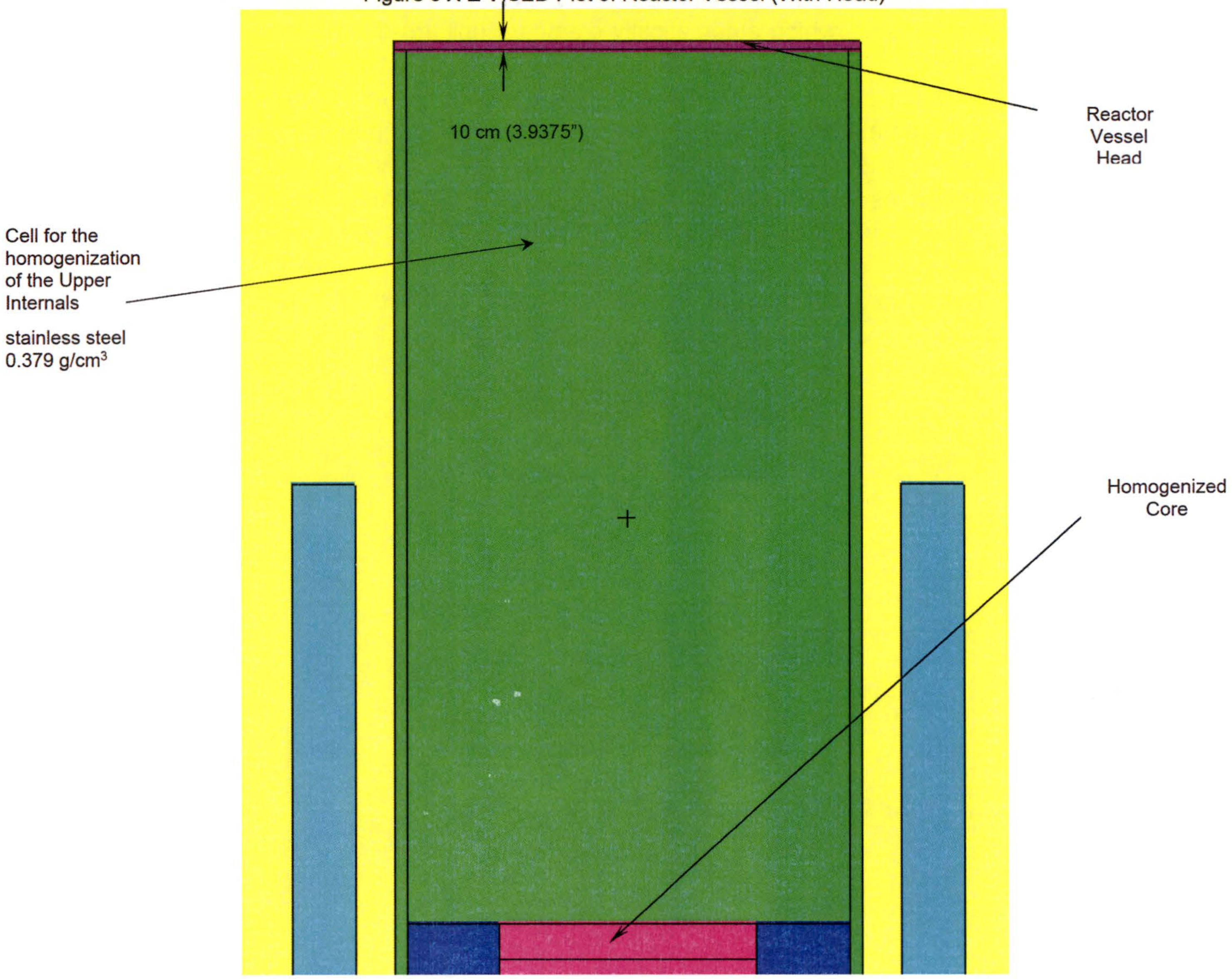
```
c surfaces
1 rcc 0 0 705.97 0 0 365.76 137.045                    $ Active Fuel Region
2 rcc 0 0 208.28 0 0 1776.73 235.43                    $ Reactor Pressure Vessel Inner Surface
3 rcc 0 0 195.58 0 0 1789.43 248.13                    $ Reactor Pressure Vessel Outer Surface
4 rpp -1120.14 1120.14 -1120.14 1120.14 -1120.14 821.86 $ Concrete Spher port drywell outer
5 so 960.12                                            $ Spher portion of drywell outer surface
6 so 958.21                                            $ Spher portion of drywell liner surface
7 pz -709.93                                           $ Bottom of Pedestal Elevation
8 rcc 0 0 -709.93 0 0 857.25 243.84                    $ Pedestal Inner Surface
9 rcc 0 0 -709.93 0 0 857.25 365.76                    $ Pedestal Outer Surface
81 rcc 0 0 147.32 0 0 1383.67 290.20                   $ Bio Shield Inner Surface
91 rcc 0 0 147.32 0 0 1383.67 356.24                   $ Bio Shield Outer Surface
82 rcc 365.76 365.76 -528.32 0 0 523.24 83.82          $ Recirc Pump IP-201A
92 rcc -365.76 -365.76 -528.32 0 0 523.24 83.82        $ Recirc Pump IP-201B
10 pz 195.58                                           $ Vessel 0
11 pz 821.86                                           $ Transition Spherical to Cylindrical
12 rcc 0 0 821.86 0 0 1889.59 518.16                   $ cylin port drywell concrete surface
13 rcc 0 0 821.86 0 0 1889.59 516.25                   $ cylin port drywell liner surface
14 rpp -693.42 693.42 -693.42 693.42 821.86 2711.45    $ Concrete cylin port drywell outer
15 pz 1071.73                                          $ Water Elevation Surface
16 pz 1985.01                                          $ Top of RPV (head level)
17 rpp -4267.2 4267.2 -4267.2 4267.2 2711.45 4006.85   $ Reactor building above drywell
18 rpp -4267.2 4267.2 -4267.2 4267.2 4006.85 4008.12   $ Reactor building roof
19 pz 147.32                                           $ Top of Ped Elevation/Bottom Bio Shield
20 pz 1530.99                                          $ Top of Ped Elevation/Bottom Bio Shield
28 rcc 0 0 1985.01 0 0 10.00 248.13                    $ Reactor Head
101 pz 742.546
102 pz 779.122
103 pz 815.698
104 pz 852.274
105 pz 888.85
106 pz 925.246
107 pz 962.002
108 pz 998.578
109 pz 1035.154
110 pz 1071.73
```

[5] The surface card for the MCNP model without the reactor vessel head does not have surface 28.

Figure 5 MCNP Model Cell Cards (No Head)

```
c cells
101 1 -4.49 -1 -101                     imp:p=256          $ Active Fuel Region
102 1 -4.49 -1 101 -102                 imp:p=128          $ Active Fuel Region
103 1 -4.49 -1 102 -103                 imp:p=64           $ Active Fuel Region
104 1 -4.49 -1 103 -104                 imp:p=32           $ Active Fuel Region
105 1 -4.49 -1 104 -105                 imp:p=16           $ Active Fuel Region
106 1 -4.49 -1 105 -106                 imp:p=8            $ Active Fuel Region
107 1 -4.49 -1 106 -107                 imp:p=4            $ Active Fuel Region
108 1 -4.49 -1 107 -108                 imp:p=3            $ Active Fuel Region
109 1 -4.49 -1 108 -109                 imp:p=2            $ Active Fuel Region
110 1 -4.49 -1 109 -110                 imp:p=1            $ Active Fuel Region
2 2 -0.9982 1 -2 -15                    imp:p=256          $ Water Region
3 3 -1.21E-03 15 -2                     imp:p=256          $ Air Region inside vessel
4 4 -7.94 2 -3 -16                      imp:p=256          $ RPV Shell
7 5 -2.3 5 -4                           imp:p=256          $ Concrete Surrounding RPV spherical
8 5 -2.3 -14 12                         imp:p=256          $ Concrete Surrounding RPV cylindrical
9 5 -2.3 -9 8 7 -19                     imp:p=256          $ Pedestal
91 5 -2.3 -91 81 19 -20                 imp:p=256          $ Bio Shield
10 5 -2.3 -6 -7                         imp:p=256          $ Concrete at bottom of pedestal
11 3 -1.21E-03 -8                       imp:p=256          $ Inside Pedestal Air
12 3 -1.21E-03 -6 7 -11 9 3
      #18 #19 #91                       imp:p=256          $ Inside Spherical portion Air
13 3 -1.21E-03 -13 3 #91                imp:p=256          $ Inside Cylindrical portion Air
14 3 -1.21E-03 -17                      imp:p=256          $ Reactor Building above drywell Air
15 4 -7.94 2 -18                        imp:p=256          $ Reactor Build Roof Stainless Steel
16 4 -7.94 6 -5 -11                     imp:p=256          $ Containment Liner Spherical portion
17 4 -7.94 13 -12                       imp:p=256          $ Containment Liner Cylin portion
18 4 -7.94 -82                          imp:p=256          $ Recirc Pump IP-201A
19 4 -7.94 -92                          imp:p=256          $ Recirc Pump IP-201B
999 0 1 #2 #3 #4 #7 #8 #9 #10 #11 #12 #13 #14
             #15 #16 #17 #18 #19 #91         imp:p=0             $ Problem Boundary
```

Figure 6 X-Z VISED Plot of Reactor Vessel (With Head)



7.5 MCNP Source Definition

The core source term is modeled as uniformly distributed throughout the homogenized core, and has an energy spectra based on the decayed core inventory (Section 7.1). Only the gamma source term is taken into account for this evaluation. The source term is generated shortly after shutdown, therefore, the fuel gamma source term will predominate, and the neutron-gamma and hardware activation source terms can be neglected (Assumption 4.7). The source is defined on the MCNP sdef card using distributions to define the particle location and energy. The radius of the core is defined with the rad parameter, which automatically creates a uniform distribution based on a cylindrical geometry. The ext and axs parameters define the direction and distance of the cylinder axis. These parameters combined define the core where the particles can be born. The erg parameter defines the energy spectrum of source particles, and is based on the results of the ORIGEN-S calculation discussed previously. This distribution is a histogram of energies represented by activities. These are automatically normalized by MCNP to create a probability distribution. The total activity is preserved in the tally multiplier. The MCNP source definition cards are shown below in Figure 7. The sb card is a source biasing card, which in this case biases the particle generation to the lower end of the core. This is a variance reduction technique to improve the statistical certainty in the results.

Figure 7 MCNP Source Definition Cards

```
sdef rad=d1 ext=d2 axs=0 0 1 erg=d8                          ←Source Definition Card
                                                                    -Radius = d1
                                                                    -Extent = d2
                                                                    -Axis = +Z
                                                                    -Energy = d8
si1 137.045                                                  ←Core Radius Distribution
si2 h 0 742.546 779.122 815.698 852.274 888.85 925.246 962.002    ←Core Axial Distribution
       998.578 1035.154 1071.73
sp2 0 1 1 1 1 1 1 1 1 1 1                                    ←Actual Uniform Distribution
sb2 0 1 1 0.1 0.1 0.1 0.01 0.01 0.01 0.001 0.001             ←Biased to Bot Distribution
c  Fuel Gamma Spectra
si8  h 1.000e-002 5.000e-002 1.000e-001 2.000e-001 3.000e-001 4.000e-001   ←Source Energy Groups
     6.000e-001 8.000e-001 1.000e+000 1.330e+000 1.660e+000 2.000e+000
     2.500e+000 3.000e+000 4.000e+000 5.000e+000 6.500e+000 8.000e+000
     1.000e+001 1.100e+001
sp8    0.00E+00 2.028E+19 6.572E+18 1.557E+19 9.672E+18 3.582E+18 7.837E+18 ←Source Emission on
                                                                              Energy Basis
       1.373E+19 2.132E+18 4.942E+17 3.579E+18 6.576E+16 7.518E+16 1.110E+17
       8.689E+14 1.553E+10 2.568E+08 3.792E+07 8.041E+06 4.352E+05
```

7.6 MCNP Tally Specification

The tallies used in this evaluation are point detectors placed at approximate locations of radiation monitors RE-9184A, and RE-9184B. Point detectors are chosen because they use quasi-deterministic dose calculations that will provide better results than surface or cell based tallies that require the particles to enter those regions. The inputs to this card are the coordinates of the dose points followed by an exclusion zone to reduce variance, as well as a multiplier card, which represents the total core activity in photons/sec. The tally cards are shown in Figure 8.

Figure 8 MCNP Tally Cards

```
f5c RE-9184A, and 9184B                        ←Tally Comment Card
f5:p -121.92  406.29 -184.15 20                ←Tally 5 (point detector)
182.88  -365.76 -184.15 20                     x     y     z    exclusion
fm5 8.370E+19                                  ← Tally Multiplier
                                                 (Total Activity)
```

In addition, the flux is multiplied by ANSI/ANS flux-dose conversion factors [Reference 3.4]. This is specified in MCNP using the de/df cards. These are shown in Figure 9.

Figure 9 ANSI/ANS-6.1.1-1977 Gamma Flux to Dose Conversion Factors

```
c ------------------------------------------------------------------
c ANSI/ANS-6.1.1-1977
c Gamma Flux to Dose Conversion Factors
c (mrem/hr)/(photons/cm2-s)
c ------------------------------------------------------------------
de0  .01 .03 .05 .07 .10 .15 .20 .25 .30 .35 .40               ←Energy Bins for Flux
     .45 .50 .55 .60 .65 .70 .80 1. 1.4 1.8 2.2                  to Dose Conversion
     2.6 2.8 3.25 3.75 4.25 4.75 5. 5.25 5.75 6.25
     6.75 7.5 9. 11.
df0  3.96E-03   5.82E-04   2.90E-04   2.58E-04   2.83E-04   3.79E-04   ←Energy Dependent
     5.01E-04   6.31E-04   7.59E-04   8.78E-04   9.85E-04   1.08E-03     Flux Multipliers
     1.17E-03   1.27E-03   1.36E-03   1.44E-03   1.52E-03   1.68E-03
     1.98E-03   2.51E-03   2.99E-03   3.42E-03   3.82E-03   4.01E-03
     4.41E-03   4.83E-03   5.23E-03   5.60E-03   5.80E-03   6.01E-03
     6.37E-03   6.74E-03   7.11E-03   7.66E-03   8.77E-03   1.03E-02
```

7.7 MCNP Material Cards

The MCNP material cards are provided in Figure 9. These are based on the compositions described in Table 6 or calculated in Section 7.2.

Figure 10 MCNP Material Cards

```
m1     92235     -0.0253         $ Homogenized Active Fuel Region
       92238     -0.6163
       8016      -0.1896
       40000     -0.1531
       50000     -0.0023
       24000     -0.0002
       26000     -0.0003
       1001      -0.0129
m2     1001 2 8016 1             $ Water
m3     6012 -0.000126            $ Air
       7014 -0.76508
       8016 -0.234793
m4     6000 -0.0008              $ SS 304
       14000 -0.01
       15031 -0.00045
       24000 -0.19
       25055 -0.02
       26000 -0.68375
       28000 -0.095
m5     26000 -0.014              $ Reg-Concrete
       1001 -0.01
       13027 -0.034
       20000 -0.044
       8016 -0.532
       14000 -0.337
       11023 -0.029
m6     6012 -0.01                 $ Carbon Steel
       26056 -0.99
```

7.8 Results

The dose rates are provided in Table 10 for the water level at the top of the fuel assemblies. The dose rate is slightly above the detectable response of 1 R/h (1E+03 mrem/h) for the no head configuration, and below the detectable response for the configuration with the reactor vessel head in place for one of the detectors. The sensitivity case shows that there is no significant impact due to reflection from the drywell cap.

Table 10 – Dose Rate Response (mrem/h)

Configuration	Dose Rate 1 RE-9184A	fsd[6]	Dose Rate 2 RE-9184B	fsd	Tally File
No Head	1.81E+03	10.81%	1.68E+03	7.31%	d0ndm
With Head	1.11E+03	10.16%	7.41E+02	8.24%	d0hgm
With Head (Sensitivity Case)	1.07E+03	15.27%	7.67E+02	15.51%	d0rdm

8.0 Computer Software

This calculation uses ORIGEN-S of the SCALE Version 6.1.2 code package [Reference 3.2] and MCNP Version 6.1.0 [Reference 3.3] in accordance with CSP 3.09.

[6] Fraction standard deviation.

9.0 Impact Assessment

This calculation is based on "realistic" assumptions for the purpose of declaring EALs, rather than typical conservative "bounding" type design basis analyses. The calculation results are intended to provide order of magnitude dose rates to assist Operations and Emergency Response personnel in determination of core uncovery in accordance with NEI 99-01 Rev. 6.

```
Origen output:
07/26/2017  04:19 PM            82,114 DAECEAL.OUT

MCNP output:

 Directory of \No head\
08/16/2017  09:13 AM           327,680 d0nao

 Directory of \With Head\
08/16/2017  10:01 AM         1,269,760 d0hgo

 Directory of \sensitivity\
08/16/2017  03:54 AM           286,720 d0rdo
```

Material	Isotope	Weight Fraction	Reference
Zry- 4	Zr	0.9823	[1]
(6.56 g/cm^3)	Sn	0.0145	
	Cr	0.001	
	Fe	0.0021	
	Hf	0.0001	
UO_2	U-235	0.0348	[1]
	U-238	0.8466	
	O	0.1186	
Air	C	0.0001	[1]
(1.21E-03 g/cm^3)	N	0.7651	
	O	0.2348	
Water	H	0.1111	[1]
(0.9982 g/cm^3)	O	0.8889	
SS-304	Fe	0.6838	[1]
(7.94 g/cm^3)	Cr	0.19	
	Ni	0.095	
	Mn	0.02	
	Si	0.01	
	C	0.0008	
	P	0.0004	
Concrete	O	0.532	[1]
(2.30 g/cm^3)	Si	0.337	
	Ca	0.044	
	Al	0.034	
	Na	0.029	
	Fe	0.014	
	H	0.01	
Carbon Steel	C	0.01	[1]
(7.82 g/cm^3)	Fe	0.99	

Material	Mass (KG)
UO_2	200.4
Zry- 4	42.92
Water	31.98

ZAID Number	Atom	Mass Fraction Active Fuel Region Homogenized
92235	U-235	0.0253
92238	U-238	0.6163
8016	O	0.1896
40000	Zr	0.1531
50000	Sn	0.0023
24000	Cr	0.0002
26000	Fe	0.0003
72000	Hf	0.0000
1001	H	0.0129
		1.0000

	A	B	C	D	E	F	G	H	I	J	K	L
1												
2		**Material**	**Isotope**	**Weight Fraction**	**Reference**		**Material**	**Mass (KG)**		**ZAID Number**	**Atom**	**Mass Fraction Active Fuel Region Homogenized**
3		**Zry- 4**	Zr	0.9823	[1]		**UO_2**	200.42		92235	U-235	=(H3/SUM(H3:H5))*D8
4		(6.56 g/cm^3)	Sn	0.0145			**Zry- 4**	42.92		92238	U-238	=(H3/SUM(H3:H5))*D9
5			Cr	0.001			**Water**	31.98		8016	O	=((H3/(SUM(H3:H5)))*D10)+((H5/(SUM(H3:H5))))*D15
6			Fe	0.0021						40000	Zr	=($H$4/SUM($H$3:$H$5))*D3
7			Hf	0.0001						50000	Sn	=($H$4/SUM($H$3:$H$5))*D4
8		**UO_2**	U-235	0.0348	[1]					24000	Cr	=($H$4/SUM($H$3:$H$5))*D5
9			U-238	0.8466						26000	Fe	=($H$4/SUM($H$3:$H$5))*D6
10			O	0.1186						72000	Hf	=($H$4/SUM($H$3:$H$5))*D7
11		**Air**	C	0.0001	[1]					1001	H	=(H5/SUM(H3:H5))*D14
12		(1.21E-03 g/cm^3)	N	0.7651								=SUM(L3:L11)
13			O	0.2348								
14		**Water**	H	0.1111	[1]							
15		(0.9982 g/cm^3)	O	0.8889								
16		**SS-304**	Fe	0.6838	[1]							
17		(7.94 g/cm^3)	Cr	0.19								
18			Ni	0.095								
19			Mn	0.02								
20			Si	0.01								
21			C	0.0008								
22			P	0.0004								
23		**Concrete**	O	0.532	[1]							
24		(2.30 g/cm^3)	Si	0.337								
25			Ca	0.044								
26			Al	0.034								
27			Na	0.029								
28			Fe	0.014								
29			H	0.01								
30		**Carbon Steel**	C	0.01	[1]							
31		(7.82 g/cm^3)	Fe	0.99								

```
=origens
0$$ a11 71 e t
BWR Source Term DAEC EAL Analysis
3$$  21 1 1 a4 27 a16 4 a33 19 e t
35$$ 0 t
54$$ a8 0 a11 2 e
56$$ 0 6 a6 1 a10 0 a13 63 3 3 0 2 0 e
57** 0  a3 1-16 e
95$$ 0 t
DAECEAL
Ci Source Terms
60** 0 24 40 50 60 70
61** 5r1-8 1+6 1+4
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
     3Z      0  1  0     1 0 0      1 0 0         3Z          6Z
     3Z      1  1  1     1 0 1      1 1 1         3Z          6Z
     3Z      1  1  1     1 1 1      1 1 1         3Z          6Z
81$$ 2 0 26 1 e
82$$ f2
83** 1.10E+07 1.00E+07 8.00E+06 6.50E+06 5.00E+06 4.00E+06 3.00E+06
     2.50E+06 2.00E+06 1.66E+06 1.33E+06 1.00E+06 8.00E+05 6.00E+05
     4.00E+05 3.00E+05 2.00E+05 1.00E+05 5.00E+04 1.00E+04 e
84** 2.00E+07 6.43E+06 3.00E+06 1.85E+06 1.40E+06 9.00E+05 4.00E+05
     1.00E+05 1.70E+04 3.00E+03 5.50E+02 1.00E+02 3.00E+01 1.00E+01
     3.05E+00 1.77E+00 1.30E+00 1.13E+00 1.00E+00 8.00E-01 4.00E-01
     3.25E-01 2.25E-01 1.00E-01 5.00E-02 3.00E-02 1.00E-02 1.00E-05 e
73$$ 561390 561400 581410 581430 581440 962420 551340 551360
    551370 531310 531320 531330 531340 531350 360831 360850
    360851 360870 360880 571400 571410 571420 420990 410950
    601470 932390 591430 942410 370860 451050 441030 441050
    441060 511270 511290 380890 380900 380910 380920 430991
    521270 521271 521290 521291 521311 521320 541311 541330
    541331 541350 541351 541380 390900 390910 390920 390930
    400950 400970
74** 9.06E+07 9.10E+07 8.39E+07 7.65E+07 6.77E+07 2.14E+06 8.99E+06
    2.85E+06 6.21E+06 5.11E+07 7.42E+07 1.04E+08 1.14E+08 9.90E+07
    5.83E+06 5.32E+05 1.18E+07 2.35E+07 3.25E+07 9.39E+07 8.28E+07
    8.05E+07 1.01E+08 8.60E+07 3.35E+07 1.09E+09 7.57E+07 8.15E+06
    1.01E+05 5.37E+07 8.30E+07 5.85E+07 2.96E+07 4.57E+07 1.66E+07
    4.61E+07 4.57E+06 5.76E+07 6.19E+07 8.36E+07 4.51E+06 7.59E+05
    1.58E+07 3.21E+06 1.03E+07 7.28E+07 6.98E+05 1.04E+08 3.29E+06
    2.72E+07 2.20E+07 8.72E+07 4.68E+07 6.06E+07 6.23E+07 4.82E+07
    8.49E+07 8.09E+07
75$$ 3 3 3 3 3 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 3 2 3 3 3 3
    3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3
t
56$$ f0 t
end
```

ENERCON Excellence—Every project. Every day.	Attachment 1 CALCULATION PREPARATION CHECKLIST	CALC NO.	NEE-323-CALC-002
		REV.	0

CHECKLIST ITEMS[1]	YES	NO	N/A
GENERAL REQUIREMENTS			
1. If the calculation is being performed to a client procedure, is the procedure being used the latest revision?	☐	☐	☒
The Calculation is performed in accordance with ENERCON procedures.			
2. Are the proper forms being used and are they the latest revision?	☒	☐	☐
The Calculation is performed in accordance with ENERCON procedures.			
3. Have the appropriate client review forms/checklists been completed?	☐	☐	☒
OAR will be performed after calculation submittal			
4. Are all pages properly identified with a calculation number, calculation revision and page number consistent with the requirements of the client's procedure?	☒	☐	☐
5. Is all information legible and reproducible?	☒	☐	☐
6. Is the calculation presented in a logical and orderly manner?	☒	☐	☐
7. Is there an existing calculation that should be revised or voided?	☐	☒	☐
There is no existing calculation that should be revised or voided.			
8. Is it possible to alter an existing calculation instead of preparing a new calculation for this situation?	☐	☒	☐
No existing calculation would be applicable.			
9. If an existing calculation is being used for design inputs, are the key design inputs, assumptions and engineering judgments used in that calculation valid and do they apply to the calculation revision being performed.	☐	☐	☒
No existing calculation is used for design inputs			
10. Is the format of the calculation consistent with applicable procedures and expectations?	☒	☐	☐
11. Were design input/output documents properly updated to reference this calculation?	☐	☐	☒
There are no design output documents.			
12. Can the calculation logic, methodology and presentation be properly understood without referring back to the originator for clarification?	☒	☐	☐
OBJECTIVE AND SCOPE			
13. Does the calculation provide a clear concise statement of the problem and objective of the calculation?	☒	☐	☐
14. Does the calculation provide a clear statement of quality classification?	☒	☐	☐
15. Is the reason for performing and the end use of the calculation understood?	☒	☐	☐
16. Does the calculation provide the basis for information found in the plant's license basis?	☐	☒	☐
This does not provide basis for license basis			
17. If so, is this documented in the calculation?	☐	☐	☒

ENERCON Excellence—Every project. Every day.	Attachment 1 CALCULATION PREPARATION CHECKLIST	CALC NO.	NEE-323-CALC-002
		REV.	0

CHECKLIST ITEMS[1]	YES	NO	N/A
See above			
18. Does the calculation provide the basis for information found in the plant's design basis documentation?	☐	☐	☒
This does not provide basis for design basis			
19. If so, is this documented in the calculation?	☐	☐	☒
See above			
20. Does the calculation otherwise support information found in the plant's design basis documentation?	☐	☒	☐
This does not provide support for information found in design basis documentation			
21. If so, is this documented in the calculation?	☐	☐	☒
See above			
22. Has the appropriate design or license basis documentation been revised, or has the change notice or change request documents being prepared for submittal?	☐	☐	☒
See above			
DESIGN INPUTS			
23. Are design inputs clearly identified?	☒	☐	☐
24. Are design inputs retrievable or have they been added as attachments?	☒	☐	☐
25. If Attachments are used as design inputs or assumptions are the Attachments traceable and verifiable?	☒	☐	☐
26. Are design inputs clearly distinguished from assumptions?	☒	☐	☐
27. Does the calculation rely on Attachments for design inputs or assumptions? If yes, are the attachments properly referenced in the calculation?	☒	☐	☐
The Design Information Transmittal is included as an Attachment is properly referenced in the calculation			
28. Are input sources (including industry codes and standards) appropriately selected and are they consistent with the quality classification and objective of the calculation?	☒	☐	☐
29. Are input sources (including industry codes and standards) consistent with the plant's design and license basis?	☒	☐	☐
30. If applicable, do design inputs adequately address actual plant conditions?	☒	☐	☐
31. Are input values reasonable and correctly applied?	☒	☐	☐
32. Are design input sources approved?	☒	☐	☐
The Design Information Transmittal contains information from a superseded calculation.			
33. Does the calculation reference the latest revision of the design input source?	☒	☐	☐
The calculation uses information from a superseded calculation. This information is provided in a Design Information Transmittal.			
34. Were all applicable plant operating modes considered?	☒	☐	☐
ASSUMPTIONS			

CHECKLIST ITEMS[1]	YES	NO	N/A
35. Are assumptions reasonable/appropriate to the objective?	☒	☐	☐
36. Is adequate justification/basis for all assumptions provided?	☒	☐	☐
37. Are any engineering judgments used?	☐	☒	☐
Engineering judgement not used as design input.			
38. Are engineering judgments clearly identified as such?	☐	☐	☒
Engineering Judgement is not used as a design input.			
39. If engineering judgments are utilized as design inputs, are they reasonable and can they be quantified or substantiated by reference to site or industry standards, engineering principles, physical laws or other appropriate criteria?	☐	☐	☒
Engineering Judgement is not used as a design input.			
METHODOLOGY			
40. Is the methodology used in the calculation described or implied in the plant's licensing basis?	☐	☐	☒
The scope of calculation is outside of plant licensing basis			
41. If the methodology used differs from that described in the plant's licensing basis, has the appropriate license document change notice been initiated?	☐	☐	☒
see above.			
42. Is the methodology used consistent with the stated objective?	☒	☐	☐
43. Is the methodology used appropriate when considering the quality classification of the calculation and intended use of the results?	☒	☐	☐
BODY OF CALCULATION			
44. Are equations used in the calculation consistent with recognized engineering practice and the plant's design and license basis?	☒	☐	☐
45. Is there reasonable justification provided for the use of equations not in common use?	☐	☐	☒
There are no uncommon equations used in the calculation.			
46. Are the mathematical operations performed properly and documented in a logical fashion?	☒	☐	☐
47. Is the math performed correctly?	☒	☐	☐
48. Have adjustment factors, uncertainties and empirical correlations used in the analysis been correctly applied?	☒	☐	☐
49. Has proper consideration been given to results that may be overly sensitive to very small changes in input?	☒	☐	☐
SOFTWARE/COMPUTER CODES			
50. Are computer codes or software languages used in the preparation of the calculation?	☒	☐	☐
MCNP and Scale are used			

CHECKLIST ITEMS[1]	YES	NO	N/A
51. Have the requirements of CSP 3.09 for use of computer codes or software languages, including verification of accuracy and applicability been met?	☒	☐	☐
52. Are the codes properly identified along with source vendor, organization, and revision level?	☒	☐	☐
53. Is the computer code applicable for the analysis being performed?	☒	☐	☐
54. If applicable, does the computer model adequately consider actual plant conditions?	☒	☐	☐
55. Are the inputs to the computer code clearly identified and consistent with the inputs and assumptions documented in the calculation?	☒	☐	☐
56. Is the computer output clearly identified?	☒	☐	☐
57. Does the computer output clearly identify the appropriate units?	☒	☐	☐
58. Are the computer outputs reasonable when compared to the inputs and what was expected?	☒	☐	☐
59. Was the computer output reviewed for ERROR or WARNING messages that could invalidate the results?	☒	☐	☐
RESULTS AND CONCLUSIONS			
60. Is adequate acceptance criteria specified?	☐	☐	☒
There is no acceptance criteria as discussed in calc.			
61. Are the stated acceptance criteria consistent with the purpose of the calculation, and intended use?	☐	☐	☒
See above			
62. Are the stated acceptance criteria consistent with the plant's design basis, applicable licensing commitments and industry codes, and standards?	☐	☐	☒
See above			
63. Do the calculation results and conclusions meet the stated acceptance criteria?	☐	☐	☒
See above.			
64. Are the results represented in the proper units with an appropriate tolerance, if applicable?	☒	☐	☐
65. Are the calculation results and conclusions reasonable when considered against the stated inputs and objectives?	☒	☐	☐
66. Is sufficient conservatism applied to the outputs and conclusions?	☒	☐	☐

ENERCON Excellence—Every project. Every day.	Attachment 1 CALCULATION PREPARATION CHECKLIST	CALC NO.	NEE-323-CALC-002
		REV.	0

CHECKLIST ITEMS[1]	YES	NO	N/A
67. Do the calculation results and conclusions affect any other calculations?	☐	☒	☐
No other calculations are affected by this calculation.			
68. If so, have the affected calculations been revised?	☐	☐	☒
No other calculations are affected by this calculation.			
69. Does the calculation contain any conceptual, unconfirmed or open assumptions requiring later confirmation?	☐	☒	☐
There are no open assumptions requiring confirmation later.			
70. If so, are they properly identified?	☐	☐	☒
There are no open assumptions requiring confirmation later.			
DESIGN REVIEW			
71. Have alternate calculation methods been used to verify calculation results?	☐	☒	☐
No a Design Review was performed.			

Note:

1. Where required, provide clarification/justification for answers to the questions in the space provided below each question. An explanation is required for any questions answered as "No' or "N/A".

Originator: Jay Bhatt

Print Name and Sign

Date

WAIT UNTIL

Spent Fuel Pool level cannot be maintained above **37 ft 1 in.**

SF/L-2 ☐

Maintain Spent Fuel Pool level above **36 ft**.

- If necessary, use alternate or external makeup sources (SEP 312).
- Use only systems not required for adequate core cooling.

SF/L-3 ☐

DISCUSSION

If spent fuel pool level cannot be restored and maintained above the low level alarm setpoint, an alternate control band is established above the higher of the spent fuel pool level LCO (36 ft.) or the Minimum Safe Operating Spent Fuel Pool Level (25.17 ft.). If necessary, normal spent fuel pool makeup may be augmented by one or more of the alternate and external sources listed in SEP 312.

The Minimum Safe Operating Spent Fuel Pool Level is generically defined to be the lowest water level providing adequate radiation shielding to (1) protect personnel performing local operations required by the EOPs and (2) allow unrestricted access to the main control room. At the DAEC, the Minimum Safe Operating Spent Fuel Pool Level is defined consistent with NEI 12-02 Level 2, described as the level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck. The corresponding spent fuel pool level at the DAEC is defined to be 25.17 ft., approximately 10 ft. above the top of the fuel racks.

Local Operations for Operating and Normal Shutdown/Cooldown

Procedure Section and Step	Step Action	If action not performed, does this prevent shutdown or cooldown?	Building	Elevation	Room	Mode
IPOI 3, Section 5, step (9)	Between 50% and 60% Reactor Power shutdown one Condensate and Reactor Feed Pump per OI 644 unless otherwise directed by CRS.	No. The Feed Pumps and Condensate Pumps can be tripped from the Control Room if necessary, and HPCI and/or RCIC can be used to maintain RPV Level.	N/A	N/A	N/A	N/A
IPOI 3, Section 5, step (10)	When turbine load is lowered to approximately 200 MWe, remove the 1E-18A[B] 2nd Stage Reheat System from service in accordance with OI 646, Extraction Steam.	No. 2nd Stage Reheat can be left in service and the turbine can be tripped if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (10)	Secure condensate demineralizers as directed by OI 639, Section 5.1.	No. Condensate Demineralizers will automatically go into the "hold" mode as power and flow are lowered.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (11)	Commence primary containment purge per OI 573.	No. This is only necessary if a Drywell entry is anticipated.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (13)	At the refueling bridge, verify that the Main Disconnect is closed and that the SYSTEM START pushbutton has been depressed.	No. Control rod insertion will not be inhibited.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (14)	Prior to disconnecting the generator from the grid, perform the following: (a) If needed, start up the Auxiliary Boiler per OI 727.	No. Aux Boiler is not required to accomplish shutdown.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (22)	Following Turbine Trip: (a) Verify that Reactor Coolant Chloride and Conductivity analyses have been performed. (b) Operate the Turbine Lube Oil and Turning Gear System per OI 693.3. (c) Shut down the generator per OI 698. (d) Shut down the turbine per OI 693.1.	No. These systems can be left in service if necessary.	N/A	N/A	N/A	N/A

Procedure Section and Step	Step Action	If action not performed, does this prevent shutdown or cooldown?	Building	Elevation	Room	Mode
IPOI 4, Section 3 step (24)	Shut down the following generator support systems, as desired: Isolated Phase Bus Cooling - OI 698, Stator Water Cooling - OI 697, H_2 Seal Oil - OI 695.1, H_2 and CO_2 Gas - OI 695.2	No. These systems can be left in service if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (26)	Secure hydrogen, oxygen and/or air injection per OI 563, Hydrogen Water Chemistry.	No. The Hydrogen Water Chemistry System will secure itself if left in service.	N/A	N/A	N/A	N/A
IPOI 4, Section 3 step (27)	As directed by the CRS, perform the following steps as necessary to limit reactor vessel depressurization following the reactor scram: (b) Start 1P32 Mechanical Vacuum Pump per OI 691. (c) Secure the SJAEs and Offgas per OI 691 and OI 672.	No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (6)	For the remainder of this section use the following methods as necessary to cooldown and depressurize the reactor vessel to maintain a controlled cooldown rate less than the TS Limit of 100°F in any 1 hour period. (a) Use the Main Turbine Bypass Valve to control cooldown per OI 693.1 Section 4.5 if available, (b) If desired cooldown with RCIC per OI 150 (preferred method if MSIVs are closed), (c) If desired cooldown with HPCI per OI 152 (RCIC may become inadequate as pressure lowers) (d) Control steam flow from the reactor vessel to the main condenser through steam seals and steam drains, (e) Secure steam seals per OI 692 as required to limit cooldown after the turbine is on the jack and vacuum is broken.	(a) No. The MSIVs can be closed if necessary to limit plant cooldown rate. (b) No – operated from the Control Room (c) No – Operated from the Control Room (d) No. The MSIVs can be closed if necessary to limit plant cooldown rate. (e) No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A

Procedure Section and Step	Step Action	If action not performed, does this prevent shutdown or cooldown?	Building	Elevation	Room	Mode
IPOI 4, Section 4 step (7)	As plant cooldown continues perform the following: (NA if MSIVs are closed) (a) Control steam seal pressure 3 to 4 psig using MO-1169, MAIN STEAM SUPPLY, MO-1170, REGULATOR BYPASS and/or MO-1171, MANUAL UNLOADER on 1C07, (b) Start 1P-32 MECHANICAL VACUUM PUMP per OI 691, (c) When reactor pressure approaches 500 psig or cooldown rate cannot be controlled within the limit, then secure SJAEs and Offgas System per OI 691 and OI 672, respectively, if not previously secured, (d) If not using EHC Pressure Set to control plant cooldown, then at 1C07, use the PRESSURE SET ADJUST pushbuttons to maintain A[B] PRESSURE SET DEMAND between 150 and 50 psig above reactor pressure as reactor pressure decreases. Otherwise, N/A.	No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (8)	At approximately 400 psig, secure the operating feed pump per OI 644.	No. The Feed Pumps and Condensate Pumps can be tripped from the Control Room if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (9)	When RHR Shutdown Cooling Isolation Interlocks can be reset (approximately 100 psig), reset the isolation, then initiate Shutdown Cooling per OI 149.	No, this system can be placed in service from the Control Room if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (10)	Perform the following after the turbine trip, if needed: (a) Verify that Reactor Coolant Chloride and Conductivity analysis has been performed, (b) Operate the Turbine Lube Oil and Turning Gear System per OI 693.3, (c) Shutdown the Main Generator per OI 698, (d) Shutdown the Main Turbine per OI 693.1.	No. These systems can be left in service if necessary.	N/A	N/A	N/A	N/A

Procedure Section and Step	Step Action	If action not performed, does this prevent shutdown or cooldown?	Building	Elevation	Room	Mode
IPOI 4, Section 4 step (11)	Shutdown the following systems as directed by the CRS/OSM. (a) Isolated Phase Bus Cooling per OI 698, (b) Stator Water Cooling per OI 697, (c) H_2 Seal Oil per OI 695.1, (d) H_2 and CO_2 Gas per OI 695.2, (e) Secure SJAEs per OI 691 and Offgas per OI 672 if not previously performed.	No. These systems can be left in service if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (12)	Perform the following at approximately 50 psig: (a) Close the BYPASS VALVE OPENING JACK SELECTOR, (b) Line up and place RFP Stuffing Box Pump 1P-134 in operation to maintain Seal Water Drain Tank 1T-135 level.	No. The Feed Pumps and Condensate Pumps can be tripped from the Control Room if necessary.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (13)	When steam seal pressure cannot be maintained or the turbine shaft has cooled per OI 693.3, open Condenser Vacuum Breaker valves V-03-67 and V-03-73.	No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (14)	Secure MECHANICAL VACUUM PUMP 1P-32 when no longer required per OI 691.	No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (15)	When the condenser is at atmospheric pressure, secure the Turbine Steam Seal System per OI 692.	No. The MSIVs can be closed if necessary to limit plant cooldown rate.	N/A	N/A	N/A	N/A
IPOI 4, Section 4 step (18)	Shut down the operating condensate pump per OI 644 when no longer required for RPV Level Control or Hotwell cleanup recirculation.	No. The Feed Pumps and Condensate Pumps can be tripped from the Control Room if necessary.	N/A	N/A	N/A	N/A

Conclusion of manual action evaluation for EALs RA3 and HA5 is shown below:

EALs RA3 and HA5 are not applicable to DAEC because the evaluation has shown that there are no rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. All areas outside the Control Room that contain equipment necessary for normal plant operation, cooldown and shutdown do not require physical access to operate.

Development of EAL Threshold values from NEE-323-CALC-005

Calculated values are provided in Calc-005 as shown below.

Table 3 – Recommended RA1, RS1, and RG1 EAL Thresholds (Modes 1, 2, 3)

Release Point	RA1 µCi/cc	RS1 µCi/cc	RG1 µCi/cc
Turbine Building	1.58E-02	1.58E-01	1.58E+00
Reactor Building	1.22E-02	1.22E-01	1.22E+00
Offgas Stack	4.39E+01	4.39E+02	4.39E+03
LLRPSF	1.51E-02	1.51E-01	1.51E+00*

Table 4 – Recommended RA1, RS1, and RG1 EAL Thresholds (Modes 4, 5)

Release Point	RA1 µCi/cc	RS1 µCi/cc	RG1 µCi/cc
Turbine Building	1.30E-02	1.30E-01	1.30E+00
Reactor Building	1.01E-02	1.01E-01	1.01E+00
Offgas Stack	4.52E+01	4.52E+02	4.52E+03
LLRPSF	1.25E-02	1.25E-01	1.25E+00*

* Per Design Input 5.8 the results in EAL threshold values exceed the range of the monitor.

The following table of threshold values was developed for use in the DAEC EAL scheme by averaging the separate Mode 1-3 and Mode 4-5 thresholds from Calc-005, and then rounding the average values for ease of EAL evaluator use, as well as to provide a step-wise progression through the emergency classification. Resulting values are shown in the Alert, SAE, and GE columns below:

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8)	1.1E+00 uci/cc	1.1E-01 uci/cc	1.1E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Turbine Building ventilation rad monitor (Kaman 1/2)	1.4E+00 uci/cc	1.4E-01 uci/cc	1.4E-02 uci/cc	8.0E-04 uci/cc
Gaseous	Offgas Stack rad monitor (Kaman 9/10)	4.5E+03 uci/cc	4.5E+02 uci/cc	4.5E+01 uci/cc	2.0E-01 uci/cc
Gaseous	LLRPSF rad monitor (Kaman 12)	---	1.4E-01 uci/cc	1.4E-02 uci/cc	1.2E-03 uci/cc
Liquid	GSW rad monitor (RIS-4767)	---	---	1.7E+04 cps	1.5E+03 cps
Liquid	RHRSW & ESW rad monitor (RM-1997)	---	---	1.2E+04 cps	8.4E+02 cps
Liquid	RHRSW & ESW Rupture Disc rad monitor (RM-4268)	---	---	1.8E+04 cps	1.0E+03 cps

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Title:	Revised Gaseous Radiological EALs per NEI 99-01 Rev. 06	**Client:** Duane Arnold Energy Center **Project Identifier:** NEE-323

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions, including preliminary information, that require confirmation? (If **YES**, identify the assumptions.)	☐	☒
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, identify the design verified calculation.) **Design Verified Calculation No.** _____	☐	☒
3	Does this calculation supersede an existing Calculation? (If **YES**, identify the design verified calculation.) **Superseded Calculation No.** _____	☐	☒

Scope of Revision:
Initial Issue

Revision Impact on Results:
Initial Issue

Study Calculation ☐ **Final Calculation** ☒

Safety-Related ☐ **Non-Safety-Related** ☒

(Print Name and Sign)

Originator: Ryan Skaggs	**Date:** 12/14/17
Design Verifier[1] (Reviewer if NSR): Jay Bhatt	**Date:** 12/14/17
Approver: Zachary Rose	**Date:** 12/14/17

Note 1: For non-safety-related calculation, design verification can be substituted by review.

ENERCON Excellence—Every project. Every day.	CALCULATION REVISION STATUS SHEET	CALC NO. NEE-323-CALC-005
		REV. 00

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
00	12/14/17	Initial Issue

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
All	00		

APPENDIX/ATTACHMENT REVISION STATUS

APPENDIX NO.	NO. OF PAGES	REVISION NO.	ATTACHMENT NO.	NO. OF PAGES	REVISION NO.
A	8	00	1	4	00

1.0 Purpose and Scope

The DAEC site is implementing new requirements of Revision 6 to the Document NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors." One of the changes included in Revision 6 to NEI 99-01 is a new basis for the Emergency Action Level (EAL) RA1. The requirements for RS1 and RG1 did not change from NEI 99-01 Rev. 05 with the implementation of NEI 99-01, Rev. 06.

The following table is extracted from Section 6 of **Revision 6** to NEI 99-01:

ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
AA1 Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE. *Op. Modes: All*	**AS1** Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE. *Op. Modes: All*	**AG1** Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE. *Op. Modes: All*

AA1, AS1, AG1 compares to DAEC terminology RA1, RS1, RG1, respectively. This calculation determines the effluent radiation monitor readings that correspond to the RA1, RS1, and RG1 thresholds.

2.0 Summary of Results and Conclusions

The results below show the RA1 EAL release concentration thresholds and associated dose rates for each release point for a decay time of five hours and 36 hours. The highlighted dose indicates which threshold was met at the release concentration.

Table 1 – RA1 EAL Release Concentration Thresholds (Decay = 5 hours (Mode 1, 2, 3))

Release Point	Release Concentration µCi/cc	CEDE mrem	EDE mrem	TEDE mrem	CDE Thyroid mrem
Turbine Building	1.58E-02	2.38	0.39	2.77	50.0
Reactor Building	1.22E-02	2.37**	0.39	2.76	49.8**
Offgas Stack	4.39E+01	1.96	8.05*	10.00	41.1
Low-Level Radwaste Processing and Storage Facility (LLRPSF)	1.51E-02	2.37	0.39	2.76	49.7

* Calculation of this value was demonstrated in Section 7.3

** Calculation of this value was demonstrated in Section 7.4

Table 2 – RA1 EAL Release Concentration Thresholds (Decay = 36 hours (Mode 4, 5))

Release Point	Release Concentration µCi/cc	CEDE mrem	EDE mrem	TEDE mrem	CDE Thyroid mrem
Turbine Building	1.30E-02	2.59	0.07	2.67	49.7
Reactor Building	1.01E-02	2.60	0.07	2.68	49.9
Offgas Stack	4.52E+01	2.61	1.41	4.02	50.0
LLRPSF	1.25E-02	2.60	0.07	2.67	49.8

Resultant EAL thresholds:

The tables below show the release concentration threshold for RA1, RS1, and RG1 based on the results above for both a decay time of five hours and a decay time of 36 hours.

From Section 1.0:

RS1 thresholds are 10 times larger than those for RA1

RG1 thresholds are 100 times larger than those for RA1

Table 3 – Recommended RA1, RS1, and RG1 EAL Thresholds (Modes 1, 2, 3)

Release Point	RA1 µCi/cc	RS1 µCi/cc	RG1 µCi/cc
Turbine Building	1.58E-02	1.58E-01	1.58E+00
Reactor Building	1.22E-02	1.22E-01	1.22E+00
Offgas Stack	4.39E+01	4.39E+02	4.39E+03
LLRPSF	1.51E-02	1.51E-01	1.51E+00*

Table 4 – Recommended RA1, RS1, and RG1 EAL Thresholds (Modes 4, 5)

Release Point	RA1 µCi/cc	RS1 µCi/cc	RG1 µCi/cc
Turbine Building	1.30E-02	1.30E-01	1.30E+00
Reactor Building	1.01E-02	1.01E-01	1.01E+00
Offgas Stack	4.52E+01	4.52E+02	4.52E+03
LLRPSF	1.25E-02	1.25E-01	1.25E+00*

* Per Design Input 5.8 the results in EAL threshold values exceed the range of the monitor.

3.0 References

3.1 NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors", Nuclear Energy Institute, November 2012.

3.2 NUREG-1940, RASCAL 4: Description of Models and Methods, United States Nuclear Regulatory Commission, Office of Nuclear Security and Incident Response, 2012.

3.3 NUREG-1940 Supplement 1, RASCAL 4.3: Description of Models and Methods, United States Nuclear Regulatory Commission, Office of Nuclear Security and Incident Response, 2015.

3.4 NUREG-1228, Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents, United States Nuclear Regulatory Commission, Division of Operational Assessment, 1988.

3.5 NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, 1995.

3.6 DAEC UFSAR, Chapter 15-0.

3.7 DAEC UFSAR, Chapter 15-2.

3.8 DAEC Offsite Dose Assessment Manual (ODAM).

3.9 Plant Chemistry Procedure PCP 8.3, Alarm Setpoints and Background Determination for KAMAN Normal Range Monitors.

3.10 DAEC Nuclear Station HRN-HRH Radiation Monitor Operation, Maintenance and Troubleshooting Manual, ©2000, by Engineering Solutions, 310 Luchana Drive, Litchfield Park, Arizona.

3.11 DAEC Emergency Plan, Section 'I', Rev. 27.

3.12 Federal Guidance Report No. 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion Office of Radiation and Indoor Air, 1999.

3.13 Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water, and Soil, 1993.

3.14 *Table of Nuclides,* *http://atom.kaeri.re.kr:8080/ton/index.html,* *retrieved 10/10/17.*

4.0 Assumptions

The following are assumptions about the receptor:

- No credit is taken for radiation shielding provided by structures.
- No decay in-transit is assumed during the time elapsed between the release point and the receptor.

Both of the above assumptions are acceptable because they will result in a higher dose to the receptor and conservatively lower thresholds.

5.0 Design Inputs

5.1 Core Inventory

The assumed isotopic mixture in Table 5 is taken from Table 1-1 of NUREG-1940.

The core inventory (curies per megawatts thermal) in the table is based on calculations made by the NRC staff in December 2003 using the SAS2H control module of SCALE (Standardized Computer Analyses for Licensing Evaluation), Version 4.4a.

Table 5 – Isotopic Mixture

NUCLIDE	CORE INVENTORY (Ci/MWt)	NUCLIDE	CORE INVENTORY (Ci/MWt)	NUCLIDE	CORE INVENTORY (Ci/MWt)
Ba-139	4.74E+04	La-141	4.33E+04	Te-127	2.36E+03
Ba-140	4.76E+04	La-142	4.21E+04	Te-127m	3.97E+02
Ce-141	4.39E+04	Mo-99	5.30E+04	Te-129	8.26E+03
Ce-143	4.00E+04	Nb-95	4.50E+04	Te-129m	1.68E+03
Ce-144*	3.54E+04	Nd-147	1.75E+04	Te-131m	5.41E+03
Cm-242	1.12E+03	Np-239	5.69E+05	Te-132	3.81E+04
Cs-134	4.70E+03	Pr-143	3.96E+04	Xe-131m	3.65E+02
Cs-136	1.49E+03	Pu-241	4.26E+03	Xe-133	5.43E+04
Cs-137*	3.25E+03	Rb-86	5.29E+01	Xe-133m	1.72E+03
I-131	2.67E+04	Rh-105	2.81E+04	Xe-135	1.42E+04
I-132	3.88E+04	Ru-103	4.34E+04	Xe-135m	1.15E+04
I-133	5.42E+04	Ru-105	3.06E+04	Xe-138	4.56E+04
I-134	5.98E+04	Ru-106*	1.55E+04	Y-90	2.45E+03
I-135	5.18E+04	Sb-127	2.39E+03	Y-91	3.17E+04
Kr-83m	3.05E+03	Sb-129	8.68E+03	Y-92	3.26E+04
Kr-85	2.78E+02	Sr-89	2.41E+04	Y-93	2.52E+04
Kr-85m	6.17E+03	Sr-90	2.39E+03	Zr-95	4.44E+04
Kr-87	1.23E+04	Sr-91	3.01E+04	Zr-97*	4.23E+04
Kr-88	1.70E+04	Sr-92	3.24E+04		
La-140	4.91E+04	Tc-99m	4.37E+04		

5.2 Release Fraction

Table 6 displays release fractions as a function of time taken from Table 1-4 which references Table 3-12 of NUREG-1465.

Table 6 – Release Fraction

NUCLIDE GROUP	BWR CORE INVENTORY RELEASE FRACTION		
	Cladding Failure (Gap Release Phase) (0.5-hour duration)	Core Melt Phase (In-Vessel Phase) (1.5-hour duration)	Postvessel Melt-Through Phase (Ex-Vessel Phase) (3.0-hour duration)
Noble gases (Kr, Xe)	0.05	0.95	0
Halogens (I, Br)	0.05	0.25	0.30
Alkali metals (Cs, Rb)	0.05	0.20	0.35
Tellurium group (Te, Sb, Se)	0	0.05	0.25
Barium, strontium (Ba, Sr)	0	0.02	0.1
Noble metals (Ru, Rh, Pd, Mo, Tc, Co)	0	0.0025	0.0025
Cerium group (Ce, Pu, Np)	0	0.0005	0.005
Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)	0	0.0002	0.005

*Reference: Table 3-12 from NUREG-1465.

5.3 Gaseous Dispersion Factors

The dispersion factors are taken from the ODAM Section 3.

Table 7 – Dispersion Factors

	Dose due to Plume/Submersion ODAM Sections 3.5.2.1 and 3.9	Organ Dose Due to Particulates and Iodine ODAM Section 3.8
Offgas Stack	2.8E-7 sec/m^3	3.1E-7 sec/m^3
Building Vents	4.3E-6 sec/m^3	3.9E-6 sec/m^3

5.4 Isotopic half-lives

Isotopic half-lives are taken from NUREG-1940, Supplement 1. For those isotopes missing from that list, denoted by *, half-lives were obtained from the following website which is maintained by the Korea Atomic Energy Research Institute:

http://atom.kaeri.re.kr:8080/ton/index.html

Table 8 contains the half-lives and calculated λ (lambda) values.

Table 8 – Half-lives and Decay Constants

Isotope	T 1/2	T 1/2 units	T 1/2 Hours	Decay Lambda hrs^{-1}
Ba-139	0.0574	days	1.38E+00	5.03E-01
Ba-140	12.7	days	3.05E+02	2.27E-03
Ce-141	32.5	days	7.80E+02	8.89E-04
Ce-143	1.38	days	3.31E+01	2.09E-02
Ce-144	284	days	6.82E+03	1.02E-04
Cm-242	163	days	3.91E+03	1.77E-04
Cs-134	753	days	1.81E+04	3.84E-05
Cs-136	13.1	days	3.14E+02	2.20E-03
Cs-137	11000	days	2.64E+05	2.63E-06
I-131	8.04	days	1.93E+02	3.59E-03
I-132	0.0958	days	2.30E+00	3.01E-01
I-133	0.867	days	2.08E+01	3.33E-02
I-134	0.0365	days	8.76E-01	7.91E-01
I-135	0.275	days	6.60E+00	1.05E-01
Kr-83m*	1.83	hours	1.83E+00	3.79E-01
Kr-85	3910	days	9.38E+04	7.39E-06
Kr-85m	0.187	days	4.49E+00	1.54E-01
Kr-87	0.053	days	1.27E+00	5.45E-01
Kr-88	0.118	days	2.83E+00	2.45E-01
La-140	1.68	days	4.03E+01	1.72E-02
La-141	0.164	days	3.94E+00	1.76E-01
La-142	0.0642	days	1.54E+00	4.50E-01
Mo-99	2.75	days	6.60E+01	1.05E-02
Nb-95	35.2	days	8.45E+02	8.20E-04
Nd-147	11	days	2.64E+02	2.63E-03
Np-239	2.36	days	5.66E+01	1.22E-02
Pr-143	13.6	days	3.26E+02	2.12E-03
Pu-241	5260	days	1.26E+05	5.49E-06
Rb-86	18.7	days	4.49E+02	1.54E-03
Rh-105	1.47	days	3.53E+01	1.96E-02
Ru-103	39.3	days	9.43E+02	7.35E-04
Ru-105	0.185	days	4.44E+00	1.56E-01
Ru-106	368	days	8.83E+03	7.85E-05
Sb-127	3.85	days	9.24E+01	7.50E-03
Sb-129*	4.4	hours	4.40E+00	1.58E-01
Sr-89	50.5	days	1.21E+03	5.72E-04
Sr-90	10600	days	2.54E+05	2.72E-06
Sr-91	0.396	days	9.50E+00	7.29E-02
Sr-92	0.113	days	2.71E+00	2.56E-01
Tc-99m	0.251	days	6.02E+00	1.15E-01
Te-127	0.39	days	9.36E+00	7.41E-02
Te-127m	109	days	2.62E+03	2.65E-04
Te-129	0.0483	days	1.16E+00	5.98E-01
Te-129m	33.6	days	8.06E+02	8.60E-04

Isotope	T 1/2	T 1/2 units	T 1/2 Hours	Decay Lambda hrs^{-1}
Te-131m	1.25	days	3.00E+01	2.31E-02
Te-132	3.26	days	7.82E+01	8.86E-03
Xe-131m*	11.934	days	2.86E+02	2.42E-03
Xe-133	5.25	days	1.26E+02	5.50E-03
Xe-133m*	2.19	days	5.26E+01	1.32E-02
Xe-135	0.379	days	9.10E+00	7.62E-02
Xe-135m*	15.29	minutes	2.55E-01	2.72E+00
Xe-138*	14.08	minutes	2.35E-01	2.95E+00
Y-90	2.67	days	6.41E+01	1.08E-02
Y-91	58.5	days	1.40E+03	4.94E-04
Y-92	0.148	days	3.55E+00	1.95E-01
Y-93	0.421	days	1.01E+01	6.86E-02
Zr-95	64	days	1.54E+03	4.51E-04
Zr-97	0.704	days	1.69E+01	4.10E-02

5.5 Reduction Factor for Sprays

NUREG-1940 Table 1-11 states that when sprays are used for longer than 1.75 hours (but less than 2.25 hours), the following factor is applied to reduce all of the particulate and iodine species.

$RF_s = Exp^{(-0.64t)}$

Where t = the amount of times sprays are in service.

Note: This reduction factor does not apply to the noble gases.

For this calculation, sprays are used for a total of 2 hours as described in Section 6.1. The reduction factor is:

$RF_s = e^{(-0.64*2)} = 0.278$

5.6 Standby Gas Treatment Filters

NUREG-1940 allows a reduction factor of 0.01 for filters like the standby gas treatment (SBGT) system. This factor is only applied to releases from the Offgas Stack.

$RF_F = 0.01$

5.7 Secondary Containment

NUREG-1228 provides a reduction factor for natural removal through settling and plate-out in the secondary containment. For a 0.5 hour holdup period, that reduction factor is 0.4. This factor is applied to the building vent releases but not the release from the Offgas Stack.

$RF_{sc} = 0.4$

5.8 Monitor Range and Exhaust Flow Rates

Table 9 is developed from the DAEC Emergency Plan Section "I", ODAM Figure 3-1, and Procedure PCP 8.3.

Table 9 – Monitor Range and Exhaust Flow Rates

Release Point	Monitor Common Name	Equipment ID	Monitor Range µCi/cc	Release Flow CFM
Turbine Building	KAMAN 1/2	RE-5945 / RE-5946	1E-7 to 1E+5	72,000
Reactor Building	KAMAN 3/4 KAMAN 5/6 KAMAN 7/8	RE-7645, RE-7644 RE-7647, RE-7646 RE-7649, RE-7648	1E-7 to 1E+5	93,000
Offgas Stack	KAMAN 9/10	RE-4176, RE-4175	1E-7 to 1E+5	10,000
LLRPSF	KAMAN 12	RE-8801	1E-7 to 3E-1	75,000

5.9 Breathing Rate

From NUREG-1940 and FGR11, the breathing rate is 3.33E-4 m^3/second.

5.10 Exposure-to-Dose Conversion Factors for Inhalation

The "Exposure-to-Dose Conversion Factors for Inhalation" by radionuclide provided in FGR11 Table 2.1 allow the determination of the committed dose equivalent to the thyroid and the effective dose equivalent per unit per unit intake, and are shown in Table 11.

5.11 Dose Coefficients for Air Submersion

The dose coefficients in Sv/Bq*s*m^{-3} from being submersed in air for each radionuclide to an effective dose are taken from Table III.1 of FGR12, and are shown in Table 11.

6.0 Methodology

This calculation will equate a radioactive material release rate as measured at the gaseous effluent radiation monitors with the dose received to a member of the public at an offsite location. The relationship is highly influenced by the mixture of radioisotopes in the effluent and the dispersion of gases after they have left the facility. Primary guidance is provided by NUREG-1940 and NUREG-1228.

6.1 Scenario

The following generalized timeline is used to determine the phenomenon that can affect the mixture of radioisotopes in effluent. This scenario is realistic, but bounds an event that could occur in a shorter total time frame:

- T= 0 hr. Major recirculating system line break occurs. Reactor is shut down.
- T= 1 hr. Core is uncovered.
- T= 1 hr. Sprays are initiated.
- T = 2 hrs. Core is covered.
- T= 4.5 hrs. A catastrophic event causes damage to the drywell and the secondary containment.
  - The gaseous mixture from the Drywell spreads into the Reactor Building, Turbine Building, and LLRPSF.
  - Mean average holdup time of the gas in these buildings is 0.5 hours.

Scenario timing will affect the mixture of radioisotopes and is summarized here:

- The core is uncovered for 1 hour.
- Core/Drywell Sprays are running for a total of 2 hours.
- Primary Containment integrity is maintained for 4 hours.
- Source holdup time in secondary containment is 0.5 hours.
- Source decay time from shutdown to the release point is 5 hours.
- When the reactor is in mode 4 or 5, the total decay time is 36 hours.

Other Factors:

- The flow rates from the effluent exhaust points are listed in Design Input 5.8.
- The gaseous effluent radiation monitors are equally efficient for the monitoring of noble gases, particulates, and iodines.
- All releases from the Offgas Stack are filtered by the Standby Gas Treatment system.
- Removal of particulates and iodines by natural process during holdup in secondary containment are credited for releases from the building vents only.

6.2 Receptor

The receptor is an adult located at the ODAM-described location of minimal dispersion who is exposed to the radioactive release for one hour. Due to this relatively short duration, the only exposure pathways are inhalation and submersion. Assumptions related to the receptor are found in Section 4.0.

6.3 General Approach

With a given mixture of radionuclides, the dose received by an individual offsite is a function of the gross activity present in the gaseous mixture.

The resultant dose received by an offsite receptor is dependent not only on the gross radioactivity levels of the effluent but also upon the isotopic mixture present in the gas. This calculation predicts the relative contribution of each radionuclide to the gross radiation monitored by the effluent monitor.

With the fractionation of the mixture of radionuclides understood, a given gross output reading ($\mu Ci/cm^3$) from an effluent radiation monitor can be scaled to determine the concentration of each isotope present in the effluent.

The calculation then uses default dispersion factors described in the Offsite Dose Assessment Manual to determine the resultant concentration of radionuclides to which an individual offsite would be exposed.

Dose conversion factors provided in Federal Guidance Report 11 (FGR11) and 12 (FGR12) are used to determine the dose (mrem) to an individual offsite due to their exposure to the gaseous mixture of radionuclides.

With the given radionuclide mixture and dispersion factors understood, an iterative process can be used to relate the effluent monitor reading to a target offsite dose.

Two types of radiation dose are calculated: 1) TEDE and 2) CDE Thyroid.

CDE or Committed Dose Equivalent is the radiation dose to a specific organ due to an uptake of radioactive material. In this case, the uptake is limited to inhalation of radioactive material in the plume.

TEDE or Total Effective Dose Equivalent is the summation of the Effective Dose Equivalent (EDE) and the Committed Dose Equivalent (CEDE).

$$TEDE = EDE + CEDE.$$

EDE is the dose due to an individual being directly exposed (by submersion) to the radiation present in the gaseous release (shine).

CEDE is the sum of the CDE for each organ of the body with weighting factors applied for each organ. In this calculation, only contributions from the inhalation pathway are considered.

An iterative process is used to determine the gross radiation monitored by the effluent monitors that correspond to the threshold doses.

6.4 Source Term

This calculation will not analyze for the total activity released from the core. It will only analyze for the ratios of the isotopic species that are released from the core. Various phenomena will act to change the composition of the isotopic mixture in the time between reactor shutdown and release from the facility. In summary the removal phenomena addressed here include:

RF_I = Fraction of the activity released from the **inventory** of damaged fuel described in Section 6.5.

RF_S = Fraction of the activity remaining after reduction by containment **spray** from Section 5.5.

RF_R = Fraction of activity remaining after 5 hours or 36 hours of **radioactive** decay described in Section 6.6.

RF_F = Fraction of the activity remaining after filter by SBGT **filters** from Section 5.6.

RF_{SC} = Fraction of activity remaining after natural removal processes in **secondary containment** from Section 5.7.

Combining these factors provides a single fraction to derive a depleted source:

$RF_{Total} = RF_I * RF_S * RF_R * RF_F * RF_{SC}$

6.5 Fuel Damage Release Fractions

Table 6 contains release fractions for three time periods representing the total amount of time the core has been assumed to be uncovered. They are: 0 to 0.5 hours, 0.5 to 2 hours, and 2 to 5 hours. For this calculation, the core is assumed to be uncovered for <u>one hour</u>. A spreadsheet is used to scale the release fraction between the 0.5 hour point and the 2 hour point.

The Reduction Factor, **RF_I**, due to the release fraction is 100% of the release expected in the first 0.5 hour PLUS 1/3 of the amount released as expected in the period between 0.5 hours and 2 hours.

Example for Alkali Metals: $0.05 * \left(\frac{0.5\ hr}{0.5\ hr}\right) + 0.2 * \left(\frac{0.5\ hr}{1.5\ hr}\right) = 0.1167$

Table 10 – Release Fractions by Time Step (Hours)

Group	Time (h) by step		
	0.5	1.5	Cumulative
Alkali Metals	0.050	0.2000	0.1167
Barium Group	0.000	0.0200	0.0067
Cerium Group	0.000	0.0005	0.0002
Halogen	0.050	0.2500	0.1333
Lanthanides	0.000	0.0002	0.0001
Noble Gas	0.050	0.9500	0.3667
Noble Metals	0.000	0.0025	0.0008
Tellurium group	0.000	0.0500	0.0167

6.6 Radioactive Decay

The total amount of time the radioactive source is allowed to decay before being exhausted as an effluent is 5 hours or 36 hours depending on the reactor mode per Section 6.1.

The generalized equation for radioactive decay is:

$$A = A_o e^{(-\lambda t)}$$

Where:

A = decayed activity

A_o = initial activity

λ = isotopic decay constant

t = elapsed time

and

$\lambda = \ln 2 / t_{1/2}$

With an end goal of a total reduction factor RF_{Total}, a radiation decay factor RF_R is derived from the general equation above:

$$RF_R = e^{(-\lambda t)}$$

6.7 Effective Dose Equivalent – Noble Gas Submersion

Submersion dose from noble gases is calculated with guidance provided in FGR12.

The concentration of an isotope ***i*** present in the plume at the receptor is calculated:

$$x_{ir} = x_{iv} * v * \left(\frac{X}{Q}\right)$$

With the isotopic concentration at the receptor known, the dose (mrem) at the receptor is calculated:

$$Dose = \sum_i (x_{ir} * h_{E50i})$$

Where

x_{ir} = concentration of radionuclide ***i*** present at the receptor (Ci/m^3)
Note: Ci/m^3 = µCi/cc

v = volume of gas released (m^3)

x_{iv} = concentration of radionuclide ***i*** released from the stack or building vent. (Ci/m^3)

i = each isotope present in the gaseous release

$\left(\frac{X}{Q}\right)$ = dispersion factor for that release point (sec/m^3)

h_{E50i} = factor converting the gas concentration to effective dose equivalent. $\left(\frac{mrem\ cm^3}{\mu Ci\ sec}\right)$

As described in Section 7.3, a spreadsheet is used to determine the EDE dose contribution for each isotope in the mixture.

6.8 Committed Dose Equivalent: Thyroid

Organ dose from airborne particulates and iodines is calculated with guidance provided in FGR11.

The concentration of an isotope ***i*** present in the plume at the receptor is calculated:

$$x_{ir} = \; x_{iv} * v * \left(\frac{X}{Q}\right)$$

With the isotopic concentration at the receptor known, the dose (mrem) at the receptor can be calculated:

$$Dose = \textstyle\sum_i (x_{ir} * B * t * h_{T50i})$$

Where

$x_{ir} =$ concentration of radionuclide ***i*** present at the receptor (Ci/m^3)
Note: $Ci/m^3 = \mu Ci/cm^3$

$v =$ volume of gas released (m^3)

$x_{iv} =$ concentration of radionuclide ***i*** released from the stack or building vent. (Ci/m^3)

$i =$ each isotope present in the gaseous release

$\left(\frac{X}{Q}\right) =$ dispersion factor for that release point (sec/m^3)

$B =$ breathing Rate (cm^3/sec)

$h_{T50i} =$ factor converting the gas concentration to effective dose equivalent. (mrem/μCi)

t = time the dose is to be integrated (sec)

As described in Section 7.4, a spreadsheet is used to determine the thyroid CDE dose contribution for each isotope in the mixture.

6.9 Committed Effective Dose Equivalent

Committed Effective Dose Equivalent from airborne particulates and iodines is calculated with guidance provided in FGR11.

The concentration x_{ir} of an isotope ***i*** present in the plume at the receptor is calculated:

$$x_{ir} = \; x_{iv} * v * \left(\frac{X}{Q}\right)$$

With the isotopic concentration at the receptor known, the dose (mrem) at the receptor can be calculated:

$$Dose = \textstyle\sum_i (x_{ir} * B * t * h_{E50i})$$

Where

$x_{ir} =$ concentration of radionuclide ***i*** present at the receptor (Ci/m^3)
Note: Ci/m^3 = µCi/ cm^3

$v =$ volume of gas released (m^3)

$x_{iv} =$ concentration of radionuclide ***i*** released from the stack or building vent. (Ci/m^3)

$i =$ each isotope present in the gaseous release

$\left(\frac{X}{Q}\right) =$ dispersion factor for that release point (sec/m^3)

$B =$ breathing Rate (cm^3/sec)

$h_{E50i} =$ factor converting the gas concentration to effective dose equivalent. (mrem/µCi)

t = time the dose is to be integrated (sec)

As described in Section 7.4, a spreadsheet is used to determine the CEDE dose contribution for each isotope in the mixture.

7.0 Calculation

All calculations were completed using Microsoft Excel. Sample calculations are shown in the subsections that follow.

7.1 Dose Factors

FGR11 and FGR 12 display dose factors in the SI units of Sv/Bq and Sv m^3/ Bq sec, respectively. Traditional units of mrem/µCi and mrem cm^3/µCi sec are desired.

FGR11:

1	~~Sv~~	1E+05	mrem	1	~~Bq~~	~~Ci~~		=	3.70E+09	mrem
	~~Bq~~		~~Sv~~	2.7E-11	~~Ci~~	1.00E+6	µCi			µCi

The conversion factor from Sv/Bq to mrem/µCi is 3.70E+09.

FGR 12:

1	~~Sv~~	~~m³~~	1E+05	mrem	1	~~Bq~~	1E+06	mL		~~Ci~~	=	3.70E+15	mrem cm³
	~~Bq~~	sec		~~Sv~~	2.7E-11	~~Ci~~		~~m³~~	1E+06	µCi			µCi sec

The conversion factor from Sv m^3/Bq sec to mrem cm^3/µCi sec is 3.70E+15.

The thyroid, CEDE, and submersion dose factors in the traditional units for each isotope are calculated in the table below. Column C, D, and H are dose factors from Sections 5.10 and 5.11 and Columns E and I are the conversion factors from above. Column F, G, and J are the h_{T50i}, h_{E50i}, and h_{E50i} factors as described in Sections 6.8, 6.9, and 6.7, respectively. Line 6 of Table 11 illustrates the formulas for Ba-139.

Table 11 – Isotopic Dose Factors

	Isotope	FGR11 Thyroid Sv/Bq	FGR11 CEDE Sv/Bq	Units Conversion Factor	Thyroid mrem/µCi	CEDE mrem/µCi	FGR 12: Sv m³/Bq sec	Units Conversion Factor	Submersion mrem cc/µCi sec
5	B	C	D	E	F	G	H	I	J
6	*Ba-139*	*2.40E-12*	*4.64E-11*	*3.70E+09*	*=E6*C6*	*=E6*D6*	*2.17E-15*	*3.70E+15*	*=I6*H6*
6	Ba-139	2.40E-12	4.64E-11	3.70E+09	8.88E-03	1.72E-01	2.17E-15	3.7E+15	8.03E+00
7	Ba-140	2.56E-10	1.01E-09	3.70E+09	9.47E-01	3.74E+00	8.58E-15	3.7E+15	3.17E+01
	Ce-141	4.61E-11	2.42E-09	3.70E+09	1.71E-01	8.95E+00	3.43E-15	3.7E+15	1.27E+01
	Ce-143	1.21E-11	9.16E-10	3.70E+09	4.48E-02	3.39E+00	1.29E-14	3.7E+15	4.77E+01
	Ce-144	1.88E-09	1.01E-07	3.70E+09	6.96E+00	3.74E+02	8.53E-16	3.7E+15	3.16E+00
	Cm-242	9.41E-10	4.67E-06	3.70E+09	3.48E+00	1.73E+04	5.69E-18	3.7E+15	2.11E-02
	Cs-134	1.11E-08	1.25E-08	3.70E+09	4.11E+01	4.63E+01	7.57E-14	3.7E+15	2.80E+02
	Cs-136	1.73E-09	1.98E-09	3.70E+09	6.40E+00	7.33E+00	1.06E-13	3.7E+15	3.92E+02
	Cs-137	7.93E-09	8.63E-09	3.70E+09	2.93E+01	3.19E+01	7.74E-18	3.7E+15	2.86E-02
	I-131	2.92E-07	8.89E-09	3.70E+09	1.08E+03	3.29E+01	1.82E-14	3.7E+15	6.73E+01
	I-132	1.74E-09	1.03E-10	3.70E+09	6.44E+00	3.81E-01	1.12E-13	3.7E+15	4.14E+02
	I-133	4.86E-08	1.58E-09	3.70E+09	1.80E+02	5.85E+00	2.94E-14	3.7E+15	1.09E+02

Isotope	FGR11 Thyroid Sv/Bq	FGR11 CEDE Sv/Bq	Units Conversion Factor	Thyroid mrem/µCi	CEDE mrem/µCi	FGR 12: Sv m³/Bq sec	Units Conversion Factor	Submersion mrem cc/µCi sec
I-134	2.88E-10	3.55E-11	3.70E+09	1.07E+00	1.31E-01	1.3E-13	3.7E+15	4.81E+02
I-135	8.46E-09	3.32E-10	3.70E+09	3.13E+01	1.23E+00	7.98E-14	3.7E+15	2.95E+02
Kr-83m						1.5E-18	3.7E+15	5.55E-03
Kr-85						1.19E-16	3.7E+15	4.40E-01
Kr-85m						7.48E-15	3.7E+15	2.77E+01
Kr-87						4.12E-14	3.7E+15	1.52E+02
Kr-88						1.02E-13	3.7E+15	3.77E+02
La-140	1.22E-10	1.31E-09	3.70E+09	4.51E-01	4.85E+00	1.17E-13	3.7E+15	4.33E+02
La-141	9.40E-12	1.57E-10	3.70E+09	3.48E-02	5.81E-01	2.39E-15	3.7E+15	8.84E+00
La-142	8.74E-12	6.84E-11	3.70E+09	3.23E-02	2.53E-01	1.44E-13	3.7E+15	5.33E+02
Mo-99	1.17E-10	1.07E-09	3.70E+09	4.33E-01	3.96E+00	7.28E-15	3.7E+15	2.69E+01
Nb-95	3.58E-10	1.57E-09	3.70E+09	1.32E+00	5.81E+00	3.74E-14	3.7E+15	1.38E+02
Nd-147	1.94E-11	1.85E-09	3.70E+09	7.18E-02	6.85E+00	6.19E-15	3.7E+15	2.29E+01
Np-239	7.62E-12	6.78E-10	3.70E+09	2.82E-02	2.51E+00	7.69E-15	3.7E+15	2.85E+01
Pr-143	1.68E-18	2.19E-09	3.70E+09	6.22E-09	8.10E+00	2.1E-17	3.7E+15	7.77E-02
Pu-241	1.24E-11	2.23E-06	3.70E+09	4.59E-02	8.25E+03	7.25E-20	3.7E+15	2.68E-04
Rb-86	1.33E-09	1.79E-09	3.70E+09	4.92E+00	6.62E+00	4.81E-15	3.7E+15	1.78E+01
Rh-105	2.57E-11	2.58E-10	3.70E+09	9.51E-02	9.55E-01	3.72E-15	3.7E+15	1.38E+01
Ru-103	5.97E-10	2.42E-09	3.70E+09	2.21E+00	8.95E+00	2.25E-14	3.7E+15	8.33E+01
Ru-105	1.50E-11	1.23E-10	3.70E+09	5.55E-02	4.55E-01	3.81E-14	3.7E+15	1.41E+02
Ru-106	1.37E-08	1.29E-07	3.70E+09	5.07E+01	4.77E+02	0	3.7E+15	0.00E+00
Sb-127	1.50E-10	1.63E-09	3.70E+09	5.55E-01	6.03E+00	3.33E-14	3.7E+15	1.23E+02
Sb-129	2.07E-11	1.74E-10	3.70E+09	7.66E-02	6.44E-01	7.14E-14	3.7E+15	2.64E+02
Sr-89	4.16E-10	1.12E-08	3.70E+09	1.54E+00	4.14E+01	7.73E-17	3.7E+15	2.86E-01
Sr-90	2.64E-09	3.51E-07	3.70E+09	9.77E+00	1.30E+03	7.53E-18	3.7E+15	2.79E-02
Sr-91	4.08E-11	4.49E-10	3.70E+09	1.51E-01	1.66E+00	3.45E-14	3.7E+15	1.28E+02
Sr-92	2.19E-11	2.18E-10	3.70E+09	8.10E-02	8.07E-01	6.79E-14	3.7E+15	2.51E+02
Tc-99m	5.01E-11	8.80E-12	3.70E+09	1.85E-01	3.26E-02	5.89E-15	3.7E+15	2.18E+01
Te-127	6.46E-12	8.60E-11	3.70E+09	2.39E-02	3.18E-01	2.42E-16	3.7E+15	8.95E-01
Te-127m	2.39E-10	5.81E-09	3.70E+09	8.84E-01	2.15E+01	1.47E-16	3.7E+15	5.44E-01
Te-129	1.63E-12	2.42E-11	3.70E+09	6.03E-03	8.95E-02	2.75E-15	3.7E+15	1.02E+01
Te-129m	3.95E-10	6.47E-09	3.70E+09	1.46E+00	2.39E+01	1.55E-15	3.7E+15	5.74E+00
Te-131m	3.61E-08	1.73E-09	3.70E+09	1.34E+02	6.40E+00	7.01E-14	3.7E+15	2.59E+02
Te-132	6.28E-08	2.55E-09	3.70E+09	2.32E+02	9.44E+00	1.03E-14	3.7E+15	3.81E+01
Xe-131m						3.89E-16	3.7E+15	1.44E+00
Xe-133						1.56E-15	3.7E+15	5.77E+00
Xe-133m						1.37E-15	3.7E+15	5.07E+00
Xe-135						1.19E-14	3.7E+15	4.40E+01
Xe-135m						2.04E-14	3.7E+15	7.55E+01
Xe-138						5.77E-14	3.7E+15	2.13E+02
Y-90	9.52E-12	2.28E-09	3.70E+09	3.52E-02	8.44E+00	1.9E-16	3.7E+15	7.03E-01
Y-91	1.10E-10	1.32E-08	3.70E+09	4.07E-01	4.88E+01	2.6E-16	3.7E+15	9.62E-01
Y-92	3.69E-12	2.11E-10	3.70E+09	1.37E-02	7.81E-01	1.3E-14	3.7E+15	4.81E+01
Y-93	5.06E-12	5.82E-10	3.70E+09	1.87E-02	2.15E+00	4.8E-15	3.7E+15	1.78E+01
Zr-95	1.44E-09	6.39E-09	3.70E+09	5.33E+00	2.36E+01	3.6E-14	3.7E+15	1.33E+02
Zr-97	9.56E-11	1.17E-09	3.70E+09	3.54E-01	4.33E+00	9.02E-15	3.7E+15	3.34E+01

7.2 Source Term

A spreadsheet is used to determine the total reduction factor **RF_{Total}** for each isotope present in the source term as described in Section 6.4. The activity per megawatt thermal from Section 5.1 is multiplied by **RF_{Total}** to find the source term for each isotope. The spreadsheet for the Offgas Stack release is presented in Table 13.

The relative activity released from damaged fuel (**RF_I**) was determined in Section 6.5.

Table 12 – 2 Hours Reduction Factor (RF_I)

Cumulative	2 Hour
Alkali Metals	0.1167
Barium Group	0.0067
Cerium Group	0.0002
Halogen	0.1333
Lanthanides	0.0001
Noble Gas	0.3667
Noble Metals	0.0008
Tellurium group	0.0167

A Spray Reduction factor of 0.278 for primary containment sprays (**RF_S**) was derived in Section 5.5.

Determination of the Radiation Decay fractions (**RF_R**) was demonstrated in Section 6.6. In the spreadsheets below, the source decay time is 5 hours.

7.2.1 Offgas Stack

For the Offgas Stack release, credit is taken for filtering (**RF**$_F$) by the Standby Gas Treatment system but not for the natural removal processes that occur in secondary containment (**RF**$_{SC}$).

Table 13 – Isotopic Depletion and Release for Offgas Stack

			RF$_I$	**RF$_S$**	**RF$_{SC}$**	**RF$_F$**	**RF$_R$**	**RF$_{Total}$**	
Form	Isotope	Ci MWTh	Release Fraction	+ 0.25 hr Sprays Reduction	Secondary Con-tainment	SBGT Filter	Decay Fraction	Total Depletion	Release Ci/MWTh
Barium Group	Ba-139	4.74E+04	0.0067	0.2780	1.0000	0.01	0.0808	1.50E-06	7.10E-02
Barium Group	Ba-140	4.76E+04	0.0067	0.2780	1.0000	0.01	0.9887	1.83E-05	8.72E-01
Cerium Group	Ce-141	4.39E+04	0.0002	0.2780	1.0000	0.01	0.9956	4.61E-07	2.03E-02
Cerium Group	Ce-143	4.00E+04	0.0002	0.2780	1.0000	0.01	0.9006	4.17E-07	1.67E-02
Cerium Group	Ce-144	3.54E+04	0.0002	0.2780	1.0000	0.01	0.9995	4.63E-07	1.64E-02
Lanthanides	Cm-242	1.12E+03	0.0001	0.2780	1.0000	0.01	0.9991	1.85E-07	2.07E-04
Alkali Metals	Cs-134	4.70E+03	0.1167	0.2780	1.0000	0.01	0.9998	3.24E-04	1.52E+00
Alkali Metals	Cs-136	1.49E+03	0.1167	0.2780	1.0000	0.01	0.9890	3.21E-04	4.78E-01
Alkali Metals	Cs-137	3.25E+03	0.1167	0.2780	1.0000	0.01	1.0000	3.24E-04	1.05E+00
Halogen	I-131	2.67E+04	0.1333	0.2780	1.0000	0.01	0.9822	3.64E-04	9.72E+00
Halogen	I-132	3.88E+04	0.1333	0.2780	1.0000	0.01	0.2215	8.21E-05	3.19E+00
Halogen	I-133	5.42E+04	0.1333	0.2780	1.0000	0.01	0.8466	3.14E-04	1.70E+01
Halogen	I-134	5.98E+04	0.1333	0.2780	1.0000	0.01	0.0191	7.09E-06	4.24E-01
Halogen	I-135	5.18E+04	0.1333	0.2780	1.0000	0.01	0.5915	2.19E-04	1.14E+01
Noble Gas	Kr-83m	3.05E+03	0.367	1.0	1.0	1.0	0.1505	5.52E-02	1.68E+02
Noble Gas	Kr-85	2.78E+02	0.367	1.0	1.0	1.0	1.0000	3.67E-01	1.02E+02
Noble Gas	Kr-85m	6.17E+03	0.367	1.0	1.0	1.0	0.4620	1.69E-01	1.05E+03
Noble Gas	Kr-87	1.23E+04	0.367	1.0	1.0	1.0	0.0656	2.40E-02	2.96E+02
Noble Gas	Kr-88	1.70E+04	0.367	1.0	1.0	1.0	0.2941	1.08E-01	1.83E+03
Lanthanides	La-140	4.91E+04	0.0001	0.2780	1.0000	0.01	0.9176	1.70E-07	8.35E-03
Lanthanides	La-141	4.33E+04	0.0001	0.2780	1.0000	0.01	0.4146	7.68E-08	3.33E-03
Lanthanides	La-142	4.21E+04	0.0001	0.2780	1.0000	0.01	0.1055	1.96E-08	8.23E-04
Noble Metals	Mo-99	5.30E+04	0.0008	0.2780	1.0000	0.01	0.9488	2.20E-06	1.17E-01
Lanthanides	Nb-95	4.50E+04	0.0001	0.2780	1.0000	0.01	0.9959	1.85E-07	8.31E-03
Lanthanides	Nd-147	1.75E+04	0.0001	0.2780	1.0000	0.01	0.9870	1.83E-07	3.20E-03
Cerium Group	Np-239	5.69E+05	0.0002	0.2780	1.0000	0.01	0.9406	4.36E-07	2.48E-01
Lanthanides	Pr-143	3.96E+04	0.0001	0.2780	1.0000	0.01	0.9894	1.83E-07	7.26E-03
Cerium Group	Pu-241	4.26E+03	0.0002	0.2780	1.0000	0.01	1.0000	4.63E-07	1.97E-03
Alkali Metals	Rb-86	5.29E+01	0.1167	0.2780	1.0000	0.01	0.9923	3.22E-04	1.70E-02
Noble Metals	Rh-105	2.81E+04	0.0008	0.2780	1.0000	0.01	0.9064	2.10E-06	5.90E-02
Noble Metals	Ru-103	4.34E+04	0.0008	0.2780	1.0000	0.01	0.9963	2.31E-06	1.00E-01
Noble Metals	Ru-105	3.06E+04	0.0008	0.2780	1.0000	0.01	0.4581	1.06E-06	3.25E-02
Noble Metals	Ru-106	1.55E+04	0.0008	0.2780	1.0000	0.01	0.9996	2.32E-06	3.59E-02
Tellurium group	Sb-127	2.39E+03	0.0167	0.2780	1.0000	0.01	0.9632	4.46E-05	1.07E-01
Tellurium group	Sb-129	8.68E+03	0.0167	0.2780	1.0000	0.01	0.4549	2.11E-05	1.83E-01
Barium Group	Sr-89	2.41E+04	0.0067	0.2780	1.0000	0.01	0.9971	1.85E-05	4.45E-01
Barium Group	Sr-90	2.39E+03	0.0067	0.2780	1.0000	0.01	1.0000	1.85E-05	4.43E-02

			RF_I	RF_S	RF_{SC}	RF_F	RF_R	RF_{Total}	
Form	Isotope	Ci / MWTh	Release Fraction	+ 0.25 hr Sprays Reduction	Secondary Con-tainment	SBGT Filter	Decay Fraction	Total Depletion	Release Ci/MWTh
Barium Group	Sr-91	3.01E+04	0.0067	0.2780	1.0000	0.01	0.6944	1.29E-05	3.87E-01
Barium Group	Sr-92	3.24E+04	0.0067	0.2780	1.0000	0.01	0.2786	5.16E-06	1.67E-01
Noble Metals	Tc-99m	4.37E+04	0.0008	0.2780	1.0000	0.01	0.5625	1.30E-06	5.70E-02
Tellurium group	Te-127	2.36E+03	0.0167	0.2780	1.0000	0.01	0.6905	3.20E-05	7.55E-02
Tellurium group	Te-127m	3.97E+02	0.0167	0.2780	1.0000	0.01	0.9987	4.63E-05	1.84E-02
Tellurium group	Te-129	8.26E+03	0.0167	0.2780	1.0000	0.01	0.0503	2.33E-06	1.93E-02
Tellurium group	Te-129m	1.68E+03	0.0167	0.2780	1.0000	0.01	0.9957	4.61E-05	7.75E-02
Tellurium group	Te-131m	5.41E+03	0.0167	0.2780	1.0000	0.01	0.8909	4.13E-05	2.23E-01
Tellurium group	Te-132	3.81E+04	0.0167	0.2780	1.0000	0.01	0.9567	4.43E-05	1.69E+00
Noble Gas	Xe-131m	3.65E+02	0.367	1.0	1.0	1.0	0.9880	3.62E-01	1.32E+02
Noble Gas	Xe-133	5.43E+04	0.367	1.0	1.0	1.0	0.9729	3.57E-01	1.94E+04
Noble Gas	Xe-133m	1.72E+03	0.367	1.0	1.0	1.0	0.9362	3.43E-01	5.90E+02
Noble Gas	Xe-135	1.42E+04	0.367	1.0	1.0	1.0	0.6832	2.50E-01	3.56E+03
Noble Gas	Xe-135m	1.15E+04	0.367	1.0	1.0	1.0	0.0000	4.55E-07	5.23E-03
Noble Gas	Xe-138	4.56E+04	0.367	1.0	1.0	1.0	0.0000	1.41E-07	6.45E-03
Lanthanides	Y-90	2.45E+03	0.0001	0.2780	1.0000	0.01	0.9474	1.76E-07	4.30E-04
Lanthanides	Y-91	3.17E+04	0.0001	0.2780	1.0000	0.01	0.9975	1.85E-07	5.86E-03
Lanthanides	Y-92	3.26E+04	0.0001	0.2780	1.0000	0.01	0.3769	6.99E-08	2.28E-03
Lanthanides	Y-93	2.52E+04	0.0001	0.2780	1.0000	0.01	0.7096	1.32E-07	3.31E-03
Lanthanides	Zr-95	4.44E+04	0.0001	0.2780	1.0000	0.01	0.9977	1.85E-07	8.21E-03
Lanthanides	Zr-97	4.23E+04	0.0001	0.2780	1.0000	0.01	0.8145	1.51E-07	6.39E-03

7.2.2 Building Vents

For releases from Building Vents, no credit is taken for filtering (**RF**$_F$) by the Standby Gas Treatment system. Credit is taken for the natural removal processes that occurs in secondary containment (**RF**$_{SC}$). This source term also has radioactive decay occurring for 5 hours.

Table 14 – Isotopic Depletion and Release for Building Vents

			RF$_I$	RF$_S$	RF$_{SC}$	RF$_F$	RF$_R$	RF$_{Total}$	
Form	Isotope	Ci/MWTh	Release Fraction	+ 0.25 hr Sprays Reduction	Secondary Con-tainment	SBGT Filter	Decay Fraction	Total Depletion	Release Ci/MWTh
Barium Group	Ba-139	4.74E+04	0.0067	0.2780	0.4000	1.00	0.0808	5.99E-05	2.84E+00
Barium Group	Ba-140	4.76E+04	0.0067	0.2780	0.4000	1.00	0.9887	7.33E-04	3.49E+01
Cerium Group	Ce-141	4.39E+04	0.0002	0.2780	0.4000	1.00	0.9956	1.85E-05	8.10E-01
Cerium Group	Ce-143	4.00E+04	0.0002	0.2780	0.4000	1.00	0.9006	1.67E-05	6.68E-01
Cerium Group	Ce-144	3.54E+04	0.0002	0.2780	0.4000	1.00	0.9995	1.85E-05	6.56E-01
Lanthanides	Cm-242	1.12E+03	0.0001	0.2780	0.4000	1.00	0.9991	7.41E-06	8.30E-03
Alkali Metals	Cs-134	4.70E+03	0.1167	0.2780	0.4000	1.00	0.9998	1.30E-02	6.10E+01
Alkali Metals	Cs-136	1.49E+03	0.1167	0.2780	0.4000	1.00	0.9890	1.28E-02	1.91E+01
Alkali Metals	Cs-137	3.25E+03	0.1167	0.2780	0.4000	1.00	1.0000	1.30E-02	4.22E+01
Halogen	I-131	2.67E+04	0.1333	0.2780	0.4000	1.00	0.9822	1.46E-02	3.89E+02
Halogen	I-132	3.88E+04	0.1333	0.2780	0.4000	1.00	0.2215	3.28E-03	1.27E+02
Halogen	I-133	5.42E+04	0.1333	0.2780	0.4000	1.00	0.8466	1.26E-02	6.80E+02
Halogen	I-134	5.98E+04	0.1333	0.2780	0.4000	1.00	0.0191	2.84E-04	1.70E+01
Halogen	I-135	5.18E+04	0.1333	0.2780	0.4000	1.00	0.5915	8.77E-03	4.54E+02
Noble Gas	Kr-83m	3.05E+03	0.367	1.0	1.0	1.0	0.1505	5.52E-02	1.68E+02
Noble Gas	Kr-85	2.78E+02	0.367	1.0	1.0	1.0	1.0000	3.67E-01	1.02E+02
Noble Gas	Kr-85m	6.17E+03	0.367	1.0	1.0	1.0	0.4620	1.69E-01	1.05E+03
Noble Gas	Kr-87	1.23E+04	0.367	1.0	1.0	1.0	0.0656	2.40E-02	2.96E+02
Noble Gas	Kr-88	1.70E+04	0.367	1.0	1.0	1.0	0.2941	1.08E-01	1.83E+03
Lanthanides	La-140	4.91E+04	0.0001	0.2780	0.4000	1.00	0.9176	6.80E-06	3.34E-01
Lanthanides	La-141	4.33E+04	0.0001	0.2780	0.4000	1.00	0.4146	3.07E-06	1.33E-01
Lanthanides	La-142	4.21E+04	0.0001	0.2780	0.4000	1.00	0.1055	7.82E-07	3.29E-02
Noble Metals	Mo-99	5.30E+04	0.0008	0.2780	0.4000	1.00	0.9488	8.79E-05	4.66E+00
Lanthanides	Nb-95	4.50E+04	0.0001	0.2780	0.4000	1.00	0.9959	7.38E-06	3.32E-01
Lanthanides	Nd-147	1.75E+04	0.0001	0.2780	0.4000	1.00	0.9870	7.32E-06	1.28E-01
Cerium Group	Np-239	5.69E+05	0.0002	0.2780	0.4000	1.00	0.9406	1.74E-05	9.92E+00
Lanthanides	Pr-143	3.96E+04	0.0001	0.2780	0.4000	1.00	0.9894	7.34E-06	2.91E-01
Cerium Group	Pu-241	4.26E+03	0.0002	0.2780	0.4000	1.00	1.0000	1.85E-05	7.90E-02
Alkali Metals	Rb-86	5.29E+01	0.1167	0.2780	0.4000	1.00	0.9923	1.29E-02	6.81E-01
Noble Metals	Rh-105	2.81E+04	0.0008	0.2780	0.4000	1.00	0.9064	8.40E-05	2.36E+00
Noble Metals	Ru-103	4.34E+04	0.0008	0.2780	0.4000	1.00	0.9963	9.23E-05	4.01E+00
Noble Metals	Ru-105	3.06E+04	0.0008	0.2780	0.4000	1.00	0.4581	4.25E-05	1.30E+00
Noble Metals	Ru-106	1.55E+04	0.0008	0.2780	0.4000	1.00	0.9996	9.26E-05	1.44E+00
Tellurium group	Sb-127	2.39E+03	0.0167	0.2780	0.4000	1.00	0.9632	1.79E-03	4.27E+00
Tellurium group	Sb-129	8.68E+03	0.0167	0.2780	0.4000	1.00	0.4549	8.43E-04	7.32E+00
Barium Group	Sr-89	2.41E+04	0.0067	0.2780	0.4000	1.00	0.9971	7.39E-04	1.78E+01

			RF_I	RF_S	RF_{SC}	RF_F	RF_R	RF_{Total}	
Form	Isotope	Ci / MWTh	Release Fraction	+ 0.25 hr Sprays Reduction	Secondary Con-tainment	SBGT Filter	Decay Fraction	Total Depletion	Release Ci/MWTh
Barium Group	Sr-90	2.39E+03	0.0067	0.2780	0.4000	1.00	1.0000	7.41E-04	1.77E+00
Barium Group	Sr-91	3.01E+04	0.0067	0.2780	0.4000	1.00	0.6944	5.15E-04	1.55E+01
Barium Group	Sr-92	3.24E+04	0.0067	0.2780	0.4000	1.00	0.2786	2.07E-04	6.69E+00
Noble Metals	Tc-99m	4.37E+04	0.0008	0.2780	0.4000	1.00	0.5625	5.21E-05	2.28E+00
Tellurium group	Te-127	2.36E+03	0.0167	0.2780	0.4000	1.00	0.6905	1.28E-03	3.02E+00
Tellurium group	Te-127m	3.97E+02	0.0167	0.2780	0.4000	1.00	0.9987	1.85E-03	7.35E-01
Tellurium group	Te-129	8.26E+03	0.0167	0.2780	0.4000	1.00	0.0503	9.32E-05	7.70E-01
Tellurium group	Te-129m	1.68E+03	0.0167	0.2780	0.4000	1.00	0.9957	1.85E-03	3.10E+00
Tellurium group	Te-131m	5.41E+03	0.0167	0.2780	0.4000	1.00	0.8909	1.65E-03	8.93E+00
Tellurium group	Te-132	3.81E+04	0.0167	0.2780	0.4000	1.00	0.9567	1.77E-03	6.76E+01
Noble Gas	Xe-131m	3.65E+02	0.367	1.0	1.0	1.0	0.9880	3.62E-01	1.32E+02
Noble Gas	Xe-133	5.43E+04	0.367	1.0	1.0	1.0	0.9729	3.57E-01	1.94E+04
Noble Gas	Xe-133m	1.72E+03	0.367	1.0	1.0	1.0	0.9362	3.43E-01	5.90E+02
Noble Gas	Xe-135	1.42E+04	0.367	1.0	1.0	1.0	0.6832	2.50E-01	3.56E+03
Noble Gas	Xe-135m	1.15E+04	0.367	1.0	1.0	1.0	0.0000	4.55E-07	5.23E-03
Noble Gas	Xe-138	4.56E+04	0.367	1.0	1.0	1.0	0.0000	1.41E-07	6.45E-03
Lanthanides	Y-90	2.45E+03	0.0001	0.2780	0.4000	1.00	0.9474	7.02E-06	1.72E-02
Lanthanides	Y-91	3.17E+04	0.0001	0.2780	0.4000	1.00	0.9975	7.40E-06	2.34E-01
Lanthanides	Y-92	3.26E+04	0.0001	0.2780	0.4000	1.00	0.3769	2.79E-06	9.11E-02
Lanthanides	Y-93	2.52E+04	0.0001	0.2780	0.4000	1.00	0.7096	5.26E-06	1.33E-01
Lanthanides	Zr-95	4.44E+04	0.0001	0.2780	0.4000	1.00	0.9977	7.40E-06	3.28E-01
Lanthanides	Zr-97	4.23E+04	0.0001	0.2780	0.4000	1.00	0.8145	6.04E-06	2.55E-01

7.3 Effective Dose Equivalent – Noble Gas Submersion

Spreadsheets are used to calculate isotopic concentration at the receptor and the resultant radiation dose to the receptor for each of the isotopes in the mixture.

For the example Effective Dose Equivalent calculation, the release point is the Offgas Stack at five hours since shutdown, and a gross concentration of 43.9 µCi/cm³ (this concentration was determined iteratively to produce 10 mrem TEDE). The secondary containment holdup hours is set at <0.5 because the natural removal process in the Secondary Containment does not occur with the Offgas Stack.

In Table 15, the column labeled "h_{E50i} Submersion mrem cm³/µCi sec," is the dose factor for air submersion dose and is calculated in Section 7.1.

The column labeled "Depleted Mix Ci/MWTh" is the "Release Ci/MWTh" calculated in Section 7.2 for each isotope.

The "Fraction" column determines the fraction each isotope contributes to the gross activity and is used to scale the activity for each isotope.

The column "x_{iv} Release Conc. µCi/cm³" contains a calculation that scales the "Depleted Mix Ci/MWTh" column to a user entered gross concentration based on the "Fraction". In this case, the gross concentration entered was 43.9 µCi/cm³ (4.39E+1).

Values in the "x_{ir} Receptor Conc. µCi/cm³" column are calculated by multiplying the release concentration by the applicable dispersion factor, the volume of the release, and requisite conversion factors. The basic equation is from Section 6.7:

$$X_{ir} = \quad X_{iv} * V * \left(\frac{X}{Q}\right)$$

For isotope I-131, an example is presented:

x_{iv} Release Conc.		Flow						(X/Q)			Receptor Conc.	
1.57E-02	µCi	10,000	~~ft³~~	2.83E-02	~~m³~~	1	~~min~~	2.80E-07	~~sec~~	=	2.08E-08	µCi
	cm³		~~min~~	1	~~ft³~~	60	~~sec~~		~~m³~~			cm³

Where:

v = 10,000 ft³/min is the rated flow from the Offgas Stack from Design Input 5.8.

(X/Q) = 2.80E-07 is the Noble Gas Dispersion coefficient *(X/Q)* for the Offgas Stack from Design Input 5.3.

2.83E-2 converts ft³ to m³

Values in the "Submersion Dose mrem" (h_{E50i}) column are calculated by multiplying the factors "x_{ir} Receptor Conc. µCi/cm³", a time-units conversion factor, and the dose conversion factor calculated in Section 7.1. The basic equation for a one hour time period is shown in Section 6.7.

$$Dose = \sum_i (x_{ir} * hE_{50i})$$

For isotope I-131, an example is presented:

x_{ir} Receptor Conc.		h_{E50i} Submersion						Submersion Dose mrem	
2.08E-08	~~µCi~~	6.73E+01	mrem	~~cm³~~	3600	~~sec~~	=	5.04E-03	mrem
	~~cm³~~		~~µCi~~	~~sec~~					

For ease of comparison, the spreadsheet row for I-131 is shown here:

Nuclide	h_{E50i} Submersion $\frac{mrem\ cc}{\mu Ci\ sec}$	Depleted Mix $\frac{Ci}{MWTh}$	Fraction	x_{iv} Release Conc. $\frac{\mu Ci}{cm^3}$	x_{ir} Receptor Conc. $\frac{\mu Ci}{cm^3}$	Submersion Dose mrem
I-131	6.73E+1	9.72E+0	3.582E-4	1.57E-2	2.08E-8	5.04E-3

Table 15 – Submersion Dose for Offgas Stack

Nuclide	h_{E50i} Submersion $\frac{mrem\ cm^3}{\mu Ci\ sec}$	Depleted Mix $\frac{Ci}{MWTh}$	Fraction	x_{iv} Release Conc. $\frac{\mu Ci}{cm^3}$	x_{ir} Receptor Conc. $\frac{\mu Ci}{cm^3}$	Submersion Dose mrem
Ba-139	8.03E+0	7.10E-2	2.62E-6	1.15E-4	1.52E-10	4.39E-6
Ba-140	3.17E+1	8.72E-1	3.21E-5	1.41E-3	1.86E-9	2.13E-4
Ce-141	1.27E+1	2.03E-2	7.46E-7	3.28E-5	4.33E-11	1.98E-6
Ce-143	4.77E+1	1.67E-2	6.15E-7	2.70E-5	3.57E-11	6.13E-6
Ce-144	3.16E+0	1.64E-2	6.04E-7	2.65E-5	3.50E-11	3.98E-7
Cm-242	2.11E-2	2.07E-4	7.64E-9	3.35E-7	4.43E-13	3.36E-11
Cs-134	2.80E+2	1.52E+0	5.62E-5	2.47E-3	3.26E-9	3.28E-3
Cs-136	3.92E+2	4.78E-1	1.76E-5	7.73E-4	1.02E-9	1.44E-3
Cs-137	2.86E-2	1.05E+0	3.88E-5	1.70E-3	2.25E-9	2.32E-7
I-131	6.73E+1	9.72E+0	3.582E-4	1.57E-2	2.08E-8	5.04E-3
I-132	4.14E+2	3.19E+0	1.17E-4	5.15E-3	6.81E-9	1.02E-2
I-133	1.09E+2	1.70E+1	6.27E-4	2.75E-2	3.64E-8	1.42E-2
I-134	4.81E+2	4.24E-1	1.56E-5	6.86E-4	9.07E-10	1.57E-3
I-135	2.95E+2	1.14E+1	4.18E-4	1.84E-2	2.43E-8	2.58E-2
Kr-83m	5.55E-3	1.68E+2	6.20E-3	2.72E-1	3.60E-7	7.19E-6
Kr-85	4.40E-1	1.02E+2	3.76E-3	1.65E-1	2.18E-7	3.45E-4
Kr-85m	2.77E+1	1.05E+3	3.85E-2	1.69E+0	2.23E-6	2.23E-1
Kr-87	1.52E+2	2.96E+2	1.09E-2	4.78E-1	6.32E-7	3.47E-1
Kr-88	3.77E+2	1.83E+3	6.75E-2	2.97E+0	3.92E-6	5.32E+0
La-140	4.33E+2	8.35E-3	3.08E-7	1.35E-5	1.78E-11	2.78E-5
La-141	8.84E+0	3.33E-3	1.23E-7	5.38E-6	7.11E-12	2.26E-7
La-142	5.33E+2	8.23E-4	3.03E-8	1.33E-6	1.76E-12	3.37E-6
Mo-99	2.69E+1	1.17E-1	4.29E-6	1.88E-4	2.49E-10	2.41E-5
Nb-95	1.38E+2	8.31E-3	3.06E-7	1.34E-5	1.78E-11	8.84E-6
Nd-147	2.29E+1	3.20E-3	1.18E-7	5.18E-6	6.84E-12	5.64E-7
Np-239	2.85E+1	2.48E-1	9.14E-6	4.01E-4	5.30E-10	5.43E-5
Pr-143	7.77E-2	7.26E-3	2.68E-7	1.17E-5	1.55E-11	4.34E-9
Pu-241	2.68E-4	1.97E-3	7.27E-8	3.19E-6	4.22E-12	4.07E-12
Rb-86	1.78E+1	1.70E-2	6.27E-7	2.75E-5	3.64E-11	2.33E-6
Rh-105	1.38E+1	5.90E-2	2.17E-6	9.54E-5	1.26E-10	6.25E-6
Ru-103	8.33E+1	1.00E-1	3.69E-6	1.62E-4	2.14E-10	6.42E-5
Ru-105	1.41E+2	3.25E-2	1.20E-6	5.25E-5	6.94E-11	3.52E-5
Ru-106	0.00E+0	3.59E-2	1.32E-6	5.81E-5	7.67E-11	0.00E+0
Sb-127	1.23E+2	1.07E-1	3.93E-6	1.73E-4	2.28E-10	1.01E-4
Sb-129	2.64E+2	1.83E-1	6.74E-6	2.96E-4	3.91E-10	3.72E-4
Sr-89	2.86E-1	4.45E-1	1.64E-5	7.20E-4	9.52E-10	9.80E-7
Sr-90	2.79E-2	4.43E-2	1.63E-6	7.16E-5	9.47E-11	9.50E-9
Sr-91	1.28E+2	3.87E-1	1.43E-5	6.27E-4	8.28E-10	3.81E-4
Sr-92	2.51E+2	1.67E-1	6.16E-6	2.71E-4	3.58E-10	3.23E-4
Tc-99m	2.18E+1	5.70E-2	2.10E-6	9.21E-5	1.22E-10	9.55E-6

Nuclide	h_{E50i} Submersion $\frac{mrem\ cm^3}{\mu Ci\ sec}$	Depleted Mix $\frac{Ci}{MWTh}$	Fraction	x_{iv} Release Conc. $\frac{\mu Ci}{cm^3}$	x_{ir} Receptor Conc. $\frac{\mu Ci}{cm^3}$	Submersion Dose mrem
Te-127	8.95E-1	7.55E-2	2.78E-6	1.22E-4	1.61E-10	5.20E-7
Te-127m	5.44E-1	1.84E-2	6.77E-7	2.97E-5	3.93E-11	7.69E-8
Te-129	1.02E+1	1.93E-2	7.09E-7	3.11E-5	4.11E-11	1.51E-6
Te-129m	5.74E+0	7.75E-2	2.86E-6	1.25E-4	1.66E-10	3.42E-6
Te-131m	2.59E+2	2.23E-1	8.23E-6	3.61E-4	4.77E-10	4.46E-4
Te-132	3.81E+1	1.69E+0	6.22E-5	2.73E-3	3.61E-9	4.95E-4
Xe-131m	1.44E+0	1.32E+2	4.87E-3	2.14E-1	2.83E-7	1.46E-3
Xe-133	5.77E+0	1.94E+4	7.14E-1	3.13E+1	4.14E-5	8.60E-1
Xe-133m	5.07E+0	5.90E+2	2.18E-2	9.55E-1	1.26E-6	2.30E-2
Xe-135	4.40E+1	3.56E+3	1.31E-1	5.75E+0	7.60E-6	1.20E+0
Xe-135m	7.55E+1	5.23E-3	1.93E-7	8.46E-6	1.12E-11	3.04E-6
Xe-138	2.13E+2	6.45E-3	2.37E-7	1.04E-5	1.38E-11	1.06E-5
Y-90	7.03E-1	4.30E-4	1.58E-8	6.96E-7	9.19E-13	2.33E-9
Y-91	9.62E-1	5.86E-3	2.16E-7	9.48E-6	1.25E-11	4.34E-8
Y-92	4.81E+1	2.28E-3	8.39E-8	3.68E-6	4.87E-12	8.43E-7
Y-93	1.78E+1	3.31E-3	1.22E-7	5.36E-6	7.08E-12	4.53E-7
Zr-95	1.33E+2	8.21E-3	3.03E-7	1.33E-5	1.75E-11	8.42E-6
Zr-97	3.34E+1	6.39E-3	2.35E-7	1.03E-5	1.36E-11	1.64E-6
		2.71E+04	100.00%	4.39E+01	5.80E-5	**8.05**
				4.39E+1		**mrem**

Given a radiation effluent monitor reading of 43.9 μCi/cm³, and the assumptions of the scenario, the EDE value is 8.05 mrem.

Spreadsheet cases are run for all four release points. See Section 2.0 for results.

7.4 CEDE and CDE Thyroid

For the example CEDE and CDE Thyroid calculation, the release point is the Reactor Building at five hours since shutdown, and a gross concentration of 1.22E-2 µCi/cc, with a Secondary Containment Holdup time of 0.5 hours per Design Input 5.7 (this concentration was determined iteratively to produce 49.8 mrem Thyroid CDE).

In Table 16, the columns labeled "h_{T50i}Thyroid mrem/µCi" and "h_{E50i} CEDE mrem/µCi" are the dose factors developed in Section 7.1.

The column labeled "Depleted Mix Ci/MWTh" is the "Release Ci/MWTh" calculated above in Section 7.2 for each isotope.

The "Fraction" column determines the fraction each isotope contributes to the gross activity, and is used to scale the activity for each isotope.

The column "x_{iv} Release Conc. µCi/cm^3" contains a calculation that scales the "Depleted Mix" column to a user entered gross concentration based on the "Fraction" and is the variable x_{iv} in the equation below. In this case, the gross concentration entered was 1.22E-2 µCi/cc.

Values in the "x_{ir} Receptor Conc. µCi/cm^3" column are calculated by multiplying the release concentration by the applicable dispersion factor, the volume of the release, and requisite conversion factors. The basic equation from Section 6.8:

$$X_{ir} = X_{iv} * V * \left(\frac{X}{Q}\right)$$

For isotope I-131, an example is presented:

x_{iv} Release Conc.		Flow						(X/Q)			Receptor Conc.	
1.63E-04	µCi	93,000	~~ft^3~~	2.83E-02	~~m^3~~	1	~~min~~	3.90E-06	~~sec~~	=	2.79E-08	µCi
	cm^3		~~min~~	1	~~ft^3~~	60	~~sec~~		~~m^3~~			cm^3

Where:

> v = 93,000 ft^3/min is the rated flow from the Reactor Building from Design Input 5.8.
>
> *(X/Q)* = 3.90E-06 is the Particulate and Iodine dispersion coefficient for the Reactor Building from Design Input 5.8.

Values in the column labeled "Inhalation Thyroid Dose mrem" are calculated by multiplying the following factors: concentration at the receptor, the breathing rate, the time, and the dose conversion factor. The basic equation is shown in Section 6.8.

$$Dose = \sum_i (x_{ir} * B * t * h_{T50i})$$

For isotope I-131, an example is presented:

x_{ir} Receptor Conc.		Time		B Breathing Rate		h_{T50i} Thyroid			Inhalation Thyroid Dose	
2.79E-08	~~μCi~~	1	~~hr~~	1.20E+06	~~cm³~~	1.08E+03	mrem	=	3.62E+01	mrem
	~~cm³~~				~~hr~~		~~μCi~~			

Where:

h_{T50i} is the thyroid dose factor for each isotope from Section 7.1.

B = 1.20E+06 cm³/hr is the breathing rate. This value is equal to 3.33E-4 m³/sec from Design Input 5.9.

Values in the "Inhalation CEDE Dose mrem" column are calculated by multiplying the following factors: concentration at the receptor, the breathing rate, the time, and the dose conversion factor. The basic equation comes from Section 6.9.

$$Dose = \sum_i (x_{ir} * B * t * h_{E50i})$$

For isotope I-131, an example is presented:

x_{ir} Receptor Conc		Time		B Breathing Rate		h_{E50i} CEDE			Inhalation CEDE Dose	
2.79E-08	~~μCi~~	1	~~hr~~	1.20E+06	~~cm³~~	3.29E+01	mrem	=	1.10E+00	mrem
	~~cm³~~				~~hr~~		~~μCi~~			

For ease of comparison, the table row for I-131 is shown here:

Nuclide	h_{T50i} Thyroid mrem/μCi	h_{E50i} CEDE mrem/μCi	Depleted Mix Ci/MWTh	Fraction	x_{iv} Release Conc. μCi/cm³	x_{ir} Receptor Conc. μCi/cm³	Inhalation Thyroid Dose mrem	Inhalation CEDE Dose mrem
I-131	1.08E+3	3.29E+1	3.89E+2	1.34E-2	1.63E-4	2.79E-8	3.62E+1	1.10E+0

Table 16 – Inhalation Thyroid and CEDE Dose for Reactor Building

Nuclide	h_{T50i} Thyroid mrem/μCi	h_{E50i} CEDE mrem/μCi	Depleted Mix Ci/MWTh	Fraction	x_{iv} Release Conc. μCi/cm³	x_{ir} Receptor Conc. μCi/cm³	Inhalation Thyroid Dose mrem	Inhalation CEDE Dose mrem
Ba-139	8.88E-3	1.72E-1	2.84E+0	9.76E-5	1.19E-6	2.04E-10	2.17E-6	4.20E-5
Ba-140	9.47E-1	3.74E+0	3.49E+1	1.20E-3	1.46E-5	2.50E-9	2.85E-3	1.12E-2
Ce-141	1.71E-1	8.95E+0	8.10E-1	2.78E-5	3.40E-7	5.82E-11	1.19E-5	6.25E-4
Ce-143	4.48E-2	3.39E+0	6.68E-1	2.30E-5	2.80E-7	4.79E-11	2.58E-6	1.95E-4
Ce-144	6.96E+0	3.74E+2	6.56E-1	2.25E-5	2.75E-7	4.71E-11	3.93E-4	2.11E-2

Nuclide	h_{T50i} Thyroid mrem/µCi	h_{E50i} CEDE mrem/µCi	Depleted Mix Ci/MWTh	Fraction	x_{iv} Release Conc. µCi/cm^3	x_{ir} Receptor Conc. µCi/cm^3	Inhalation Thyroid Dose mrem	Inhalation CEDE Dose mrem
Cm-242	3.48E+0	1.73E+4	8.30E-3	2.85E-7	3.48E-9	5.96E-13	2.49E-6	1.23E-2
Cs-134	4.11E+1	4.63E+1	6.10E+1	2.10E-3	2.56E-5	4.38E-9	2.16E-1	2.43E-1
Cs-136	6.40E+0	7.33E+0	1.91E+1	6.57E-4	8.02E-6	1.37E-9	1.05E-2	1.21E-2
Cs-137	2.93E+1	3.19E+1	4.22E+1	1.45E-3	1.77E-5	3.03E-9	1.07E-1	1.16E-1
I-131	1.08E+3	3.29E+1	3.89E+2	1.34E-2	1.63E-4	2.79E-8	3.62E+1	1.10E+0
I-132	6.44E+0	3.81E-1	1.27E+2	4.38E-3	5.34E-5	9.15E-9	7.07E-2	4.18E-3
I-133	1.80E+2	5.85E+0	6.80E+2	2.34E-2	2.85E-4	4.88E-8	1.05E+1	3.43E-1
I-134	1.07E+0	1.31E-1	1.70E+1	5.83E-4	7.12E-6	1.22E-9	1.56E-3	1.92E-4
I-135	3.13E+1	1.23E+0	4.54E+2	1.56E-2	1.91E-4	3.26E-8	1.23E+0	4.81E-2
Kr-83m	0.00E+0	0.00E+0	1.68E+2	5.79E-3	7.06E-5	1.21E-8	0.00E+0	0.00E+0
Kr-85	0.00E+0	0.00E+0	1.02E+2	3.50E-3	4.27E-5	7.32E-9	0.00E+0	0.00E+0
Kr-85m	0.00E+0	0.00E+0	1.05E+3	3.59E-2	4.38E-4	7.50E-8	0.00E+0	0.00E+0
Kr-87	0.00E+0	0.00E+0	2.96E+2	1.02E-2	1.24E-4	2.12E-8	0.00E+0	0.00E+0
Kr-88	0.00E+0	0.00E+0	1.83E+3	6.30E-2	7.69E-4	1.32E-7	0.00E+0	0.00E+0
La-140	4.51E-1	4.85E+0	3.34E-1	1.15E-5	1.40E-7	2.40E-11	1.30E-5	1.39E-4
La-141	3.48E-2	5.81E-1	1.33E-1	4.58E-6	5.58E-8	9.55E-12	3.99E-7	6.66E-6
La-142	3.23E-2	2.53E-1	3.29E-2	1.13E-6	1.38E-8	2.36E-12	9.17E-8	7.18E-7
Mo-99	4.33E-1	3.96E+0	4.66E+0	1.60E-4	1.95E-6	3.35E-10	1.74E-4	1.59E-3
Nb-95	1.32E+0	5.81E+0	3.32E-1	1.14E-5	1.39E-7	2.39E-11	3.79E-5	1.66E-4
Nd-147	7.18E-2	6.85E+0	1.28E-1	4.40E-6	5.37E-8	9.19E-12	7.92E-7	7.55E-5
Np-239	2.82E-2	2.51E+0	9.92E+0	3.41E-4	4.16E-6	7.12E-10	2.41E-5	2.14E-3
Pr-143	6.22E-9	8.10E+0	2.91E-1	9.99E-6	1.22E-7	2.09E-11	1.56E-13	2.03E-4
Pu-241	4.59E-2	8.25E+3	7.90E-2	2.71E-6	3.31E-8	5.67E-12	3.12E-7	5.61E-2
Rb-86	4.92E+0	6.62E+0	6.81E-1	2.34E-5	2.86E-7	4.89E-11	2.89E-4	3.89E-4
Rh-105	9.51E-2	9.55E-1	2.36E+0	8.11E-5	9.90E-7	1.69E-10	1.93E-5	1.94E-4
Ru-103	2.21E+0	8.95E+0	4.01E+0	1.38E-4	1.68E-6	2.88E-10	7.63E-4	3.09E-3
Ru-105	5.55E-2	4.55E-1	1.30E+0	4.47E-5	5.45E-7	9.33E-11	6.21E-6	5.09E-5
Ru-106	5.07E+1	4.77E+2	1.44E+0	4.94E-5	6.02E-7	1.03E-10	6.27E-3	5.90E-2
Sb-127	5.55E-1	6.03E+0	4.27E+0	1.47E-4	1.79E-6	3.06E-10	2.04E-4	2.22E-3
Sb-129	7.66E-2	6.44E-1	7.32E+0	2.52E-4	3.07E-6	5.25E-10	4.83E-5	4.06E-4
Sr-89	1.54E+0	4.14E+1	1.78E+1	6.12E-4	7.47E-6	1.28E-9	2.36E-3	6.36E-2
Sr-90	9.77E+0	1.30E+3	1.77E+0	6.09E-5	7.43E-7	1.27E-10	1.49E-3	1.98E-1
Sr-91	1.51E-1	1.66E+0	1.55E+1	5.33E-4	6.50E-6	1.11E-9	2.02E-4	2.22E-3
Sr-92	8.10E-2	8.07E-1	6.69E+0	2.30E-4	2.81E-6	4.80E-10	4.67E-5	4.65E-4
Tc-99m	1.85E-1	3.26E-2	2.28E+0	7.83E-5	9.55E-7	1.64E-10	3.64E-5	6.39E-6
Te-127	2.39E-2	3.18E-1	3.02E+0	1.04E-4	1.27E-6	2.17E-10	6.22E-6	8.28E-5
Te-127m	8.84E-1	2.15E+1	7.35E-1	2.53E-5	3.08E-7	5.28E-11	5.60E-5	1.36E-3
Te-129	6.03E-3	8.95E-2	7.70E-1	2.65E-5	3.23E-7	5.53E-11	4.00E-7	5.94E-6
Te-129m	1.46E+0	2.39E+1	3.10E+0	1.07E-4	1.30E-6	2.23E-10	3.90E-4	6.39E-3
Te-131m	1.34E+2	6.40E+0	8.93E+0	3.07E-4	3.75E-6	6.41E-10	1.03E-1	4.93E-3
Te-132	2.32E+2	9.44E+0	6.76E+1	2.32E-3	2.83E-5	4.85E-9	1.35E+0	5.49E-2
Xe-131m	0.00E+0	0.00E+0	1.32E+2	4.55E-3	5.55E-5	9.49E-9	0.00E+0	0.00E+0
Xe-133	0.00E+0	0.00E+0	1.94E+4	6.66E-1	8.12E-3	1.39E-6	0.00E+0	0.00E+0
Xe-133m	0.00E+0	0.00E+0	5.90E+2	2.03E-2	2.48E-4	4.24E-8	0.00E+0	0.00E+0
Xe-135	0.00E+0	0.00E+0	3.56E+3	1.22E-1	1.49E-3	2.55E-7	0.00E+0	0.00E+0
Xe-135m	0.00E+0	0.00E+0	5.23E-3	1.80E-7	2.19E-9	3.75E-13	0.00E+0	0.00E+0
Xe-138	0.00E+0	0.00E+0	6.45E-3	2.22E-7	2.70E-9	4.63E-13	0.00E+0	0.00E+0
Y-90	3.52E-2	8.44E+0	1.72E-2	5.92E-7	7.22E-9	1.24E-12	5.22E-8	1.25E-5
Y-91	4.07E-1	4.88E+1	2.34E-1	8.06E-6	9.83E-8	1.68E-11	8.22E-6	9.86E-4
Y-92	1.37E-2	7.81E-1	9.11E-2	3.13E-6	3.82E-8	6.54E-12	1.07E-7	6.13E-6

Nuclide	h_{T50i} Thyroid mrem/µCi	h_{E50i} CEDE mrem/µCi	Depleted Mix Ci/MWTh	Fraction	x_{iv} Release Conc. µCi/cm^3	x_{ir} Receptor Conc. µCi/cm^3	Inhalation Thyroid Dose mrem	Inhalation CEDE Dose mrem
Y-93	1.87E-2	2.15E+0	1.33E-1	4.56E-6	5.56E-8	9.52E-12	2.14E-7	2.46E-5
Zr-95	5.33E+0	2.36E+1	3.28E-1	1.13E-5	1.38E-7	2.36E-11	1.51E-4	6.69E-4
Zr-97	3.54E-1	4.33E+0	2.55E-1	8.78E-6	1.07E-7	1.83E-11	7.78E-6	9.53E-5
			2.91E+4	100.00%	1.22E-2 **1.22E-2**	2.09E-6	**50 mrem Thyroid**	**2.37 mrem CEDE**

Given a radiation effluent monitor reading of 1.22E-2 µCi/cm^3, and the assumptions of the scenario, the CDE thyroid value is 50 mrem and the CEDE is 2.37 mrem.

Spreadsheet cases are run for all four release points. See Section 2.0 for results.

7.5 Resultant Dose Summary

A single spreadsheet was used to calculate EDE, CEDE, and thyroid CDE. With the given source term, when the user changes the effluent gross concentration value, the spreadsheet calculates resultant doses.

Results and variables for the reactor building case are shown below. As can be seen here, an effluent release rate of 1.22E-02 µCi/cc at the Reactor Building will result in an offsite dose of approximately 50 mrem CDE thyroid. This value corresponds to the new RA1 EAL entry threshold of 50 mrem CDE thyroid.

Dose totals are taken from the tabular spreadsheet data presented on the preceding pages.

Inhalation CEDE:	**2.37**	**mrem**
Submersion EDE:	**0.39**	**mrem**
TEDE :	**2.76**	**mrem**
Inhalation Thyroid CDE:	**49.8**	**mrem**

Release Point:	**Reactor Building**		**SBGT ?:**	**off**	
Effluent Conc.:	**1.22E-02**	µCi/cc	**Release Flow CFM:**	**93,000**	
Release: Hrs. since Rx. Shutdown:	**5**		**Hrs. Core Uncovered:**	**1**	
Exposure Time (hrs.):	**1**		**Secondary Containment Holdup Hrs.:**	**0.5**	
Hours w/ Sprays On:	**2**		cm3 per ft3:	0.0283168	
Submersion X/Q:	**4.30E-06**	sec/m³	**Inhalation X/Q:**	**3.90E-06**	sec/m3
Breathing Rate	**3.33E-4**	m³/sec	=	**1.20E+6**	cm³/hr

Spreadsheet cases are run for all four release points and for decay times of five hours.

Cases were also run for all four release points for decay times of 36 hours in consideration of EAL entry thresholds that are mode dependent.

The output for all release points and decay times are shown in Appendix 1.

See Section 2.0 for results.

8.0 Computer Software

No computer software is used in this calculation.

9.0 Impact Assessment

This calculation is based on "realistic" assumptions for the purpose of declaring EALs, rather than typical conservative "bounding" type design basis analyses. The calculation documents the EAL threshold values for specific plan monitors to assist Operations and Emergency Response personnel in determining the new basis for EALs RA1, RS1, and RG1 in accordance with NEI 99-01 Rev. 6.

Turbine Building: Modes 1, 2, and 3

Inhalation CEDE:	2.38	mRem
Submersion EDE:	0.39	mRem
TEDE :	2.77	mRem
Inhalation Thyroid CDE:	50.0	mRem

Variables

Release Point:	Turbine Building		SBGT ?:	off	
Effluent Conc.:	1.58E-02	uCi/cc	Release Flow CFM:	72,000	
Release: Hrs. since Rx. Shutdown:	5		Hrs. Core Uncovered:	1	
Exposure Time (hrs.):	1		Secondary Containment Holdup Hrs.:	0.5	
Hours w/ Sprays On:	2		cm3 per ft3:	0.0283168	
Submersion X/Q:	4.30E-06	sec/m3	Inhalation X/Q:	3.90E-06	sec/m3
Breathing Rate	3.33E-4	m3/sec	=	1.20E+6	cm3/hr

Turbine Building: Modes 4 and 5

Inhalation CEDE:	2.59	mRem
Submersion EDE:	0.07	mRem
TEDE :	2.67	mRem
Inhalation Thyroid CDE:	49.7	mRem

Variables

Release Point:	Turbine Building		SBGT ?:	off	
Effluent Conc.:	1.30E-02	uCi/cc	Release Flow CFM:	72,000	
Release: Hrs. since Rx. Shutdown:	36		Hrs. Core Uncovered:	1	
Exposure Time (hrs.):	1		Secondary Containment Holdup Hrs.:	0.5	
Hours w/ Sprays On:	2		cm3 per ft3:	0.0283168	
Submersion X/Q:	4.30E-06	sec/m3	Inhalation X/Q:	3.90E-06	sec/m3
Breathing Rate	3.33E-4	m3/sec	=	1.20E+6	cm3/hr

Reactor Building: Modes 1, 2, and 3

Inhalation CEDE:	**2.37**	**mRem**
Submersion EDE:	**0.39**	**mRem**
TEDE :	**2.76**	**mRem**
Inhalation Thyroid CDE:	**49.8**	**mRem**

Variables

Release Point:	**Reactor Building**		**SBGT ?:**	**off**	
Effluent Conc.:	**1.22E-02**	uCi/cc	**Release Flow CFM:**	**93,000**	
Release: Hrs. since Rx. Shutdown:	**5**		**Hrs. Core Uncovered:**	**1**	
Exposure Time (hrs.):	**1**		**Secondary Containment Holdup Hrs.:**	**0.5**	
Hours w/ Sprays On:	**2**		cm3 per ft3:	0.0283168	
Submersion X/Q:	**4.30E-06**	sec/m3	**Inhalation X/Q:**	**3.90E-06**	sec/m3
Breathing Rate	**3.33E-4**	m3/sec	=	**1.20E+6**	cm3/hr

Reactor Building: Modes 4 and 5

Inhalation CEDE:	**2.60**	**mRem**
Submersion EDE:	**0.07**	**mRem**
TEDE :	**2.68**	**mRem**
Inhalation Thyroid CDE:	**49.9**	**mRem**

Variables

Release Point:	**Reactor Building**		**SBGT ?:**	**off**	
Effluent Conc.:	**1.01E-02**	uCi/cc	**Release Flow CFM:**	**93,000**	
Release: Hrs. since Rx. Shutdown:	**36**		**Hrs. Core Uncovered:**	**1**	
Exposure Time (hrs.):	**1**		**Secondary Containment Holdup Hrs.:**	**0.5**	
Hours w/ Sprays On:	**2**		cm3 per ft3:	0.0283168	
Submersion X/Q:	**4.30E-06**	sec/m3	**Inhalation X/Q:**	**3.90E-06**	sec/m3
Breathing Rate	**3.33E-4**	m3/sec	=	**1.20E+6**	cm3/hr

Offgas Stack: Modes 1, 2, and 3

Inhalation CEDE:	1.96	mRem
Submersion EDE:	8.05	mRem
TEDE :	10.00	mRem
Inhalation Thyroid CDE:	41.1	mRem

Variables

Release Point:	Offgas Stack		SBGT ?:	on	
Effluent Conc.:	4.39E+01	uCi/cc	Release Flow CFM:	10,000	
Release: Hrs. since Rx. Shutdown:	5		Hrs. Core Uncovered:	1	
Exposure Time (hrs.):	1		Secondary Containment Holdup Hrs.:	<0.5	
Hours w/ Sprays On:	2		cm3 per ft3:	0.0283168	
Submersion X/Q:	2.80E-07	sec/m3	Inhalation X/Q:	3.10E-07	sec/m3
Breathing Rate	3.33E-4	m3/sec	=	1.20E+6	cm3/hr

Offgas Stack: Modes 4 and 5

Inhalation CEDE:	2.61	mRem
Submersion EDE:	1.41	mRem
TEDE :	4.02	mRem
Inhalation Thyroid CDE:	50.0	mRem

Variables

Release Point:	Offgas Stack		SBGT ?:	on	
Effluent Conc.:	4.52E+01	uCi/cc	Release Flow CFM:	10,000	
Release: Hrs. since Rx. Shutdown:	36		Hrs. Core Uncovered:	1	
Exposure Time (hrs.):	1		Secondary Containment Holdup Hrs.:	<0.5	
Hours w/ Sprays On:	2		cm3 per ft3:	0.0283168	
Submersion X/Q:	2.80E-07	sec/m3	Inhalation X/Q:	3.10E-07	sec/m3
Breathing Rate	3.33E-4	m3/sec	=	1.20E+6	cm3/hr

LLRPSF: Modes 1, 2, and 3

Inhalation CEDE:	2.37	mRem
Submersion EDE:	0.39	mRem
TEDE :	2.76	mRem
Inhalation Thyroid CDE:	49.7	mRem

Variables

Release Point:	LLRPSF		SBGT ?:	off	
Effluent Conc.:	1.51E-02	uCi/cc	Release Flow CFM:	75,000	
Release: Hrs. since Rx. Shutdown:	5		Hrs. Core Uncovered:	1	
Exposure Time (hrs.):	1		Secondary Containment Holdup Hrs.:	0.5	
Hours w/ Sprays On:	2		cm3 per ft3:	0.0283168	
Submersion X/Q:	4.30E-06	sec/m3	Inhalation X/Q:	3.90E-06	sec/m3
Breathing Rate	3.33E-4	m3/sec	=	1.20E+6	cm3/hr

LLRPSH: Modes 4 and 5

Inhalation CEDE:	2.60	mRem
Submersion EDE:	0.07	mRem
TEDE :	2.67	mRem
Inhalation Thyroid CDE:	49.8	mRem

Variables

Release Point:	LLRPSF		SBGT ?:	off	
Effluent Conc.:	1.25E-02	uCi/cc	Release Flow CFM:	75,000	
Release: Hrs. since Rx. Shutdown:	36		Hrs. Core Uncovered:	1	
Exposure Time (hrs.):	1		Secondary Containment Holdup Hrs.:	0.5	
Hours w/ Sprays On:	2		cm3 per ft3:	0.0283168	
Submersion X/Q:	4.30E-06	sec/m3	Inhalation X/Q:	3.90E-06	sec/m3
Breathing Rate	3.33E-4	m3/sec	=	1.20E+6	cm3/hr

ENERCON Excellence—Every project. Every day.	Attachment 1 CALCULATION PREPARATION CHECKLIST	CALC NO.	NEE-323-CALC-005
		REV.	00

CHECKLIST ITEMS[1]	YES	NO	N/A
GENERAL REQUIREMENTS			
1. If the calculation is being performed to a client procedure, is the procedure being used the latest revision?	☐	☐	☒
The calculation is being prepared to ENERCON's procedures.			
2. Are the proper forms being used and are they the latest revision?	☒	☐	☐
3. Have the appropriate client review forms/checklists been completed?	☐	☐	☒
The calculation is being prepared to ENERCON's procedures.			
4. Are all pages properly identified with a calculation number, calculation revision and page number consistent with the requirements of the client's procedure?	☒	☐	☐
5. Is all information legible and reproducible?	☒	☐	☐
6. Is the calculation presented in a logical and orderly manner?	☒	☐	☐
7. Is there an existing calculation that should be revised or voided?	☐	☒	☐
This is a new calculation to support implementing NEI 99-01 Rev. 6			
8. Is it possible to alter an existing calculation instead of preparing a new calculation for this situation?	☐	☒	☐
9. If an existing calculation is being used for design inputs, are the key design inputs, assumptions and engineering judgments used in that calculation valid and do they apply to the calculation revision being performed.	☐	☒	☐
10. Is the format of the calculation consistent with applicable procedures and expectations?	☒	☐	☐
11. Were design input/output documents properly updated to reference this calculation?	☐	☐	☒
12. Can the calculation logic, methodology and presentation be properly understood without referring back to the originator for clarification?	☒	☐	☐
OBJECTIVE AND SCOPE			
13. Does the calculation provide a clear concise statement of the problem and objective of the calculation?	☒	☐	☐
14. Does the calculation provide a clear statement of quality classification?	☒	☐	☐
15. Is the reason for performing and the end use of the calculation understood?	☒	☐	☐
16. Does the calculation provide the basis for information found in the plant's license basis?	☒	☐	☐
17. If so, is this documented in the calculation?	☒	☐	☐
18. Does the calculation provide the basis for information found in the plant's design basis documentation?	☐	☒	☐

	CHECKLIST ITEMS[1]	YES	NO	N/A
19.	If so, is this documented in the calculation?	☐	☐	☒
20.	Does the calculation otherwise support information found in the plant's design basis documentation?	☐	☒	☐
21.	If so, is this documented in the calculation?	☐	☐	☒
22.	Has the appropriate design or license basis documentation been revised, or has the change notice or change request documents being prepared for submittal?	☐	☐	☒
DESIGN INPUTS				
23.	Are design inputs clearly identified?	☒	☐	☐
24.	Are design inputs retrievable or have they been added as attachments?	☒	☐	☐
25.	If Attachments are used as design inputs or assumptions are the Attachments traceable and verifiable?	☐	☐	☒
26.	Are design inputs clearly distinguished from assumptions?	☒	☐	☐
27.	Does the calculation rely on Attachments for design inputs or assumptions? If yes, are the attachments properly referenced in the calculation?	☐	☒	☐
28.	Are input sources (including industry codes and standards) appropriately selected and are they consistent with the quality classification and objective of the calculation?	☒	☐	☐
29.	Are input sources (including industry codes and standards) consistent with the plant's design and license basis?	☒	☐	☐
30.	If applicable, do design inputs adequately address actual plant conditions?	☒	☐	☐
31.	Are input values reasonable and correctly applied?	☒	☐	☐
32.	Are design input sources approved?	☒	☐	☐
33.	Does the calculation reference the latest revision of the design input source?	☒	☐	☐
34.	Were all applicable plant operating modes considered?	☒	☐	☐
ASSUMPTIONS				
35.	Are assumptions reasonable/appropriate to the objective?	☒	☐	☐
36.	Is adequate justification/basis for all assumptions provided?	☒	☐	☐
37.	Are any engineering judgments used?	☐	☒	☐
38.	Are engineering judgments clearly identified as such?	☐	☐	☒
39.	If engineering judgments are utilized as design inputs, are they reasonable and can they be quantified or substantiated by reference to site or industry standards, engineering principles, physical laws or other appropriate criteria?	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
METHODOLOGY			
40. Is the methodology used in the calculation described or implied in the plant's licensing basis?	☐	☒	☐
41. If the methodology used differs from that described in the plant's licensing basis, has the appropriate license document change notice been initiated?	☐	☐	☒
42. Is the methodology used consistent with the stated objective?	☒	☐	☐
43. Is the methodology used appropriate when considering the quality classification of the calculation and intended use of the results?	☒	☐	☐
BODY OF CALCULATION			
44. Are equations used in the calculation consistent with recognized engineering practice and the plant's design and license basis?	☒	☐	☐
45. Is there reasonable justification provided for the use of equations not in common use?	☐	☐	☒
46. Are the mathematical operations performed properly and documented in a logical fashion?	☒	☐	☐
47. Is the math performed correctly?	☒	☐	☐
48. Have adjustment factors, uncertainties and empirical correlations used in the analysis been correctly applied?	☒	☐	☐
49. Has proper consideration been given to results that may be overly sensitive to very small changes in input?	☒	☐	☐
SOFTWARE/COMPUTER CODES			
50. Are computer codes or software languages used in the preparation of the calculation?	☐	☒	☐
51. Have the requirements of CSP 3.09 for use of computer codes or software languages, including verification of accuracy and applicability been met?	☐	☐	☒
52. Are the codes properly identified along with source vendor, organization, and revision level?	☐	☐	☒
53. Is the computer code applicable for the analysis being performed?	☐	☐	☒
54. If applicable, does the computer model adequately consider actual plant conditions?	☐	☐	☒
55. Are the inputs to the computer code clearly identified and consistent with the inputs and assumptions documented in the calculation?	☐	☐	☒
56. Is the computer output clearly identified?	☐	☐	☒
57. Does the computer output clearly identify the appropriate units?	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
58. Are the computer outputs reasonable when compared to the inputs and what was expected?	☐	☐	☒
59. Was the computer output reviewed for ERROR or WARNING messages that could invalidate the results?	☐	☐	☒
RESULTS AND CONCLUSIONS			
60. Is adequate acceptance criteria specified?	☐	☐	☒
61. Are the stated acceptance criteria consistent with the purpose of the calculation, and intended use?	☒	☐	☐
62. Are the stated acceptance criteria consistent with the plant's design basis, applicable licensing commitments and industry codes, and standards?	☒	☐	☐
63. Do the calculation results and conclusions meet the stated acceptance criteria?	☒	☐	☐
64. Are the results represented in the proper units with an appropriate tolerance, if applicable?	☒	☐	☐
65. Are the calculation results and conclusions reasonable when considered against the stated inputs and objectives?	☒	☐	☐
66. Is sufficient conservatism applied to the outputs and conclusions?	☒	☐	☐
67. Do the calculation results and conclusions affect any other calculations?	☐	☒	☐
68. If so, have the affected calculations been revised?	☐	☐	☒
69. Does the calculation contain any conceptual, unconfirmed or open assumptions requiring later confirmation?	☐	☒	☐
70. If so, are they properly identified?	☐	☐	☒
DESIGN REVIEW			
71. Have alternate calculation methods been used to verify calculation results?	☐	☐	☒
No, a Design Review was performed.			

Note:

1. Where required, provide clarification/justification for answers to the questions in the space provided below each question. An explanation is required for any questions answered as "No' or "N/A".

Originator: Ryan Skaggs 12/14/17

Print Name and Sign Date

BEFORE

SF/L-4 ☐ Spent Fuel Pool level drops to **16.36 ft**

↓

SF/L-5 ☐ Operate Spent Fuel Pool sprays (SAMP 712).

☛ Use only systems not required for adequate core cooling.

DISCUSSION

If spent fuel pool level cannot be controlled using alternate or external makeup sources, sprays are used to add water to the spent fuel pool, cool exposed bundles, and reduce radioactivity releases. However, spray operation may damage electrical equipment and flood lower elevations of the secondary containment, complicating implementation of other emergency response strategies, and runoff from sprays could spread radioactivity release. Use of sprays is therefore delayed until it is determined that spent fuel pool level cannot be maintained above the top of the fuel racks. As long as the spent fuel assemblies are covered with water , the fuel will not overheat and efforts should focus on providing sufficient makeup flow to keep the assemblies submerged.

The lowest measurable spent fuel pool level using the wide range instrument is 16.16 ft., approximately one foot above the top of the spent fuel racks. The action level in SF/L-4 corresponds to NEI 12-02 Level 3, the level at which fuel remains covered but actions to implement make-up water addition should no longer be deferred. The "before" condition permits appropriate anticipatory action based on the spent fuel pool leakage rate, radiation levels, available resources, and the time required to place sprays in service. Steps to prepare spray equipment for use should be initiated while radiation levels permit access to the refueling floor and timed to optimize use of available resources.

As in Steps SF/T-3 and SF/L-3, available spray sources may be alternated between RPV injection and spent fuel pool spray modes as long as adequate core cooling can be maintained, but maintaining adequate core cooling takes precedence over spent fuel pool cooling (refer to the discussions of Steps SF/T-3 and SF/L-3 above).

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources — Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred and alternate preferred), and the onsite standby power sources (Diesel | Generators (DGs) 1G-31 and 1G-21). As discussed in UFSAR Section 3.1.2.2.8 (Ref. 1), the design of the AC Electrical Power System provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) Systems via essential buses 1A3 and 1A4.

The Class 1E AC Distribution System is divided into redundant load groups, so loss of any one group does not prevent the minimum safety functions from being performed. Each load group has connections to two preferred offsite power supplies and a single DG.

Offsite power is supplied to the 161 kV and 345 kV switchyards from the transmission network by six transmission lines. The 345 kV switchyard and the 161 kV switchyard are connected via the autotransformer, and both sections of the switchyard are connected to the transmission grid by at least two independent lines. From the 161 kV switchyard (the preferred power source), a single overhead transmission line feeds the startup transformer. From the startup transformer, dual isolated secondary windings provide feeds to the 4160 volt essential buses, 1A3 and 1A4, through separate bus supply lines and circuit breakers. The startup transformer is sized to supply all plant power (both essential and non-essential loads) during unit startup. From the tertiary winding on the autotransformer (the alternate preferred power source), a single 34.5 kV underground line feeds the standby transformer. From the standby transformer, a single 4160 volt line feeds both essential buses through separate bus supply circuit breakers. A detailed description of the offsite power network and circuits to the onsite Class 1E essential buses is found in the UFSAR, Sections 8.2.1.3 and 8.3.1.1.5 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls

(continued)

BASES

BACKGROUND (continued)

required to transmit power from the offsite transmission network to the onsite Class 1E essential bus or buses. Startup transformer (1X3) provides the normal source of power to the essential buses 1A3 and 1A4. If either 4.16 kV essential bus loses power, an automatic transfer from the startup transformer to the standby transformer (1X4) occurs.

The startup transformer and standby transformer are both sized to accommodate the starting of all ESF loads on receipt of an accident signal. Emergency loads are sequenced onto the essential buses regardless of the source of power (onsite or offsite).

The onsite standby power source for 4.16 kV essential buses 1A3 and 1A4 consists of two DGs. DGs 1G-31 and 1G-21 are dedicated to essential buses 1A3 and 1A4, respectively. A DG starts automatically on a Loss of Coolant Accident (LOCA) signal (i.e., low reactor water level signal or high drywell pressure signal) or on an essential bus degraded voltage or undervoltage signal. After the DG has started, it automatically ties to its respective bus after offsite power is tripped as a consequence of essential bus undervoltage or degraded voltage, independent of or coincident with a LOCA signal. The DGs also start and operate in the standby mode without tying to the essential bus on a LOCA signal alone. Following the trip of offsite power, non emergency loads powered from essential buses are load shed. When the DG is tied to the essential bus, loads are then sequentially connected to its respective essential bus. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG.

In the event of a loss of both the preferred power source and the alternate preferred power source, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a LOCA.

Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading of the DGs in the process. Within 25 seconds after the initiating signal is received, all automatic and permanently

(continued)

ANNUNCIATOR RESPONSE PROCEDURE

ARP 1C08A

GENERATOR AND AUXILIARY POWER

Usage Level
Reference Use

Record the following: Date/Time: ____________ / ________ Initials: _______

NOTE: *User shall perform and document a Temp Issue/Rev. Check to ensure revision is current, in accordance with procedure use and adherence requirements.*

Prepared By: ____________________ / ____________________________ Date: __________
Print Signature

CROSS-DISCIPLINE REVIEW (AS REQUIRED)		
Reviewed By: ____________________ Print	/ ____________________________ Signature	Date: __________
Reviewed By: ____________________ Print	/ ____________________________ Signature	Date: __________

PROCEDURE APPROVAL		
Approved By ____________________ Print	/ ____________________________ Signature	Date: __________

ALARM WINDOW ENGRAVINGS AND GRID LAYOUT

1C08A

Same on annunciator panel 1C08B for Division 2

	1	2	3	4	5	6	7	8	9	10	11	12
A	AUX XFMR TO 1A1 BREAKER 1A101 TRIP	BUS 1A1 LOCKOUT TRIP OR LOSS OF VOLTAGE	S/U XFMR TO 1A1 BREAKER 1A102 TRIP	STBY XFMR TO 1A3 BREAKER 1A301 TRIP	BUS 1A3 LOCKOUT TRIP	S/U XFMR TO 1A3 BREAKER 1A302 TRIP	STARTUP XFMR 1X3 TROUBLE	UNINTERRUPTIBLE AC 1Y23 UNDERVOLTAGE OR INVERTER TROUBLE	125 VDC SYSTEM 1 TROUBLE	"A" DIESEL GEN 1G-31 RUNNING	A DG TO BUS 1A3 BREAKER 1A311 TRIP	"A" DIESEL GEN 1G-31 LOCKOUT TRIP
B	1A1 TO XFMR 1X11 BREAKER 1A107 TRIP	1A1 TO XFMR 1X71 BREAKER 1A108 TRIP	1A1 TO XFMR 1X51 BREAKER 1A109 TRIP	SWITCHYARD SUPPLY BREAKER 1A110 TRIP	LC XFMR 1X31 BREAKER 1A303 TRIP	LC XFMR 1X91 BREAKER 1A312 OR MCC 1B91 BKR 1B903 TRIP	MAIN GENERATOR IMPROPER PHASE SEQUENCE	INSTRUMENT AC 1Y21 UNDERVOLTAGE OR INVERTER TROUBLE	125 VDC CHARGER 1D12 TROUBLE	"A" DIESEL GEN FUEL OIL DAY TANK 1T-37A LO-LO-LEVEL	"A" DIESEL GEN 1G-31 PHASE OVERCURRENT OR GROUND FAULT	"A" DIESEL GEN 1G-31 OVERSPEED TRIP
C	XFMR 1X11 TO LC1B1 BREAKER 1B101 TRIP	LC 1B1/1B2 CROSS TIE BREAKER 1B107 TRIP	XFMR 1X51 TO LC 1B5 BREAKER 1B501 TRIP	125 VDC SYSTEM 1 BATTERY 1D1 DISCONNECTED	LC 1B3 BREAKER 1B301, 1B302 1B303 OR 1B304 TRIP	BUS 1A3 LOSS OF VOLTAGE	STARTUP XFMR LOCKOUT TRIP	INSTRUMENT AC 1Y11 UNDERVOLTAGE OR INVERTER TROUBLE	125 VDC CHARGER 1D120 TROUBLE	AUX BOILER FUEL TANK 1T-34 LO LEVEL	"A" DIESEL GEN PANEL 1C-93 TROUBLE	"A" DIESEL GEN 1G-31 ENGINE CRANKING
D	LC 1B1 BREAKER 1B102, 1B103 1B104 OR 1B105 TRIP	LC 1B5/1B6 CROSSTIE BREAKER 1B505 TRIP	LOAD CENTER 1B5 BREAKER 1B502 1B503 OR 1B504 TRIP		MCC 1B34A TIE BKR 1B3401 TRIP	MCC 1B34A/1B44A TIE BREAKER 1B3402 OR 1B4402 TRIP	4KV BUS AUTO TRANSFER INOP	DIESEL FUEL OIL STORAGE TANK 1T-35 LO LEVEL	"A" DIESEL GEN 1G-31 CONTROL POWER FAILURE	"A" DIESEL GEN 1G-31 AUTO START INHIBITED	"A" DIESEL GEN 1G-31 ENGINE SHUTDOWN	"A" DIESEL GEN 1G-31 START FAILURE

125V DC SYSTEM 1 TROUBLE

TITLE: 125 VDC SYSTEM 1 TROUBLE (GROUND OR LOW VOLTAGE)

1.0 PROBABLE CAUSE(S) / INITIATING DEVICE(S) / SETPOINT(S)

	PROBABLE CAUSE(S)	INITIATING DEVICE(S)	SETPOINT(S)
1.1	Ground fault on 125 VDC System 1	Positive to ground Relay 64 or negative to ground	9 Volt differential Relay 64
1.2	125 VDC System 1 low voltage	Relay 1D10-27	105 VDC (dec)
1.3	Positive or negative metering fuse blown	FU 3 amp fuse blown	blown

2.0 AUTOMATIC ACTIONS

2.1 If due to a complete loss of 125 VDC System 1:

a. Various control systems half trip.

b. Scoop Tube for Recirc Pump A locks up.

c. Breaker 1B3401 auto trips open after 6 second time delay, 1B4401 auto closes.

d. Static Switch JS1501 transfers from Inverter 1D15 to Regulating Transformer 1Y1A.

2.2 If the 125 VDC System 1 is not lost, no AUTOMATIC ACTIONS occur.

125V DC
CHARGER 1D12
TROUBLE

TITLE: 125 VDC CHARGER 1D12 TROUBLE

> **NOTE**
> This is a normal alarm anytime Charger 1D12 is being changed over to Charger 1D120.

1.0 PROBABLE CAUSE(S) / INITIATING DEVICE(S) / SETPOINT(S)

	PROBABLE CAUSE(S)	INITIATING DEVICE(S)	SETPOINT(S)
1.1	AC Breaker 1D12-01 open	Relay K-8	OPEN position
1.2	DC Breaker 1D12-02 open	Relay K-7	OPEN position
	AND		
	DC Breaker 1D12-03 open	Relay K-7	OPEN position
1.3	Charger Failure	Relay K-5	< 5 Amps (dec) 40 second time delay
1.4	Reverse Current	Relay K-4	Reverse Current Detected
1.5	DC Undervoltage	Relay K-2	105 VDC (dec)
1.6	AC Undervoltage	Relay K-1	340 VAC (dec)

2.0 AUTOMATIC ACTIONS

2.1 If due to CAUSE 1.3, 1.4, or 1.5, Charger 1D12 front panel trouble light illuminates.

**125V DC
CHARGER 1D120
TROUBLE**

TITLE: 125 VDC CHARGER 1D120 TROUBLE

NOTE

This is a normal alarm anytime 1D120 is being changed over to Chargers 1D12 or 1D22.

1.0 PROBABLE CAUSE(S) / INITIATING DEVICE(S) / SETPOINT(S)

	PROBABLE CAUSE(S)	INITIATING DEVICE(S)	SETPOINT(S)
1.1	AC Breaker 1D120-01 open	Relay K-8	Open Position
1.2	DC Breaker 1D120-02 open	Relay K-7	OPEN Position
1.3	DC Breaker 1D120-03 open	Relay K-7	OPEN Position
1.4	Charger Failure	Relay K5	< 5 Amps (dec) 40 Second TD
1.5	Reverse Current	Relay K-4	Reverse Current Detected
1.6	DC Undervoltage	Relay K-2	105 VDC (dec)
1.7	AC Undervoltage	Relay K-1	340 VAC (dec)

2.0 AUTOMATIC ACTIONS

2.1 If due to CAUSE 1.4, 1.5, or 1.6, Charger 1D120 front panel trouble light illuminates.

Table B 3.8.7-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	DIVISION 1[a]	DIVISION 2[a]
AC safety buses	4160 V	Essential Bus 1A3	Essential Bus 1A4
	480 V	Load Centers 1B3, 1B9	Load Centers 1B4, 1B20
	480 V	Motor Control Centers 1B32, 1B34	Motor Control Centers 1B42, 1B44
125 VDC buses	125 V	Distribution Panels 1D10, 1D11, 1D13 RCIC Motor Control Center 1D14	Distribution Panels 1D20, 1D21, 1D23
250 VDC buses	250 V	N/A	Distribution Panel 1D40 Motor Control Centers 1D41 and 1D42

[a] Each division of the AC and DC electrical power distribution systems is a subsystem.

ABNORMAL OPERATING PROCEDURE

AOP 302.1

LOSS OF 125 VDC POWER

Usage Level
Reference Use

NOTE

This AOP is normally coordinated by the Reactor Operator.

Record the following: Date/Time: _____________ / _________ Initials: _______

NOTE: *User shall perform and document a Temp Issue/Rev. Check to ensure revision is current, in accordance with procedure use and adherence requirements.*

Enter the following as applicable:

IMMEDIATE ACTIONS	
1. Place RX WATER LEVEL CONTROL INPUT SELECT HSS-4560 in A LEVEL at 1C05.	______
2. Place 1 OR 3 ELEMENT CONTROL SELECT HSS 4450 in 1 ELEMENT at 1C05.	______

AUTOMATIC ACTIONS

- 1G001 Voltage Regulator transfers to manual with no adjustment capability
- Loss of 1A1/1A3 breaker control
- MG Set A Scoop Tube Power Failure Lockout initiates (no Amber lockup light)
- SBGTS A SV-5801A fails closed, SV-5815A, SV-5817A and SV-5825A fail open
- MCC 1B34A/1B44A will auto transfer to 1B44
- JS1501 will transfer from Inverter 1D15 to Reg Trans 1Y1A
- Group 3A Primary Containment Isolation (Lockout relay will not trip)
- Rx FEEDWATER FLOW B FEEDLINE FI-1626 fails downscale
- STEAM FLOW B STEAMLINE FI-4409 fails downscale
- Recirc Lube Oil Pumps 1P202A & 1P202B trip, and 1P202C auto starts on low oil pressure (continued operation of the A Recirc MG Set is allowable in this condition).
- B GEMAC LEVEL LI4560 fails low
- HPCI will not trip on high level
- Loss of 1D10 with B LEVEL selected and no operator action results in:
  - Feedwater control opens Feed Reg Valves
  - Reactor water level goes high
  - "B" Reactor Feed Pump and Main Turbine Trip on high level ("A" Reactor Feed Pump cannot be tripped remotely)
  - Reactor Scram (Main turbine trip)
  - Loss of 1A1 and 1A2 power (failure to auto transfer)
  - If >26% power, Reactor Recirc. Pumps 1P-201A and 1P-201B trip (RPT)

FOLLOW-UP ACTIONS

NOTE

Follow up actions may be performed in any order.

1. Establish critical parameter monitoring of RPV Water Level, as priorities allow.

2. Stabilize reactor power level and maintain recirculation loop flows balanced. Use manual control of the A MG Set and Recirc Pump from the MG Set Room in accordance with OI 264.

NOTE

Buses 1A1 and 1A2 will not auto transfer to the startup transformer on a turbine trip.

3. Transfer Bus 1A2 to the startup transformer per OI 304.1.

4. **IF** 1A4 has power available **THEN** verify TIE BREAKER 1B4401 MCC 1B44A/1B34A is closed.

5. **IF** a Reactor Scram occurs: **THEN** perform the following:

   a. Verify main turbine trip.

   b. Verify the H and I breakers open.

   c. Only after H and I are open, direct an operator to trip the GENERATOR EXCITER FIELD BREAKER locally.

6. Reference EPIP 1.1 for EAL assessment, for instrumentation, perform alarm panel checks as needed to confirm threshold is met.

7. Suspend all evolutions in progress associated with electrical switchgear and switching operations.

8. Locally operate affected switchgear to start and stop equipment as required.

FOLLOW-UP ACTIONS (continued)

NOTE

Loss of 125 VDC DIV 1 causes a loss of 1A1 control power. If 1A1 is on the aux transformer, all 1A1 loads will remain energized with no automatic trips or starts until the Main Generator is tripped. Likewise, when the main generator is tripped, All 1A1 loads will be lost until manually restored.

1A1 loads:

- A Feed Pump (1A103) – no 211" trip
- A Condensate Pump (1A106)
- A Recirc MG Set (1A104)
- A Circ Water Pump (1A105)
- Load Center 1B1 (1A107)
- Load Center 1B5 (1A109)
- Load Center 1B7 (1A108)

9. **IF** 1A1 is deenergized **THEN** reenergize 1A1 locally:

   a. Trip 1A101.

   b. Strip 1A1 loads.

   c. Close 1A102.

   d. Restart loads as required.

NOTE

If operation of the RCIC System is required while the Division I 125 VDC System is deenergized, use the HPCI System in manual mode.

10. Evaluate the status of the 125 VDC Electrical Distribution System to determine the cause of the malfunction.

FOLLOW-UP ACTIONS (continued)

11. **IF** 1D10 is totally deenergized **THEN** perform the following:

    a. Open all branch circuit breakers at Panels 1D10, 1D11, 1D13, and 1D14. ________

    b. Verify MCC 1B32 energized. ________

    c. Locally inspect circuit breakers at Panels 1D10, 1D11, 1D13, and 1D14. ________

    d. Verify Battery Room ventilation. ________

    e. Locally inspect Battery 1D1 ________

    f. Locally inspect Battery Chargers 1D12 and 1D120. ________

    g. Restore power to 1D10 and branch panels. ________

    h. Comply with Tech Spec Requirements for "Distribution Systems - Operating" or "Distribution Systems - Shutdown", as applicable. ________

    i. Comply with ACP 1412.4 Impairments to Fire Protection System. ________

12.	**IF**	1D10 is not lost, but one or more loads are lost	**THEN**	investigate and evaluate the status of the individually affected loads and comply with Tech Spec Requirements for "Distribution Systems - Operating" or "Distribution Systems - Shutdown", as applicable.	________
13.	**IF**	125 VDC Div I Battery 1D1 is made or found to be inoperable	**THEN**	comply with Tech Spec Requirements for "DC Sources - Operating" or "DC Sources - Shutdown", as applicable.	________
14.	**IF**	125 VDC Div I Battery 1D1 is < 110 VDC	**THEN**	Verify battery cell parameters per STP 3.8.4-02 within 24 hours.	________
15.	**WHEN**	Div I 125 VDC System is restored and the use of-TIE BREAKER 1B3401 is desired	**THEN**	send an operator to TIE BREAKER 1B3401 MCC 1B34A/44A to press reset button prior to closing.	________
16.	**WHEN**	power is restored	**THEN**	reset A Scoop Tube lockout per OI 264.	________

PROBABLE ANNUNCIATORS

1C08A,	A-2	Bus 1A1 LOCKOUT TRIP OR LOSS OF VOLTAGE
	A-7	STARTUP XFMR 1X3 TROUBLE
	A-9	125 VDC SYSTEM 1 TROUBLE
	B-9	125 VDC CHARGER 1D12 TROUBLE
	C-4	125 VDC SYSTEM 1 BATTERY 1D1 DISCONNECTED
	C-6	BUS 1A3 LOSS OF VOLTAGE
	C-8	INSTRUMENT AC 1Y11 UNDERVOLTAGE OR INVERTER TROUBLE
	C-9	125 VDC CHARGER 1D120 TROUBLE
	D-5	MCC 1B34A TIE BKR 1B3401 TRIP
	D-9	A DIESEL GEN 1G-31 CONTROL POWER FAILURE
1C08C,	A-3	MAIN GENERATOR LOCKOUT RELAY CKT LOSS OF 125 VDC
	B-3	MAIN GENERATOR VOLTAGE REGULATION IN MANUAL
	B-6	H_2/STATOR COOLING PANEL 1C83 LOSS OF DC

PROBABLE INDICATIONS

Annunciators - Loss of power to the following panels:

1C03	1C05	1C23A	1C25A	1C34	1C40
1C04	1C08B	1C24A	1C26A	1C35A	1C40A
1C14	1C09A	1C22			

1C08 - Loss of the following:

- 4160V BUS 1A1 switchgear control, indication and automatic trip functions
- 480V LC 1B1 switchgear control and indication
- 480V LC 1B7 switchgear control and indication
- 480V LC 1B5 switchgear control and indication
- 4160V BUS 1A3 switchgear control, indication and automatic trip functions
- 480V LC 1B3 switchgear control and indication
- 480V LC 1B9 switchgear control and indication
- TIE BREAKER 1B3401 MCC 1B34A/1B44A control and indication
- GENERATOR EXCITER FIELD BREAKER control and indication

1C07 - Loss of the following:

- EMERGENCY BEARING OIL PUMP 1P-40 control and indication

PROBABLE INDICATIONS (continued)

<u>1C06</u> - Loss of the following:

- A CIRC WATER PUMP 1P-4A control and indication
- A CONDENSATE PUMP 1P-8A control and indication
- A REACTOR FEEDWATER PUMP 1P-1A control and indication
- A GSW PUMP 1P-89A control and indication
- A[C] RWS PUMP 1P-117A and C control and indication

<u>1C05</u> - Loss of the following:

- A WIDE RANGE LEVEL LI-4539 indication fails low
- B GEMAC LEVEL LI-4560 fails low
- B REACTOR PRESS PI-4564 fails low
- A CRD PUMP 1P-209A control and indication
- Rx FEEDWATER FLOW B FEEDLINE FI-1626 fails downscale
- STEAM FLOW B STEAMLINE FI-4409 fails downscale

<u>1C04</u> - Loss of the following:

- MG SET LUBE OIL PUMP 1P-202A control and indication
- MG SET LUBE OIL PUMP 1P-202B control and indication
- A MG SET SPEED CONTROL control and indication
- A MG SET EMERG AUX OIL PUMP 1P-204A control and indication
- RCIC TURBINE CONTROL VALVE HV-2406 position indication
- RCIC System Drain Valves RCIC STEAM LINE DRAIN ISOL CV-2410, and CLOSED RADWASTE DISCH ISOL CV-2435
- RCIC System Condensate Pump Motor and Motor Operated Valves control and indication
- Inboard MSIVs indication and DC Solenoid Valve control

<u>1C03</u> - Loss of the following:

- A CORE SPRAY PUMP 1P-211A control and indication
- A RHR PUMP 1P-229A and C RHR PUMP 1P-229C control and indication
- A RHRSW PUMP 1P-22A and C RHRSW PUMP 1P-22C control and indication
- A RHR HX SHELL OUTBD VENT MO-2044A and A RHR HX SHELL INBD VENT MO-2044B percent indication
- HPCI STEAM LINE DRAIN ISOL CV-2211 and CLOSED RADWASTE DISCH ISOL CV-2234 control and indication

INFORMATION

- 1D10 125 VDC Distribution Panel loads:

    1D10 ckt 04 1D13 125 VDC Distribution Panel C
    1D10 ckt 05 1D14 125 VDC RCIC MCC
    1D10 ckt 06 1D11 125 VDC Distribution Panel A
    1D10 ckt 07 Instrument AC Inverter 1D15 Supply

- 1D11 125 VDC Distribution Panel A loads:

    1D11 ckt 01 Load Center 1B1 switchgear control
    1D11 ckt 02 RWCU F/D Panel 1C82 annunciators
    1D11 ckt 03 Load Center 1B5 switchgear control
    1D11 ckt 04 Load Center 1B7 switchgear control
    1D11 ckt 05 4160V Bus 1A1 switchgear control
        4KV BREAKER 1A101 AUX XFMR TO BUS 1A1
        4KV BREAKER 1A102 STARTUP XFMR TO BUS 1A1
        Reactor Feed Pump 1P-1A Supply Breaker 1A103
        Reactor Recirculation MG Set 1G-201A Supply Breaker 1A104
        Circulating Water Pump 1P-4A Supply Breaker 1A105
        Condensate Pump 1P-8A Supply Breaker 1A106
        FEEDER BREAKER 1A107 1A1 TO LC XFMR 1X11
        FEEDER BREAKER 1A108 1A1 TO LC XFMR 1X71
        FEEDER BREAKER 1A109 1A1 TO LC XFMR 1X51
        FEEDER BREAKER 1A110 1A1 TO SWYD LOAD CENTER
        Reactor pressure Channel B calculation and indication
        Reactor water level Channel B calculation and indication
    1D11 ckt 06 Load Center 1B9 switchgear control
    1D11 ckt 07 MCC breaker 1B3401 control (Normal)
    1D11 ckt 08 Main Generator excitation control
        Generator Exciter Field Breaker control and indication
        Motor Driven DC Regulator Setpoint Adjust (1C08)
        Motor Driven AC Regulator Setpoint Adjust (1C08)
        Exciter Field Flashing
        Regulator Transfer and Lockout Relay
        Exciter Field Bridge Overcurrent Alarm
        Generator Field Bridge Over temperature Alarm
        Exciter Field Current Limit Circuit
        Volts/Hertz Protective Panel
    1D11 ckt 09 Generator H_2 Cooling Panel 1C83
        Associated Generator Trip and Load Runback Relays Annunciators

INFORMATION

- 1D11 125 VDC Distribution Panel A loads (continued):

  1D11 ckt 10 Main Transformer 1X1 control power
  1D11 ckt 11 1G-31 Diesel Gen. Control Panel 1C117
  1D11 ckt 12 1G-31 Diesel Gen. Control Panel 1C117
  1D11 ckt 13 Startup Transformer 1X3 control power
  1D11 ckt 15 Core Spray Channel A Relay Vertical Board 1C43
  Core Spray System Loop A Logic
  1D11 ckt 17 Radwaste Panel 1C84 annunciators
  1D11 ckt 18 1G-31 Diesel Gen. Exciter Panel 1C93
  1D11 ckt 19 Standby Gas Treatment System Control Panel 1C24A control
  PASS System Valves SV-4594A, SV-4595A, and SV-8772A (FC)
  A SBGTS valves SV-5801A, SV-5815A, SV-5817A, and SV-5825A (CV-5815A, CV-5817A, and CV-5825A (FO) (AV5801 (FC)
  A SBGTS vent shaft Rad Monitor Aux Relay 95-K134A (PCIS GP 3A input)
  A SBGTS Fire Deluge SV-5837A (CV-5837A (FC))
  A SBGTS Preheater control
  (TORUS) EXTERNAL VACUUM BKR ISOL (CV-4304 (FO))
  CAMS Loop A Isolation Valve control and indication
  Offgas Stack Exhaust Fan 1V-EF-18A remote control
  Panel 1C24A annunciators
  Panel 1C25A annunciators
  1D11 ckt 20 Turbine Building and Control Room HVAC Panel 1C26
  SFU Fire Deluge SV-7328A (CV-7328A (FC))
  A DIESEL GENERATOR 1G-31 Room Supply Fan 1V-SF-20 remote control
  A DIESEL GENERATOR 1G-31 Room Supply Fan 1V-SF-20 dampers DO-7001A and DO-7002A position indication
  1V-SFU-30A valves CV-7301A and SV-7318A (AV-7301A and AV-7318A (FO))
  Miscellaneous Reactor, Turbine, and Control Building isolation dampers
  1C23A annunciators
  1C26A annunciators and indications

INFORMATION

- 1D13 125V DC Distribution Panel C loads:

1D13 ckt 01 Reactor Recirculation Pump MG Set 1G-201A Control Panel 1C112A
and Scoop Tube Power Failure Lock circuitry
1D13 ckt 13 Recirculation Pump MG Set 1G-201A control (Normal and Standby Power)
MG Set A Emergency Lube Oil Pump 1P-204A control and indication
Loss of Division I ATWS/ARI/RPT Trip System (1D1313 only)

1D13 ckt 02 Reactor Core Cooling Benchboard 1C03
RHR Heat Exchanger A Vent Valves MO-2044A and MO-2044B position indication (ZI-2044A and ZI-2044B)
HPCI System Drain Valves SV-2211 and SV-2234 (CV-2211 and CV-2234) control and indication
Position indication for CV-4309 (ZS-4309)

1D13 ckt 03 Reactor Water Cleanup and Recirculation Benchboard 1C04
RCIC Inverter
RCIC Governor Valve HV-2406 Position indication
RCIC System Drain Valves SV-2410, and SV-2435 (CV-2410 and CV-2435) control and indication

1D13 ckt 04 Annunciator Power Panels 1C03, 1C04, 1C05, 1C08B, 1C34, Panel 1C22 Frequency Converter

1D13 ckt 05 CAD Panel 1C35A, CAM Panel 1C09A
SV-4332A, Upper Drywell Spray CAD N2 Primary Containment Isolation
SV-4334A North Torus Spray Header Primary Containment Isol
SV-4332B, and SV-4334B control and indication
1C35A Annunciators
1C09A Annunciators
1C014A EOP Annunciators

1D13 ckt 06 1C40 Annunciators

1D13 ckt 07 1C32 Channel A RHR Relay Vertical Board
RHR Loop A Logic
HPCI Low Water Level initiation signal
HPCI Isolation Logic A
HPCI Rx Hi-Level trip logic (1/2 of logic, HPCI will not trip on Hi-Level)

INFORMATION

- 1D13 125 VDC Distribution Panel C loads:

1D13 ckt 08 Reactor Protection System Channel A Vertical Board 1C15
Recirculation Pump A Trip circuitry
Backup Scram Valve SV-1840A (FC)
1D13 ckt 09 Inboard Isolation Valve Relay Panel 1C41
Inboard MSIVs position indication and DC solenoid valve control
1D13 ckt 10 1C40A Annunciators
1D13 ckt 11 EBO Pump 1P-40 Starter Control (HS-3151) & Indication
Emergency Bearing Oil Pump 1P-40 control and indication
1D13 ckt 12 Generator and Plant Relay Panel 1C31
Generator Primary Lockout Relay 286/P
Startup Transformer - Bus 1A3 Breaker 1A302 Lockout Relay
Essential Bus 1A3 Load Shedding Circuit
Non-Essential Auto Transfer and Load Shed
1D13 ckt 14 Auto Blowdown Panel 1C45
ADS main control power
1D13 ckt 15 4160V Bus 1A3 switchgear control
4KV BREAKER 1A301 STANDBY TRANSFORMER TO BUS 1A3
4KV BREAKER 1A302 STARTUP TRANSFORMER TO BUS 1A3
FEEDER BREAKER 1A303 1A3 TO LC XFMR 1X31
Core Spray Pump 1P-211A Supply Breaker 1A304
RHR Pump 1P-229A Supply Breaker 1A305
RHR Pump 1P-229C Supply Breaker 1A306
RHR SW Pump 1P-22A Supply Breaker 1A307
RHR SW Pump 1P-22C Supply Breaker 1A308
GSW Pump 1P-89A Supply Breaker 1A309
CRD Pump 1P-209A Supply Breaker 1A310
A DG TO BUS 1A3 BREAKER 1A311
FEEDER BREAKER 1A312, 1A3 TO LC XFMR 1X91
Essential Bus 1A3 Degraded Voltage Detection Circuit
1D13 ckt 16 Load Center 1B3 switchgear control
1D13 ckt 17 RCIC Relay Vertical Board 1C30
RCIC Turbine Speed Controller
RCIC Turbine Trip circuitry
RCIC Initiation and Isolation Relay Logic A
RCIC Instrumentation
1D13 ckt 19 NSSS Temperature and Leak Detection Panel 1C21
RCIC Area Steam Leak Detection circuitry Division I
RCIC Timer Logic Division I
1D13 ckt 20 Remote Shutdown Panels 1C389 and 1C390
Transfer Switch Position Status Indication

INFORMATION

- 1D14 125 VDC RCIC MCC loads:

  1D14 ckt 01 Steam Outboard Isolation Valve MO-2401
  1D14 ckt 02 Steam Supply Valve MO-2404
  1D14 ckt 03 Turbine Stop Valve MO-2405
  1D14 ckt 04 Bypass to Condensate Valve MO-2426
  1D14 ckt 05 Suction from Condensate Storage Tank Valve MO-2500
  1D14 ckt 06 Minimum Flow Bypass Valve MO-2510
  1D14 ckt 07 Normally Open Discharge Valve MO-2511
  1D14 ckt 08 Normally Closed Discharge Valve MO-2512
  1D14 ckt 09 Test Discharge Valve MO-2515
  1D14 ckt 10 Suction at Pool Valve MO-2516
  1D14 ckt 11 Suppression Pool Suction Valve MO-2517
  1D14 ckt 12 Gland Seal Vacuum Pump 1P-227
  1D14 ckt 13 Gland Seal Vacuum Tank Condensate Pump 1P-228

FIGURE F-5
DAEC TELEPHONE SYSTEMS

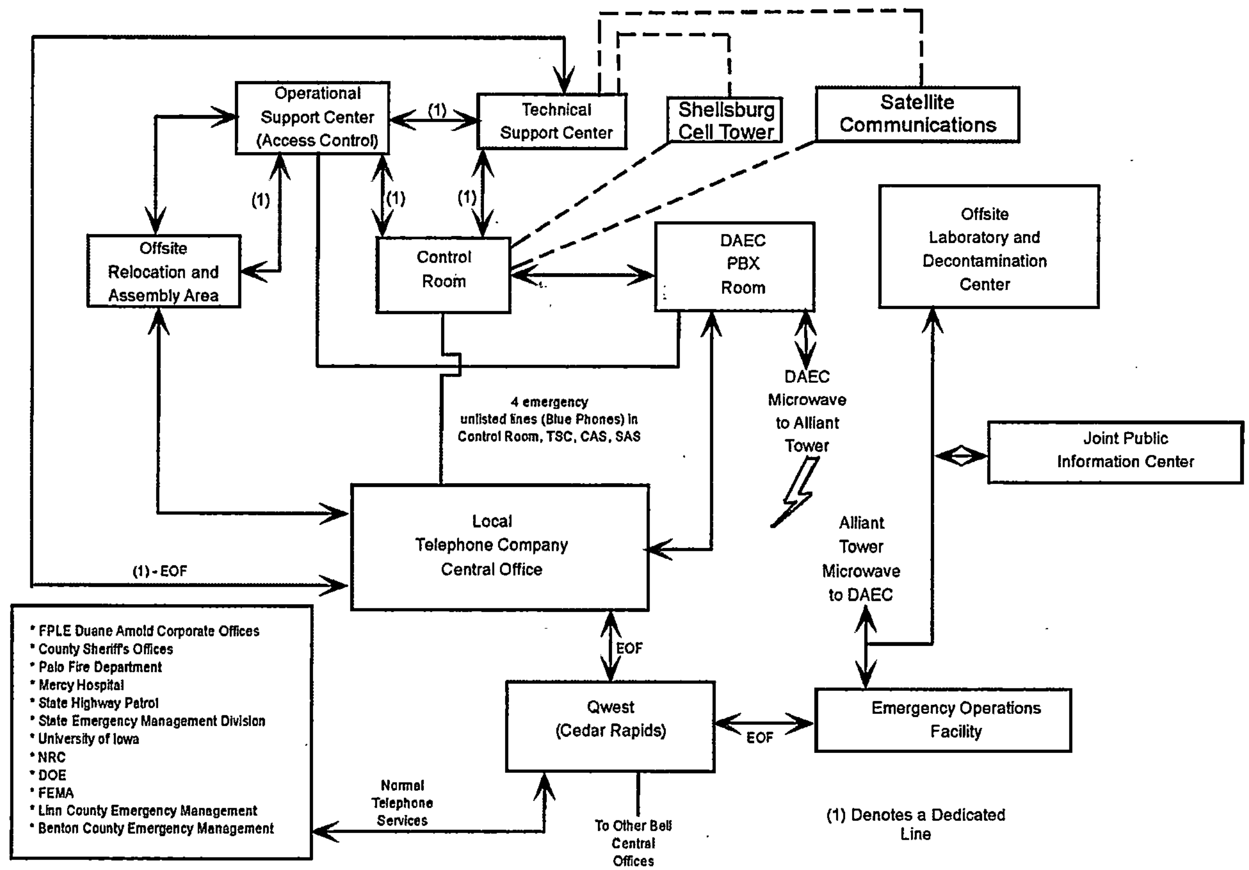


FIGURE F-6
FEDERAL TELEPHONE SYSTEM (FTS-2001)

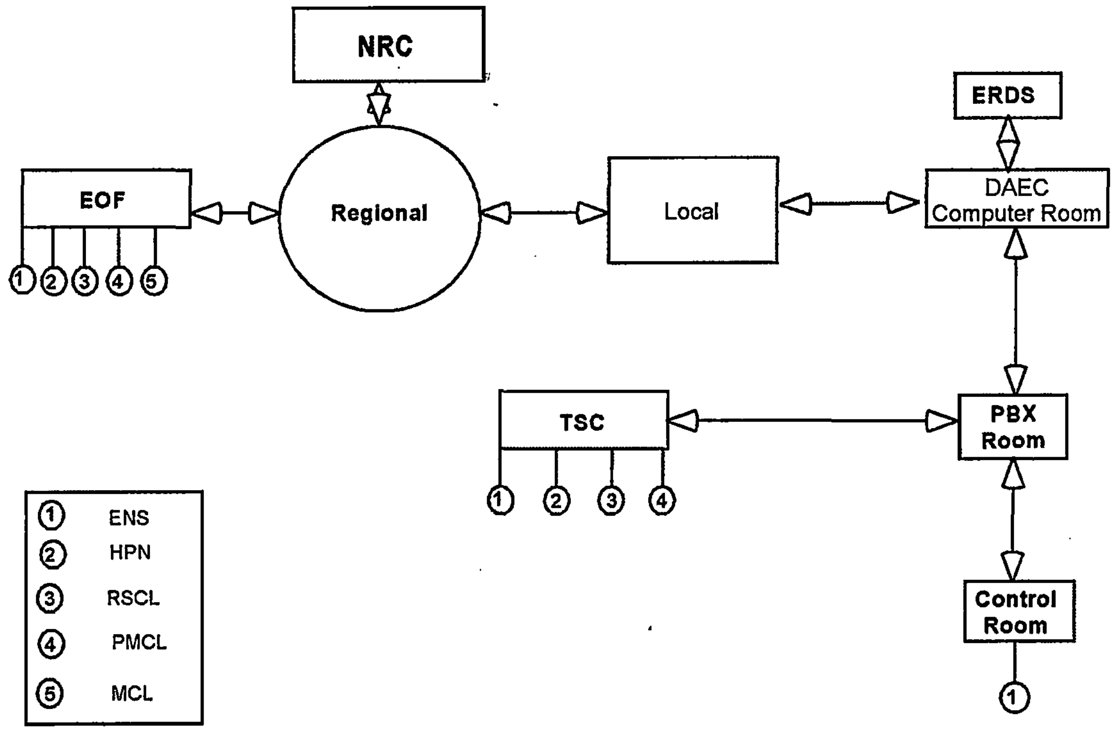
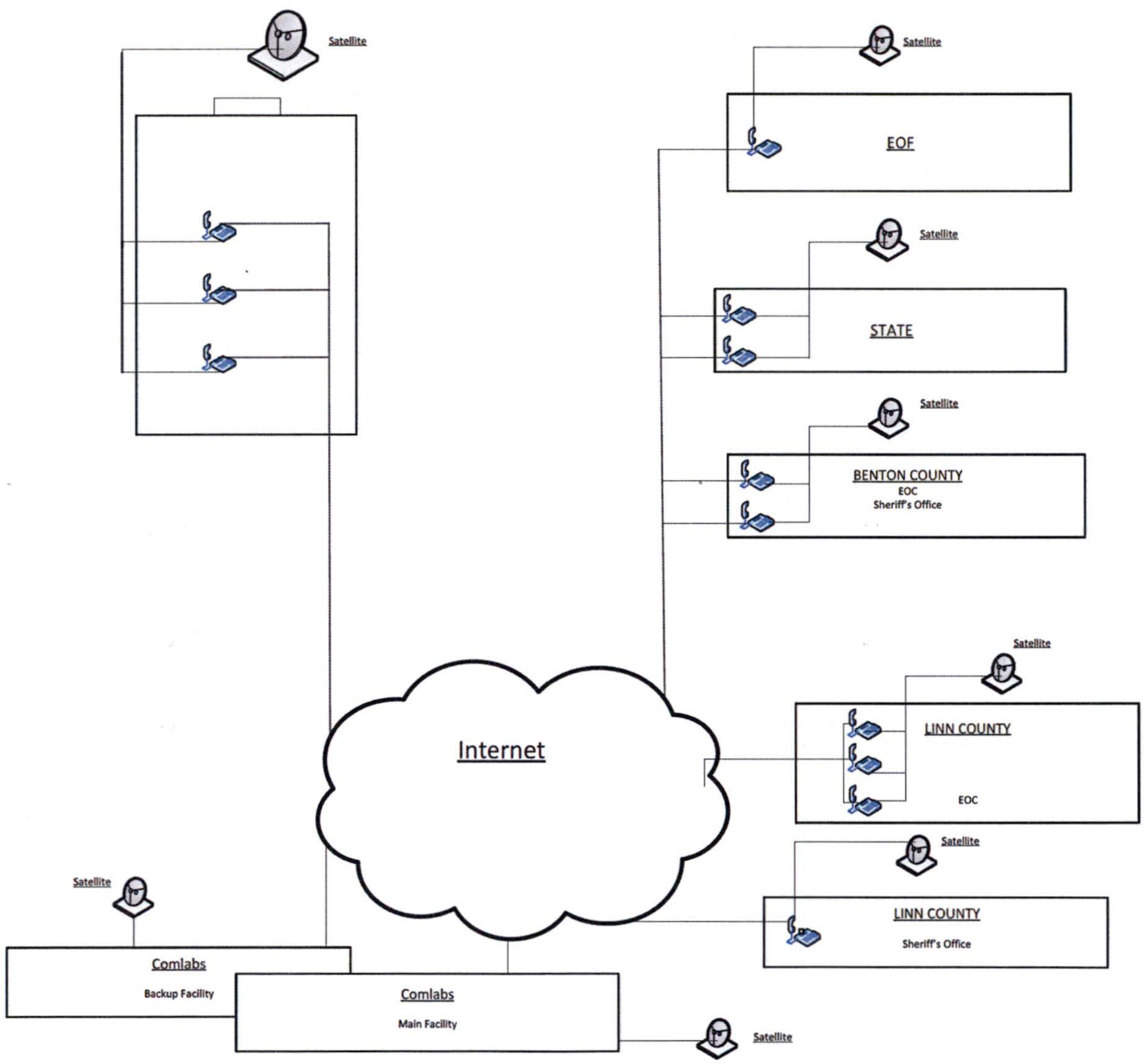


FIGURE F-7
ALL-CALL TELEPHONE SYSTEM



1.0 PURPOSE

(1) This section describes the methods and procedures used by FPLE Duane Arnold to transmit emergency information to the Emergency Response Organization, local and state authorities, and subsequently, from such authorities to the public. Details required in the initial and follow-up message are described, along with a description of the types of news statements that will be used to provide the public with information and protective actions.

2.0 REQUIREMENTS

(1) Methods used to accomplish notification of the Emergency Response Organization include the use of call lists contained in the Emergency Telephone Book, pager and automated telephone callout process.

(2) The Emergency Telephone Book includes phone numbers and pager numbers (where applicable) of emergency response personnel who may be required to respond to an emergency condition. It also includes the 24-hour telephone numbers of local, state, and federal support agencies including the NRC. The NRC would normally be notified using the NRC ENS Telephone (FTS-2001 System) from the Control Room. The state and counties would normally be notified by dedicated microwave telecommunications link.

2.1 INITIAL NOTIFICATION

(1) After declaration of an emergency condition, the Operations Shift Manager/ Supervisor will ensure that the following personnel and agencies are notified:

- Linn and Benton Counties
- State of Iowa
- NRC Operations Center
- Emergency Coordinator
- Emergency Response and Recovery Director
- NRC Resident Inspectors

(2) Verification of Notification

(a) The authenticity of initial notifications provided to Linn and Benton Counties and the State of Iowa do not require verification if the notification is made by the dedicated phone system.

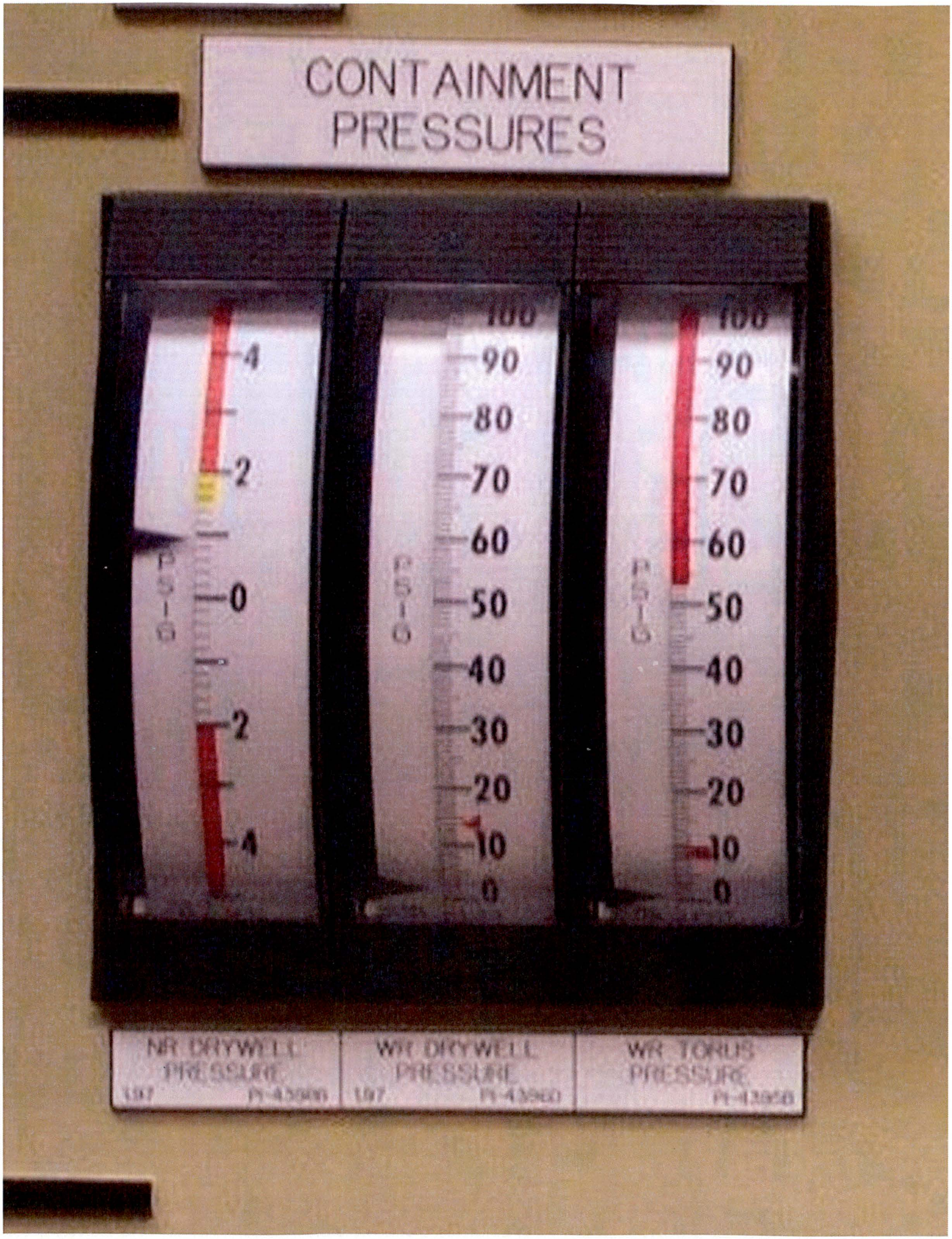
(b) Local and state agencies notified by commercial communication system (telephone or facsimile) may require verification of the identity and authenticity of the caller and the message received.

BREAKPOINTS FOR REACTOR LEVEL CONTROL
Page 1 of 2

RPV Level (inches)	Item of Interest	Significance
+211	High Level Trip Setpoint, Main Turbine Trip	• Loss of high pressure injection (FW, HPCI, RCIC) • Loss of 100% Heat Sink
+170	Low Water Level Scram, PCIS Groups 2, 3, 4 Isolations	• RPS defeats needed in ATWS • Containment Isolation, • Shutdown Cooling Valves Close
+119.5	High Pressure Injection, PCIS Group 5 Isolation, ARI	• HPCI/RCIC Auto Initiation • RWCU Isolation • ARI Initiation & Recirc Pump ATWS Trip
+87	Two Feet Below Feedwater Sparger	During ATWS if power >5% or unknown, lower level to +87 inches to reduce core inlet subcooling
+64	ECCS Auto Start, PCIS Group 1 Isolation	• ADS Timers start • CS/RHR Auto Initiation MSIVs close and result in loss of main condenser
+15	Top of Active Fuel (TAF) (Note 1)	• Loss of Adequate Core Cooling (ACC) through core submergence • If no preferred Injection Subsystem is available, maximize injection with Alternate Injection Systems in EOP 1 when level < +15"

Note 1: +15 inches is used for TAF than 0 inches for the following reasons:

- To allow monitoring RPV level on the Wide Range instrumentation - prevents risk of uncovering the core if using Fuel Zone instruments.
- Fuel Zone instruments use the same tap as jet pump instrumentation and any flow through the jet pumps including LPCI flow will cause the Fuel Zone instruments to read high.


CONTAINMENT PRESSURES
PSIG
NR DRYWELL PRESSURE
WR DRYWELL PRESSURE
WR TORUS PRESSURE

Emergency Preparedness Program Frequently Asked Question (EPFAQ)

EPFAQ Number:	2016-002
Originator:	David Young
Organization:	NEI
Relevant Guidance:	NEI 99-01, *Methodology for Development of Emergency Action Levels*, Revisions 4 and 5; and NEI 99-01, *Development of Emergency Action Levels for Non-Passive Reactors*, Revision 6. NUMARC/NESP-007, *Methodology for Development of Emergency Action Levels.*
Applicable Section(s):	Initiating Condition (IC) HA2 in NEI 99-01, Revisions 4 and 5, and NUMARC/NESP-007, "FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown" ICs CA6 and SA9 in NEI 99-01, Revision 6: "Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode" Definition of VISIBLE DAMAGE in NEI 99-01, Revisions 4, 5 and 6, and NUMARC/NESP-007
Status:	Complete

NOTE:

Based on NRC staff consideration of industry comments provided by letter dated February 16, 2017 (ADAMS Accession No. ML17079A228), a revision to these ICs was proposed at the public meeting held on April 4, 2017. These changes were attached to the public meeting notice (ADAMS Accession No. ML17089A458). Based on comments provided by the industry during the April 4, 2017 public meeting, the NRC staff revised the proposed revisions to these ICs.

QUESTION OR COMMENT:

A review of industry Operating Experience has identified a need to clarify an aspect of the definition of VISIBLE DAMAGE as it relates to the ICs cited above; adding this clarity is necessary to minimize the potential for an over-classification of an equipment failure. There may be cases where VISIBLE DAMAGE is the result of an equipment failure and limited to the failed component (i.e., the failure did not cause damage to any other component or a structure). The current definition of VISIBLE DAMAGE does not adequately differentiate between damage resulting from, and affecting only, the failed piece of equipment vs. an equipment failure causing damage to another component or a structure (e.g., by a failure-induced fire or explosion). Can the definition of VISIBLE DAMAGE be clarified to help avoid an inappropriate emergency declaration in cases where an equipment failure does not result in damage to another component or a structure (i.e., VISIBLE DAMAGE affects only the failed component)?

A related question is also posed – Consistent with the approach used in other ICs, should a note be added to preclude an emergency declaration if the safety system affected by a hazard was not functional before the event occurred (e.g., tagged out for maintenance)?

PROPOSED SOLUTION:

Yes; the sentence below may be added to the definition of VISIBLE DAMAGE [as defined in NEI 99-01, Revisions 4, 5, and 6].

Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did not cause damage to a structure or any other equipment) is not VISIBLE DAMAGE.

From a plant safety and change-in-risk perspective, the consequences from the failure of a piece of equipment, accompanied by a hazard (e.g., a fire or explosion) that does not damage any other equipment or a structure, are essentially the same as the equipment failing with no attendant hazard. Neither event would appear to meet the definition of an Alert because the outcome does not involve an actual or potential substantial degradation of the level of safety of the plant (e.g., there has been no significant reduction in the margin to a loss or potential loss of a fission product barrier). Nuclear power plants are designed with redundant safety system trains that are required to be separated (i.e., installed in separate plant areas or have separation within an individual area).

Absent any collateral damage to another component or a structure, a hazard associated with an equipment failure does not affect the ability to protect public health and safety, and there is no additional response benefit to be gained by declaring an emergency. The normal plant organization has sufficient resources and adequate guidance to respond to an equipment failure – guidance includes operating procedures and Technical Specifications; the fire protection [program], industrial safety and corrective action programs; and work management and maintenance requirements.

Concerning the second question, an emergency declaration would not be appropriate in response to a hazard affecting a piece of equipment or system that was non-functional prior to the event (e.g., tagged out for maintenance). For this reason and consistent with the approach used in other ICs, the following note may be added to IC HA2 (NEI 99-01 R4 and R5), or ICs CA6 and SA9 (NEI 99-01 R6).

Note: If the affected safety system (or component) was already non-functional before the event occurred, then no emergency classification is warranted.

Consistent with the guidance in Regulatory Issue Summary (RIS) 2003-18, Supplement 2, *Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4*, dated January 2003, it is reasonable to conclude that the changes proposed above would be considered as a "deviation."

NRC RESPONSE:

The proposed guidance is intended to ensure that an Alert should be declared only when actual or potential performance issues with SAFETY SYSTEMS have occurred as a result of a hazardous event. The occurrence of a hazardous event will result in a Notification of Unusual Event (NOUE) classification at a minimum. In order to warrant escalation to the Alert classification, the hazardous event should cause indications of degraded performance to one train of a SAFETY SYSTEM with either indications of degraded performance on the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second SAFETY SYSTEM train, such that the operability or reliability of the second train is a concern. In addition, escalation to the Alert classification should not occur if the damage from the hazardous event is limited to a SAFETY SYSTEM that was inoperable, or out of service, prior to the event occurring. As such, the proposed guidance will reduce the potential of declaring an Alert when events are in progress that do not involve an actual or potential substantial degradation of the level of safety of the plant, i.e., does not cause significant concern with shutting down or cooling down the plant.

IC HA2 (NEI 99-01 R4 and R5; NUMARC/NESP-007), or ICs CA6 and SA9 (NEI 99-01 R6), do not directly escalate to a Site Area Emergency or a General Emergency due to a hazardous event. The Fission Product Barrier and/or Abnormal Radiation Levels/Radiological Effluent recognition categories would provide an escalation path to a Site Area Emergency or a General Emergency.

The proposed addition of the following notes, applicable to ICs HA2 (NEI 99-01 R4 and R5; NUMARC/NESP-007), or ICs CA6 and SA9 (NEI 99-01 R6), provide further clarification as to how these Alert emergency classifications are considered. The revisions to these EALs, including the addition of the notes, are consistent with the current NRC-endorsed Alert classification language.

1. Adding the following note to the applicable EALs, per this EPFAQ, is acceptable as it meets the intent of the EALs, is consistent with other EALs (e.g., EAL HA5 from NEI 99-01, Revision 6; this revision was endorsed by the NRC in a letter dated March 28, 2013, available at ADAMS Accession No. ML12346A463), and ensures that declared emergencies are based upon unplanned events with the potential to pose a radiological risk to the public.

   If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.

2. Adding the following note to help explain the EAL is reasonable to succinctly capture the more detailed information from the Basis section related to when conditions would require the declaration of an Alert.

   If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

Revising the EALs and the Basis sections to ensure potential escalations from a NOUE to an Alert, due to a hazardous event, is appropriate as the concern with these EALs is: (1) a hazardous event has occurred, (2) one SAFETY SYSTEM train is having performance issues as a result of the hazardous event, and (3) either the second SAFETY SYSTEM train is having performance issues or the VISIBLE DAMAGE is enough to be concerned that the second SAFETY SYSTEM train may have operability or reliability issues.

Revising the definition for VISIBLE DAMAGE is appropriate as this definition is only used for these EALs and the revised EALs are based upon SAFETY SYSTEM trains rather than individual components or structures.

All of the changes discussed above are addressed in the attached markups to NEI 99-01, Revision 6. Licensees that use NESP-007, NEI 99-01 Revision 4, or NEI 99-01 Revision 5 EAL schemes can adopt this language in the relevant format the staff approved for their use.

Consistent with the guidance in Regulatory Issue Summary (RIS) 2003-18, Supplement 2, *Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4*, dated January 2003, a licensee's scheme change based on this EPFAQ should be considered as a "deviation" because a classification based on NRC-endorsed industry guidance in NEI 99-01, Revisions 4, 5 and 6, as well as in NUMARC/NESP-007, could be different from a classification based on this EPFAQ.

RECOMMENDED FUTURE ACTION(S):

☐ INFORMATION ONLY, MAINTAIN EPFAQ

☒ UPDATE GUIDANCE DURING NEXT REVISION

Emergency Preparedness Program Frequently Asked Question (EPFAQ)

CA6

ECL: Alert

Initiating Condition: Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode.

Operating Mode Applicability: Cold Shutdown, Refueling

Example Emergency Action Levels:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

(1) a. The occurrence of **ANY** of the following hazardous events:

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- (site-specific hazards)
- Other events with similar hazard characteristics as determined by the Shift Manager

AND

b. 1. Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:
   - Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or
   - Event damage has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria 1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance address damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Operators will make a determination of VISIBLE DAMAGE based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC AS1.

Developer Notes:

For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).

Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.

ECL Assignment Attributes: 3.1.2.B

Emergency Preparedness Program Frequently Asked Question (EPFAQ)

SA9

ECL: Alert

Initiating Condition: Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode.

Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown

Example Emergency Action Levels:

Notes:

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

(1) a. The occurrence of **ANY** of the following hazardous events:

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- (site-specific hazards)
- Other events with similar hazard characteristics as determined by the Shift Manager

AND

b. 1. Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:
   - Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode, or
   - Event damage has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for criteria 1.b.1 of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance address damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Operators will make a determination of VISIBLE DAMAGE based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via ICs FS1 or AS1.

Developer Notes:

For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).

Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.

ECL Assignment Attributes: 3.1.2.B

Emergency Preparedness Program Frequently Asked Question (EPFAQ)

VISIBLE DAMAGE: Damage to a SAFETY SYSTEM train that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected SAFETY SYSTEM train.

EPFAQ Number:	2018-04
Originator:	David Young
Organization:	NEI
Relevant Guidance:	This question concerns NEI 99-01, *Development of Emergency Action Levels for Non-Passive Reactors*, Revision 6 and EPFAQ 2016-002.
Applicable Section(s):	Initiating Conditions (ICs) CA6 and SA9, and the associated Emergency Action Levels (EALs) and Bases
Date Accepted for Review:	5/31/2018
Status:	Under Review

QUESTION OR COMMENT:

Background

EPFAQ 2016-002 provided guidance intended to reduce the potential of declaring an Alert when events are in progress that do not involve an actual or potential substantial degradation of the level of safety of the plant, i.e., does not cause significant concern with shutting down or cooling down the plant. In responding to the EPFAQ, the staff determined that revising the EALs and the Basis sections of ICs CA6 and SA9 would be appropriate to ensure potential escalations from a NOUE to an Alert, due to a hazardous event, occur when there is: (1) a hazardous event, and (2) one SAFETY SYSTEM train having performance issues as a result of the hazardous event, and (3) either the second SAFETY SYSTEM train is having performance issues or VISIBLE DAMAGE sufficient to be concerned that the second SAFETY SYSTEM train may have operability or reliability issues. The response to EPFAQ 2016-002 works well for situations involving a safety system with two trains (a typical configuration); however, industry operating experience indicates that additional clarification is needed for three other cases as described in the questions below.

Because this EPFAQ is based on material in EPFAQ 2016-002, the response to this EPFAQ may be considered only by sites that have implemented EPFAQ 2016-002 in a manner approved through an NRC Safety Evaluation Report (SER).

Question

Concerning ICs CA6 and SA9, how should an event leading to indications of degraded performance and/or VISIBLE DAMAGE be classified when:

1. The event affects equipment common to two or more safety systems or safety system trains? For example, a unit with a tank that is the water source for multiple safety injection systems or trains, such as a Refueling Water Storage Tank (RWST).

2. The event affects a safety system that has only one train. For example, a Boiling Water Reactor (BWR) unit with a single-train Reactor Core Isolation Cooling (RCIC) or High-Pressure Coolant Injection (HPCI) system.

3. The event affects two trains of a safety system having more than two trains. For example, a unit that has an Auxiliary/Emergency Feedwater system with three trains.

PROPOSED SOLUTION:

The following answers to the above questions are proposed:

1. An event affecting equipment common to two or more safety systems or safety system trains (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the common equipment) should be classified as an Alert under CA6 or SA9, as appropriate to the plant mode. By affecting the operability or reliability of multiple system trains, the loss of the common equipment effectively meets the two-train impact criteria that underlie the EALs and Bases.

2. An event affecting a single-train safety system (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under CA6 or SA9 because the two-train impact criteria that underlie the EALs and Bases would not be met. If an event affects a single-train safety system, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on Shift Manager/Emergency Director judgement.

3. An event that affects two trains of a safety system (e.g., one train has indications of degraded performance and the other VISIBLE DAMAGE) that also has one or more additional trains should be classified as an Alert under CA6 or SA9, as appropriate to the plant mode. This approach maintains consistency with the two-train impact criteria that underlie the EALs and Bases, and is warranted because the event was severe enough to affect the operability or reliability of two trains of a safety system despite plant design criteria associated with system and system train separation and protection. Such an event may have caused other plant impacts that are not immediately apparent.

As stated above, this EPFAQ may be considered only by sites that have implemented EPFAQ 2016-002 in a manner approved through an NRC Safety Evaluation Report (SER). With this proviso met, the response to EPFAQ 2018-004 would then provide clarification of expected emergency classifications for cases not explicitly addressed by ICs CA6 and SA9 (from NEI 99- 01, Revision 6), and EPFAQ 2016-002; therefore, implementation of the guidance in this EPFAQ would improve the accuracy and timeliness of a classification following a hazardous event affecting a safety system. Moreover, the answers provided in EPFAQ 2018-004 would result in EAL interpretations that are consistent with the meaning and intent of NRC-approved EAL bases such that the classification of the event would not be different from that approved by the NRC in a site-specific application. For this reason, it is reasonable to conclude that incorporation of the guidance from this EPFAQ into an NRC-approved site-specific scheme reflecting the guidance in EPFAQ 2016-002 would be considered a "difference" in accordance with Regulatory Issue Summary (RIS) 2003-18, Supplement 2, *Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4*, dated January 2003. This "difference" determination is contingent upon incorporating any or all of the three answer statements (as applicable to a facility) verbatim; any change to the scope or intent of the answers would make incorporation into a site-specific scheme a "deviation" per RIS 2003- 018, Supplement 2.

NRC RESPONSE:

RECOMMENDED FUTURE ACTION(S):

☐ INFORMATION ONLY, MAINTAIN EPFAQ

☒ UPDATE GUIDANCE DURING NEXT REVISION

BREAKPOINTS FOR REACTOR LEVEL CONTROL
Page 1 of 2

RPV Level (inches)	Item of Interest	Significance
+211	High Level Trip Setpoint, Main Turbine Trip	• Loss of high pressure injection (FW, HPCI, RCIC) • Loss of 100% Heat Sink
+170	Low Water Level Scram, PCIS Groups 2, 3, 4 Isolations	• RPS defeats needed in ATWS • Containment Isolation, • Shutdown Cooling Valves Close
+119.5	High Pressure Injection, PCIS Group 5 Isolation, ARI	• HPCI/RCIC Auto Initiation • RWCU Isolation • ARI Initiation & Recirc Pump ATWS Trip
+87	Two Feet Below Feedwater Sparger	During ATWS if power >5% or unknown, lower level to +87 inches to reduce core inlet subcooling
+64	ECCS Auto Start, PCIS Group 1 Isolation	• ADS Timers start • CS/RHR Auto Initiation MSIVs close and result in loss of main condenser
+15	Top of Active Fuel (TAF) (Note 1)	• Loss of Adequate Core Cooling (ACC) through core submergence • If no preferred Injection Subsystem is available, maximize injection with Alternate Injection Systems in EOP 1 when level < +15"

Note 1: +15 inches is used for TAF than 0 inches for the following reasons:

- To allow monitoring RPV level on the Wide Range instrumentation - prevents risk of uncovering the core if using Fuel Zone instruments.
- Fuel Zone instruments use the same tap as jet pump instrumentation and any flow through the jet pumps including LPCI flow will cause the Fuel Zone instruments to read high.

SAG-3 HYDROGEN CONTROL

CAUTIONS

3 Operation of HPCI, RCIC, Core Spray, or RHR with suction from the torus and pump flow above the NPSH or vortex limit may damage equipment.

7 H2 and O2 instruments may indicate higher concentrations than actually exist inside the containment following a large break LOCA due to moisture condensation in the sample lines. During the 24 hr period following a large break LOCA, the monitors should not be independently used to support operational decisions but may be used for trending.

START

While in this procedure:

IF	THEN
H_2 or O_2 monitor is unavailable	Notify Chemistry to manually sample the drywell and torus for H_2 and O_2 (PASAP 2.6).
Drywell pressure drops below 2.0 psig	Verify containment sprays isolate.
Torus pressure drops below 2.0 psig	Terminate torus sprays.

H-1

DRYWELL

Drywell O_2 (Caution 7)

Drywell H_2	Drywell O_2 < 5%	Drywell O_2 ≥ 5% or unknown, Torus H_2 None	Drywell O_2 ≥ 5% or unknown, Torus H_2 < 6%	Drywell O_2 ≥ 5% or unknown, Torus H_2 ≥ 6% or unknown
None	No action	No action	②	③
< 6%	①	②	②	③
≥ 6% or unknown	①	③	③	③

H-2

1

IF........offsite release rate is expected to stay below normal limits (Detail D),
THEN...vent and purge the primary containment:

- OK to defeat isolations except high radiation (Defeat 9).
- If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. Vent as follows:
   - IF........torus water level is below 16 ft, THEN...vent the drywell through the torus (SEP 301.1).
   - IF........torus water level is at or above 16 ft, OR......the torus cannot be vented, THEN...vent directly from the drywell (SEP 301.2).
2. IF........the primary containment can be vented, THEN...purge the drywell with nitrogen using N_2 purge (SEP 303.2).
3. Stop the vent and purge (if not required by other SAG steps) when:
   - Hydrogen is no longer detected in the drywell,
   OR
   - Offsite release rate reaches normal limits (Detail D).

H-3

2

IF........offsite release rate is expected to stay below General Emergency Levels (EAL RG1),
OR......RPV water level cannot be maintained above +15 in. (TAF),
THEN...vent and purge the primary containment:

- OK to defeat all isolations (Defeat 10).
- If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. Vent as follows:
   - IF........torus water level is below 16 ft, THEN...vent the drywell through the torus (SEP 301.1).
   - IF........torus water level is at or above 16 ft, OR......the torus cannot be vented, THEN...vent directly from the drywell (SEP 301.2).
2. IF........the primary containment can be vented, THEN...purge the drywell with nitrogen at max flow using N_2 purge (SEP 303.2).
3. Stop the vent and purge (if not required by other SAG steps) when:
   - Hydrogen is no longer detected in either the drywell or the torus,
   OR
   - Hydrogen is no longer detected in the drywell and drywell oxygen is less than 5%,
   OR
   - RPV water level can be maintained above +15 in. (TAF) and offsite release rate reaches a General Emergency Level (EAL RG1).

H-4

3

Vent and purge the primary containment:

- OK to defeat all isolations (Defeat 10).
- OK to exceed release rate limits.
- If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. Vent as follows:
   - IF........torus water level is below 16 ft, THEN...vent the drywell through the torus (SEP 301.1).
   - IF........torus water level is at or above 16 ft, OR......the torus cannot be vented, THEN...vent directly from the drywell (SEP 301.2).
2. IF........the primary containment can be vented, THEN...purge the drywell with air or nitrogen at max flow. Use whichever method will reduce hydrogen below 6% or oxygen below 5% faster:
   - Air purge (SEP 303.1)
   - N_2 purge (SEP 303.2)
3. IF........permitted by SAG-1, THEN...operate drywell sprays (Detail E).

H-5

TORUS

Torus O_2 (Caution 7)

Torus H_2	Torus O_2 < 5%	Torus O_2 ≥ 5% or unknown, Drywell H_2 None	Torus O_2 ≥ 5% or unknown, Drywell H_2 < 6%	Torus O_2 ≥ 5% or unknown, Drywell H_2 ≥ 6% or unknown
None	No action	No action	⑤	⑥
< 6%	④	⑤	⑤	⑥
≥ 6% or unknown	④	⑥	⑥	⑥

H-6

4

IF........offsite release rate is expected to stay below normal limits (Detail D),
THEN...vent and purge the primary containment:

- OK to defeat isolations except high radiation (Defeat 9).
- If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. Vent as follows:
   - IF........torus water level is below 16 ft, THEN...vent directly from the torus (SEP 301.1).
   - IF........the torus cannot be vented, AND....torus water level is below 13.5 ft, THEN...vent the torus through the drywell (SEP 301.2).
2. IF........the primary containment can be vented, THEN...purge the torus with nitrogen using N_2 purge (SEP 303.2).
3. Stop the vent and purge (if not required by other SAG steps) when:
   - Hydrogen is no longer detected in the torus,
   OR
   - Offsite release rate reaches normal limits (Detail D).

H-7

5

IF........offsite release rate is expected to stay below General Emergency Levels (EAL RG1),
OR......RPV water level cannot be maintained above +15 in. (TAF),
THEN...vent and purge the primary containment:

- OK to defeat all isolations (Defeat 10).
- If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. Vent as follows:
   - IF........torus water level is below 16 ft, THEN...vent directly from the torus (SEP 301.1).
   - IF........the torus cannot be vented, AND....torus water level is below 13.5 ft, THEN...vent the torus through the drywell (SEP 301.2).
2. IF........the primary containment can be vented, THEN...purge the torus with nitrogen at max flow using N_2 purge (SEP 303.2).
3. Stop the vent and purge (if not required by other SAG steps) when:
   - Hydrogen is no longer detected in either the drywell or the torus,
   OR
   - Hydrogen is no longer detected in the torus and torus oxygen is less than 5%,
   OR
   - RPV water level can be maintained above +15 in. (TAF) and offsite release rate reaches a General Emergency Level (EAL RG1).

H-8

6

Vent and purge the primary containment:

- OK to defeat all isolations (Defeat 10).
- OK to exceed release rate limits.
- If pneumatic supplies are unavailable, use SAMP 706, Venting the Primary Containment Following Loss of Pneumatic Supply.

1. IF........permitted by SAG-1, THEN...operate torus sprays (Detail E).
2. Vent as follows:
   - IF........torus water level is below 16 ft, THEN...vent directly from the torus (SEP 301.1).
   - IF........the torus cannot be vented, AND....torus water level is below 13.5 ft, THEN...vent the torus through the drywell (SEP 301.2).
2. IF........the primary containment can be vented, THEN...purge the torus with air or nitrogen at max flow. Use whichever method will reduce hydrogen below 6% or oxygen below 5% faster:
   - Air purge (SEP 303.1)
   - N_2 purge (SEP 303.2)

H-9

Detail D — Normal Release Rate Limits

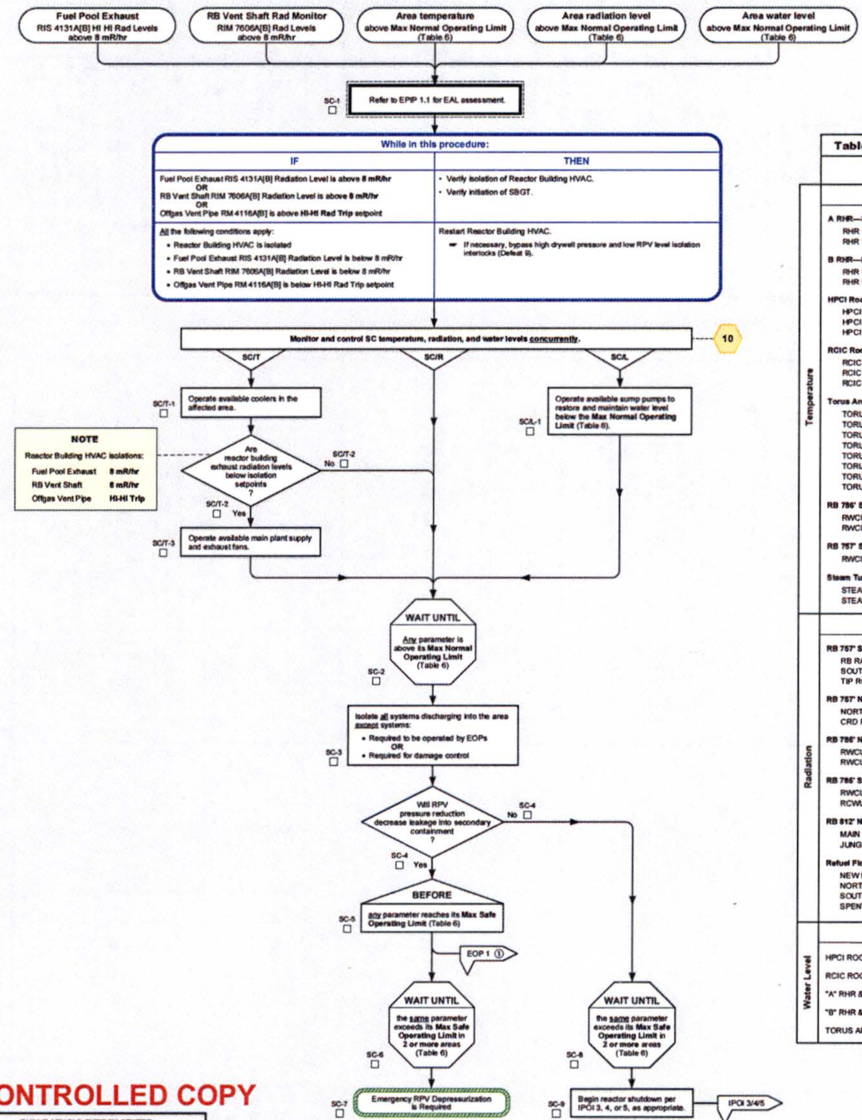
A determination that the offsite release rate is below normal limits may be made by either:

- Containment Atmosphere Radiation Monitor GASEOUS Channels on-scale and operable.
  - Monitored locally on RIT-8102A/B at 1C-219A/B or RR-4379A/B at 1C-29 (blue channel).
- Containment sample (PASAP 7.4).

Detail E — Spray Sources (Caution 3)

- RHR (OI 149)
- RHR Service Water (AIP 401)
- Fire System (AIP 404)
- Well Water (AIP 403)
- GSW (AIP 403)
- ESW (AIP 402)
- Condensate Service Water (AIP 405)

EOP 3
SECONDARY CONTAINMENT CONTROL

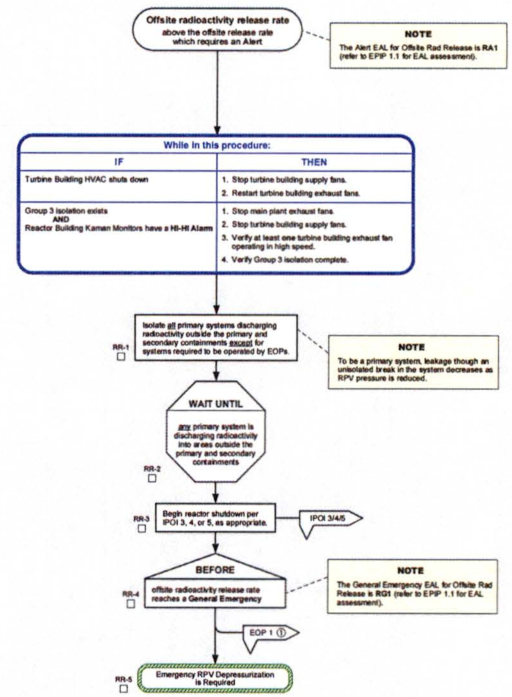


CAUTIONS

10 — High spent fuel pool temperature or low spent fuel pool water level may affect conditions within secondary containment.

Table 6	Secondary Containment Limits				
	Parameter and Areas		Max Normal Operating Limit	Max Safe Operating Limit	Value/Trend
Temperature	Area/Location	Indicator	°F	°F	
	A RHR—SE Corner Room Area				
	RHR SE CORNER ROOM AMBIENT	TR/TDR 2000A Ch 1	130	140	
	RHR SE CORNER ROOM DIFFERENTIAL	TR/TDR 2000A Ch 2	50	N/A	
	B RHR—NW Corner Room Area				
	RHR NW CORNER ROOM AMBIENT	TR/TDR 2000B Ch 1	130	140	
	RHR NW CORNER ROOM DIFFERENTIAL	TR/TDR 2000B Ch 2	50	N/A	
	HPCI Room Area				
	HPCI EMER COOLER AMBIENT	TR/TDR 2225A[B] Ch 1	175	310	
	HPCI ROOM AMBIENT	TR/TDR 2225A Ch 2	175	310	
	HPCI ROOM DIFFERENTIAL	TR/TDR 2225A[B] Ch 4[3]	50	N/A	
	RCIC Room Area				
	RCIC EMER COOLER AMBIENT	TR/TDR 2425A[B] Ch 1	175	300	
	RCIC ROOM AMBIENT	TR/TDR 2425A Ch 2	175	300	
	RCIC ROOM DIFFERENTIAL	TR/TDR 2425A[B] Ch 4	50	N/A	
	Torus Area				
	TORUS CATWALK NORTH AMBIENT	TR/TDR 2425A Ch 3	150	165	
	TORUS CATWALK WEST AMBIENT	TR/TDR 2425B Ch 2	150	165	
	TORUS CATWALK SOUTH AMBIENT	TR/TDR 2225A Ch 3	150	165	
	TORUS CATWALK EAST AMBIENT	TR/TDR 2225B Ch 2	150	165	
	TORUS CATWALK EAST DIFF	TR/TDR 2425A Ch 5	50	N/A	
	TORUS CATWALK WEST DIFF	TR/TDR 2425B Ch 5	50	N/A	
	TORUS CATWALK SOUTHWEST DIFF	TR/TDR 2225A Ch 5	50	N/A	
	TORUS CATWALK SOUTH DIFF	TR/TDR 2225B Ch 4	50	N/A	
	RB 786' South Area				
	RWCU PUMP ROOM AMBIENT	TR/TDR 2700A[B] Ch 1	130	212	
	RWCU HX ROOM AMBIENT	TR/TDR 2700A[B] Ch 2,3	130	212	
	RB 757' South Area				
	RWCU ABOVE TIP ROOM AMBIENT	TR/TDR 2700A[B] Ch 4,5	111.5	150	
	Steam Tunnel Area				
	STEAM TUNNEL AMBIENT	TR/TDR 2425B Ch 3	160	300	
	STEAM TUNNEL DIFFERENTIAL	TR/TDR 2225B Ch 5	70	N/A	
Radiation	Area/Location	Indicator	mR/hr	mR/hr	
	RB 757' South Area				
	RB RAILROAD ACCESS AREA	RI 9167	10	100	
	SOUTH CRD MODULE AREA	RI 9169	10	100	
	TIP ROOM	RI 9176	60	600	
	RB 757' North Area				
	NORTH CRD MODULE	RI 9168	10	100	
	CRD REPAIR ROOM	RI 9170	15	150	
	RB 786' North Area				
	RWCU SPENT RESIN ROOM	RI 9173	100	10^3	
	RWCU PHASE SEP TANK ROOM	RI 9177	20	200	
	RB 786' South Area				
	RWCU PUMP ROOM	RI 9156	10^3	10^4	
	RCWU HX ROOM	RI 9157	10^3	10^4	
	RB 812' North Area				
	MAIN PLANT EXHAUST FAN ROOM	RI 9171	60	600	
	JUNGLE ROOM	RI 9155	60	600	
	Refuel Floor Area				
	NEW FUEL VAULT AREA	RI 9153	10	100	
	NORTH REFUEL FLOOR	RI 9163	10	100	
	SOUTH REFUEL FLOOR	RI 9164	10	100	
	SPENT FUEL POOL AREA	RI 9178	100	10^3	
Water Level	Area/Location	Indicator	inches	inches	
	HPCI ROOM	LI 3768	2	6	
	RCIC ROOM	LI 3769	3	6	
	"A" RHR & CS SECR	LI 3770	2	10	
	"B" RHR & CS NWCR	LI 3771	2	10	
	TORUS AREA	LI 3772	2	12	

EOP 4
RADIOACTIVITY RELEASE CONTROL



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DUANE ARNOLD ENERGY CENTER	
EOP 3 – SECONDARY CONTAINMENT CONTROL EOP 4 – RADIOACTIVITY RELEASE CONTROL	REV. 21

1.2.7 HSM Dose Rates with a Loaded 24P, 52B or 61BT DSC

Limit/Specification:

Dose rates at the following locations shall be limited to levels which are less than or equal to:

a. 400 mrem/hr at 3 feet from the HSM surface.

b. Outside of HSM door on center line of DSC 100 mrem/hr.

c. End shield wall exterior 20 mrem/hr.

Applicability: This specification is applicable to all HSMs which contain a loaded 24P, 52B or 61BT DSC.

Objective: The dose rate is limited to this value to ensure that the cask (DSC) has not been inadvertently loaded with fuel not meeting the specifications in Section 1.2.1 and to maintain dose rates as-low-as-is-reasonably achievable (ALARA) at locations on the HSMs where surveillance is performed, and to reduce off-site exposures during storage.

Action:

a. If specified dose rates are exceeded, the following actions should be taken:

61BT DSC Dose Rate Thresholds = 2 X TS limits

Therefore:

3 feet from HSM Surface = 800 mrem/hr

Outside HSM Door – Centerline of DSC = 200 mrem/hr

End Shield Wall Exterior = 40 mrem/hr

1. Ensure that the DSC is properly positioned on the support rails.

2. Ensure proper installation of the HSM door.

3. Ensure that the required module spacing is maintained.

4. Confirm that the spent fuel assemblies contained in the DSC conform to the specifications of Section 1.2.1.

5. Install temporary or permanent shielding to mitigate the dose to acceptable levels in accordance with 10 CFR Part 20, 10 CFR 72.104(a), and ALARA.

b. Submit a letter report to the NRC within 30 days summarizing the action taken and the results of the surveillance, investigation and findings. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to the administrator of the appropriate NRC regional office.

Surveillance: The HSM and ISFSI shall be checked to verify that this specification has been met after the DSC is placed into storage and the HSM door is closed.

Basis: The basis for this limit is the shielding analysis presented in Section 7.0, Appendix J, and Appendix K of the FSAR. The specified dose rates provide as-low-as-is-reasonably-achievable on-site and off-site doses in accordance with 10 CFR Part 20 and 10 CFR 72.104(a).

Development of Fission Product Barrier EAL Threshold Values from NEE-323-CALC-001

NOTE: Fuel Clad barrier LOSS 4.A(B) threshold values below are scaled from the 100% gap release instead of calculated based on 300uci/gm DEI as assumed in NEI 99-01 Revision 6 developer guidance. This variation from the NRC endorsed guidance is due to the calculated value not reflecting the intended 2-5% gap release threshold due to differences in plant design. The calculation will be formally revised to reflect this change in methodology.

	Drywell dose rate R/hr	Torus dose rate R/hr	Drywell dose rate R/hr	Torus dose rate R/hr
1691 MWth 100% Gap release	22700	2140	Values below are rounded for ease of use, as well as to provide a step-wise progression	Values below are rounded for ease of use, as well as to provide a step-wise progression
Scaling factor to account for power uprate to 1912 MWth	1.13	1.13		
Updated 100% Gap release	25667	2420		
After application of 0.2 scaling factor for 20% Gap release *CTMNT barrier LOSS 4.A(B)*	5133	484	**5000**	**500**
After application of 0.05 scaling factor for 5% Gap release *Fuel Clad barrier LOSS 4.A(B)*	1283	121	**1250**	**125**

ENERCON *Excellence—Every project. Every day.*	CALCULATION COVER SHEET	**CALC NO.** NEE-323-CALC-001
		REV. 00
		PAGE NO. 1 of 10

Title:	Primary Containment Radiation EAL Threshold Determination	**Client:** Duane Arnold Energy Center
		Project Identifier: NEE-323

Item	**Cover Sheet Items**	**Yes**	**No**
1	Does this calculation contain any open assumptions, including preliminary information, that require confirmation? (If **YES**, identify the assumptions.)	☐	☒
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, identify the design verified calculation.) **Design Verified Calculation No.** ______	☐	☒
3	Does this calculation supersede an existing Calculation? (If **YES**, identify the design verified calculation.) **Superseded Calculation No.** ______	☐	☒

Scope of Revision:
Initial Issue

Revision Impact on Results:
Initial Issue

Study Calculation ☐ **Final Calculation** ☒

Safety-Related ☐ **Non-Safety-Related** ☒

(Print Name and Sign)

Originator: Aaron Holloway	**Date:** **12/12/17**
Design Verifier[1] (Reviewer if NSR): Jay Bhatt	**Date:** **12/12/17**
Approver: Zachary Rose	**Date:** **12/12/17**

Note 1: For non-safety-related calculation, design verification can be substituted by review.

ENERCON	CALCULATION REVISION STATUS SHEET	CALC NO. NEE-323-CALC-001
		REV. 00

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
00	12/12/17	Initial Issue

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
All	00		

APPENDIX/ATTACHMENT REVISION STATUS

APPENDIX NO.	NO. OF PAGES	REVISION NO.	ATTACHMENT NO.	NO. OF PAGES	REVISION NO.
A	1	00	1	4	00
B	1	00			

1.0 Purpose and Scope

The purpose of this calculation is to determine the site-specific threshold for primary containment radiation in the event of a loss or potential loss of the three fission product barriers (fuel clad, Reactor Coolant System, containment). These site-specific values can be used to determine the Emergency Action Level (EAL) (FA1, FS1, or FG1) in accordance with Table 9-F-2 of NEI 99-01, Rev. 6. This calculation is nonsafety-related as it intended for emergency classification and not design basis purposes. There are no acceptance criteria associated with this calculation since the purpose is only to determine site-specific radiation thresholds.

2.0 Summary of Results and Conclusions

The calculated primary containment radiation readings for each of the three fission product barriers are listed below. Note that the results presented below are calculated dose rates and do not account for background radiation or any installed detector check sources.

Table 1 - Calculated Containment Atmospheric Monitoring System (CAMS) radiation readings for a release into the drywell

Failure	Drywell Monitor (9184A/B) Reading (R/hr)	
Reactor Coolant System (Loss)	1.33	5 R/hr chosen as minimum readable
Fuel Clad (Loss)	2000	used as 2000
Containment (Potential Loss)	5130	rounded to 5000

Table 2 - Calculated CAMS radiation readings for a release into the torus

Failure	Torus Monitor (9185A/B) Reading (R/hr)	
Reactor Coolant System (Loss)	0.125 (not on scale)	
Fuel Clad (Loss)	188	rounded to 200
Containment (Potential Loss)	484	rounded to 500

3.0 References

3.1 NG-88-0966, "Nuclear Generation Division Office Memo, G.E. Fuel Damage Documentation / Dose Rate Calculations", dated 03/18/88

3.2 IPOI 8, "Outage and Refueling Operations", Rev. 91

3.3 Bech-M115, "Reactor Vessel Instrumentation P&ID", Rev. 62

3.4 Duane Arnold Energy Center Facility Operating License Appendix A - Technical Specifications, as revised through Amendment No. 297

3.5 NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors", Rev. 6

3.6 Shultis, J.K., "Fundamentals of Nuclear Science and Engineering", 2002

3.7 Lindeburg, M.R., "Mechanical Engineering Reference Manual for the PE Exam", Twelfth Edition, 2006

3.8 Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", 1989

3.9 I.RIM-V115-01, "Victoreen Model 876A Containment Radiation Monitor Calibration", Rev. 10

4.0 Assumptions

4.1 Reactor Pressure Vessel (RPV) water level is 535 inches above vessel zero for the purposes of calculating the total Reactor Coolant System (RCS) water volume. This corresponds to the middle of the band between the high and low RPV water level alarm points from Table 4 (Design Input 5.7) and represents the most realistic water inventory during normal operation.

4.2 The fission product isotopic distribution in the reactor coolant will be similar to that of the fission product gap inventory. This is reasonable since, in the event of a fuel cladding failure, the isotopes of concern (iodines) would be released to the reactor coolant at the same time and distribution.

4.3 All reactor coolant mass is assumed to be released into the primary containment. This is consistent with the NEI 99-01 Rev. 6 developer notes.

5.0 Design Inputs

5.1 Duane Arnold's license thermal power limit is 1912 MW_{th}, taken from Reference 3.4.

5.2 The specific volume of saturated liquid water at 1000 psia is 0.02160 ft^3/lb_m per Appendix 23.B of Reference 3.7.

5.3 The following unit conversions are used within this calculation:

Table 3 - Applicable Unit Conversions

Base Unit	Equivalent	Reference
1 Sievert (Sv)	1.0E5 mrem	3.6
1 Curie (Ci)	3.7E10 Bq	3.6
1 Gallon	3785.4 cubic centimeters (cc)	3.7
1 lb_m/ft^3	0.016018 g/cc	3.7

5.4 The Technical Specifications limit for RCS activity is 0.2 μCi/gm Dose Equivalent I-131 (DEI) per LCO 3.4.6 of Reference 3.4.

5.5 The RCS volume at the centerline of the Main Steam lines is 72,000 gallons per Reference 3.2.

5.6 The change in RCS volume per unit change in height is 100 gallons/inch per Reference 3.2.

5.7 The following elevations are taken from Reference 3.3:

Table 4 - Pertinent RPV elevations relative to Vessel 0

Point	Height above Vessel 0 (inches)
Nozzle N3A,B,C,D (centerline of Main Steam Lines)	620.5

Point	Height above Vessel 0 (inches)
High Level Alarm	539.5
Low Level Alarm	530.5

5.8 The drywell dose rate at the CAMS monitor location for 100% gap release into the drywell of a 1691 MW_{th} core at 0.01 hours decay time is 2.27E4 R/hr, per Table 3 of Reference 3.1.

5.9 The torus dose rate at the CAMS monitor location for 100% gap release into the torus of a 1691 MW_{th} core at 0.01 hours decay time is 2.14E3 R/hr, per Table 4 of Reference 3.1.

5.10 The drywell (9184 A/B) and torus (9185 A/B) radiation monitor ranges (1 to 10^7 R/hr) are taken from Reference 3.9.

5.11 The fission product gap inventories for Iodine isotopes are taken from Table 1 of Reference 3.1. These inventories correspond to a core with a rated thermal power of 1691 MW_{th}.

Table 5 - Iodine Gap Inventories for 1691 MW_{th} Core

Nuclide	1691 MW_{th} Gap Inventory (Ci)
I-130	7.25E+03
I-131	5.34E+05
I-132	8.45E+04
I-133	3.63E+05
I-134	8.19E+04
I-135	1.93E+05

5.12 Dose conversion factors for effective dose due to inhalation are taken from Reference 3.8, Table 2.1.

Table 6 - Dose Conversion Factors for Total Effective Dose from Inhalation

Nuclide	Dose Conversion Coefficient (Sv/Bq)
I-130	7.14E-10
I-131	8.89E-09
I-132	1.03E-10
I-133	1.58E-09
I-134	3.55E-11
I-135	3.32E-10

6.0 Methodology

The approach of this calculation is to scale the results of a previous calculation (NG-88-0966, Reference 3.1) based on the specific RCS activities as specified in NEI 99-01 Rev. 6. Scaling factors are determined for each of the fission product barrier failure thresholds specified in NEI 99-01 (i.e. Loss of RCS, Loss of Fuel Cladding, and Potential Loss of Primary Containment). The RCS activity concentrations are taken from the NEI 99-01 Rev. 6 developer notes and are listed below.

Table 7 - RCS Activities for fission product barrier failures

Failure	RCS Activity
Reactor Coolant System (Loss)	Technical Specification Limit
Fuel Clad (Loss)	300 μCi/g Dose Equivalent I-131
Containment (Potential Loss)	20% fuel cladding failure

These scaling factors are then applied to the CAMS radiation response from calculation NG-88-0966 to determine the site-specific values for the three thresholds. Additionally, the previous calculation determined the CAMS radiation monitor response for an assumed release of 100% gap activity from the core with a power level of 1691 MW_{th}. However, Duane Arnold's licensed power level is now 1912 MW_{th}. This does not impact the Reactor Coolant System and Fuel Clad barrier thresholds because the radiation responses are scaled based on the DEI levels. However, the difference in licensed power level will need to be accounted for in the Potential Loss of Containment threshold because this threshold is related to the total gap inventory. Scaling of the gap inventory based on power level is consistent with calculation NG-88-0966 as Table 1 of NG-88-0966 provides gap inventory per megawatt.

6.1 Determination of RCS water volume and mass

The NEI 99-01 Rev. 6 primary containment radiation thresholds are based on specific RCS radioactivity concentrations. However, the corresponding total RCS activity must be known in order to compare these thresholds to the gap release assumed in calculation NG-88-0966. Therefore, the total mass of water in the RCS must first be determined so that the total RCS activity can be calculated for each threshold.

The total RCS water volume at normal operation is determined by taking the RCS volume when filled to the centerline of the main steam lines, and then subtracting the difference in volume between the centerline of the main steam lines and the normal water level. This is presented in Equation 1 below:

$$V_{normal} = V_{MSL} - aH \qquad \text{[Equation 1]}$$

Where:

V_{normal} = The RCS water volume at normal operation (gallons)

V_{MSL} = The RCS water volume when filled to the centerline of the main steam lines (gallons)

a = The change in RCS water volume per inch change in vessel height (gallons/inch)

H = The distance between the centerline of the main steam lines and the normal RCS water level (inches)

It should be noted that calculating the RCS volume as shown above does not include the volume of the steam lines. However, the volume of the steam lines filled to the centerline of the nozzle is very small compared to the total RCS volume, and therefore does not significantly impact the results of the calculation.

The total mass of water in the RCS can then be determined based on the water density, as outlined in Equation 2 below:

$$M_{normal} = \frac{V_{normal}}{v} \qquad \text{[Equation 2]}$$

Where:

M_{normal} = The mass of water in the RCS at normal operation (grams)

v = *The specific volume of water at normal operation (grams/gallon)*

6.2 Determination of Scaling Factors

The scaling factors for the fuel clad and RCS barrier thresholds are determined by comparing the corresponding dose at each RCS activity concentration to the DEI of the fission product gap inventory. This is presented in Equation 3 below:

$$F = \frac{DCF_{I-131}A_{I-131}M_{normal}}{\sum_{i=130}^{i=135} I_i DCF_i} \qquad \text{[Equation 3]}$$

Where:

F = The scaling factor for a given RCS activity concentration threshold.

DCF_i= The dose conversion factor for isotope "i" in mrem/Ci. These values are developed from Table 6 above.

A_{I-131} = The I-131 concentration in the RCS for a given threshold in Ci/gram. These values are developed from Table 7 and Design Input 5.4 above.

M_{normal} = The mass of RCS water at normal operation in grams. This value is determined from Equation 2 above.

I_i = The gap inventory of iodine isotope "i" at a power level of 1691 MW_{th} in Ci. These values are taken from Table 5 above.

For the potential loss of containment threshold, NEI 99-01 specifies that 20% of fuel cladding has failed, rather than giving a specific RCS activity concentration. Therefore, the scaling factor for this case is simply 0.2 (20% of the 100% gap release case) multiplied by the ratio of the new to previous licensed power levels (1912/1691) to account for the increased gap inventory.

6.3 Determination of CAMS Detector Response

Once the scaling factors have been determined for each of the three RCS activity concentration thresholds, they can be applied to the results of calculation NG-88-0966 to determine the CAMS detector response for each threshold. Specifically, the CAMS detector response can be obtained from Equation 3 below:

$$D_j = F_j D_{GAP} \qquad \text{[Equation 3]}$$

Where:

D_j = The dose rate at the CAMS detector for an RCS activity concentration of "j" in R/hr

F_j = The scaling factor for an RCS activity concentration of "j", determined from Equation 2 above

D_{GAP} = The dose rate at the CAMS monitor location for 100% gap release of a 1691 MWth core in R/hr

7.0 Calculation

All calculations were completed using Microsoft Excel. The calculation results and spreadsheet formulas are presented in Appendix A and B, respectively.

8.0 Impact Assessment

This calculation is based on "realistic" assumptions for the purpose of declaring EALs, rather than typical conservative "bounding" type design basis analyses. The calculation results are intended to provide order of magnitude dose rates to assist Operations and Emergency Response personnel in determining the state of the three fission product barriers in accordance with NEI 99-01 Rev. 6.

	A	B	C	D	E
1	1 Sievert	100000	mrem		
2	1 Ci	3.70E+10	Bq		
3					
4	Isotope	DCF (Sv/Bq)	DCF (mrem/Ci)	1691 MW Gap Inventory (Ci)	Dose (mrem)
5	I-130	7.14E-10	2.64E+06	7.25E+03	1.92E+10
6	I-131	8.89E-09	3.29E+07	5.34E+05	1.76E+13
7	I-132	1.03E-10	3.81E+05	8.45E+04	3.22E+10
8	I-133	1.58E-09	5.85E+06	3.63E+05	2.12E+12
9	I-134	3.55E-11	1.31E+05	8.19E+04	1.08E+10
10	I-135	3.32E-10	1.23E+06	1.93E+05	2.37E+11
11					
12				Total	2.00E+13
13	V_{MSL}	72000.0	gal		
14	a	100.0	gal/inch		
15	Elevation of the Main Steam Lines	620.5	inches above vessel 0		
16	Elevation of the Normal Water Level	535.0	inches above vessel 0		
17	H	85.5	inches		
18	V_{Normal}	63450	gal		
19	V_{Normal}	240184264.5	cc		
20	Specific Volume @ 1000 psia	0.0216	ft^3/lbm		
21	water density @ 1000 psia	0.7416	g/cc		
22	M_{Normal}	178114424	grams		
23					
24	Drywell Dose Rate for 100% Gap Release (1691 MWth)	2.27E+04	R/hr		
25	Torus Dose Rate for 100% Gap Release (1691 MWth)	2.14E+03	R/hr		
26					
27	**Threshold**	**Scaling Factors (F)**	**Drywell Dose Rate (R/hr)**	**Torus Dose Rate (R/hr)**	
28	**0.2 µCi/gm (TS Limit)**	**5.86E-05**	**1.33E+00**	**1.25E-01**	
29	**300 µCi/gm**	**8.79E-02**	**2.00E+03**	**1.88E+02**	
30	**20% Failed Fuel**	**2.26E-01**	**5.13E+03**	**4.84E+02**	

	A	B	C	D	E
1	1 Sievert	100000	mrem		
2	1 Ci	37000000000	Bq		
3					
4	Isotope	DCF (Sv/Bq)	DCF (mrem/Ci)	1691 MW Gap Inventory (Ci)	Dose (mrem)
5	I-130	0.000000000714	=B5*$B$1*$B$2	7250	=D5*C5
6	I-131	0.00000000889	=B6*$B$1*$B$2	534000	=D6*C6
7	I-132	0.000000000103	=B7*$B$1*$B$2	84500	=D7*C7
8	I-133	0.00000000158	=B8*$B$1*$B$2	363000	=D8*C8
9	I-134	0.0000000000355	=B9*$B$1*$B$2	81900	=D9*C9
10	I-135	0.000000000332	=B10*$B$1*$B$2	193000	=D10*C10
11					
12				Total	=SUM(E5:E10)
13	V_{MSL}	72000	gal		
14	a	100	gal/inch		
15	Elevation of the Main Steam Lines	620.5	inches above vessel 0		
16	Elevation of the Normal Water Level	535	inches above vessel 0		
17	H	=B15-B16	inches		
18	V_{Normal}	=B13-B14*B17	gal		
19	V_{Normal}	=B18*3785.41	cc		
20	Specific Volume @ 1000 psia	0.0216	ft^3/lbm		
21	water density @ 1000 psia	=0.016018/B20	g/cc		
22	M_{Normal}	=B19*B21	grams		
23					
24	Drywell Dose Rate for 100% Gap Release (1691 MWth)	22700	R/hr		
25	Torus Dose Rate for 100% Gap Release (1691 MWth)	2140	R/hr		
26					
27	**Threshold**	**Scaling Factors (F)**	**Drywell Dose Rate (R/hr)**	**Torus Dose Rate (R/hr)**	
28	**0.2 µCi/gm (TS Limit)**	**=$C$6*0.0000002*$B$22/($E$12)**	**=B28*$B$24**	**=B28*$B$25**	
29	**300 µCi/gm**	**=$C$6*0.0003*$B$22/($E$12)**	**=B29*$B$24**	**=B29*$B$25**	
30	**20% Failed Fuel**	**=0.2*1912/1691**	**=B30*$B$24**	**=B30*$B$25**	

CHECKLIST ITEMS[1]	YES	NO	N/A
GENERAL REQUIREMENTS			
1. If the calculation is being performed to a client procedure, is the procedure being used the latest revision?	☐	☐	☒
The calculation is being prepared to ENERCON's procedures.			
2. Are the proper forms being used and are they the latest revision?	☒	☐	☐
3. Have the appropriate client review forms/checklists been completed?	☐	☐	☒
The calculation is being prepared to ENERCON's procedures.			
4. Are all pages properly identified with a calculation number, calculation revision and page number consistent with the requirements of the client's procedure?	☒	☐	☐
5. Is all information legible and reproducible?	☒	☐	☐
6. Is the calculation presented in a logical and orderly manner?	☒	☐	☐
7. Is there an existing calculation that should be revised or voided?	☐	☒	☐
This is a new calculation to support implementing NEI 99-01 Rev. 6			
8. Is it possible to alter an existing calculation instead of preparing a new calculation for this situation?	☐	☒	☐
9. If an existing calculation is being used for design inputs, are the key design inputs, assumptions and engineering judgments used in that calculation valid and do they apply to the calculation revision being performed.	☒	☐	☐
10. Is the format of the calculation consistent with applicable procedures and expectations?	☒	☐	☐
11. Were design input/output documents properly updated to reference this calculation?	☐	☐	☒
12. Can the calculation logic, methodology and presentation be properly understood without referring back to the originator for clarification?	☒	☐	☐
OBJECTIVE AND SCOPE			
13. Does the calculation provide a clear concise statement of the problem and objective of the calculation?	☒	☐	☐
14. Does the calculation provide a clear statement of quality classification?	☒	☐	☐
15. Is the reason for performing and the end use of the calculation understood?	☒	☐	☐
16. Does the calculation provide the basis for information found in the plant's license basis?	☒	☐	☐
17. If so, is this documented in the calculation?	☒	☐	☐
18. Does the calculation provide the basis for information found in the plant's design basis documentation?	☐	☒	☐

	CHECKLIST ITEMS[1]	YES	NO	N/A
19.	If so, is this documented in the calculation?	☐	☐	☒
20.	Does the calculation otherwise support information found in the plant's design basis documentation?	☐	☒	☐
21.	If so, is this documented in the calculation?	☐	☐	☒
22.	Has the appropriate design or license basis documentation been revised, or has the change notice or change request documents being prepared for submittal?	☐	☐	☒
DESIGN INPUTS				
23.	Are design inputs clearly identified?	☒	☐	☐
24.	Are design inputs retrievable or have they been added as attachments?	☒	☐	☐
25.	If Attachments are used as design inputs or assumptions are the Attachments traceable and verifiable?	☐	☐	☒
26.	Are design inputs clearly distinguished from assumptions?	☒	☐	☐
27.	Does the calculation rely on Attachments for design inputs or assumptions? If yes, are the attachments properly referenced in the calculation?	☐	☒	☐
28.	Are input sources (including industry codes and standards) appropriately selected and are they consistent with the quality classification and objective of the calculation?	☒	☐	☐
29.	Are input sources (including industry codes and standards) consistent with the plant's design and license basis?	☒	☐	☐
30.	If applicable, do design inputs adequately address actual plant conditions?	☒	☐	☐
31.	Are input values reasonable and correctly applied?	☒	☐	☐
32.	Are design input sources approved?	☒	☐	☐
33.	Does the calculation reference the latest revision of the design input source?	☒	☐	☐
34.	Were all applicable plant operating modes considered?	☒	☐	☐
ASSUMPTIONS				
35.	Are assumptions reasonable/appropriate to the objective?	☒	☐	☐
36.	Is adequate justification/basis for all assumptions provided?	☒	☐	☐
37.	Are any engineering judgments used?	☐	☒	☐
38.	Are engineering judgments clearly identified as such?	☐	☐	☒
39.	If engineering judgments are utilized as design inputs, are they reasonable and can they be quantified or substantiated by reference to site or industry standards, engineering principles, physical laws or other appropriate criteria?	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
METHODOLOGY			
40. Is the methodology used in the calculation described or implied in the plant's licensing basis?	☐	☒	☐
41. If the methodology used differs from that described in the plant's licensing basis, has the appropriate license document change notice been initiated?	☐	☐	☒
42. Is the methodology used consistent with the stated objective?	☒	☐	☐
43. Is the methodology used appropriate when considering the quality classification of the calculation and intended use of the results?	☒	☐	☐
BODY OF CALCULATION			
44. Are equations used in the calculation consistent with recognized engineering practice and the plant's design and license basis?	☒	☐	☐
45. Is there reasonable justification provided for the use of equations not in common use?	☐	☐	☒
46. Are the mathematical operations performed properly and documented in a logical fashion?	☒	☐	☐
47. Is the math performed correctly?	☒	☐	☐
48. Have adjustment factors, uncertainties and empirical correlations used in the analysis been correctly applied?	☒	☐	☐
49. Has proper consideration been given to results that may be overly sensitive to very small changes in input?	☒	☐	☐
SOFTWARE/COMPUTER CODES			
50. Are computer codes or software languages used in the preparation of the calculation?	☐	☒	☐
51. Have the requirements of CSP 3.09 for use of computer codes or software languages, including verification of accuracy and applicability been met?	☐	☐	☒
52. Are the codes properly identified along with source vendor, organization, and revision level?	☐	☐	☒
53. Is the computer code applicable for the analysis being performed?	☐	☐	☒
54. If applicable, does the computer model adequately consider actual plant conditions?	☐	☐	☒
55. Are the inputs to the computer code clearly identified and consistent with the inputs and assumptions documented in the calculation?	☐	☐	☒
56. Is the computer output clearly identified?	☐	☐	☒
57. Does the computer output clearly identify the appropriate units?	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
58. Are the computer outputs reasonable when compared to the inputs and what was expected?	☐	☐	☒
59. Was the computer output reviewed for ERROR or WARNING messages that could invalidate the results?	☐	☐	☒
RESULTS AND CONCLUSIONS			
60. Is adequate acceptance criteria specified?	☒	☐	☐
61. Are the stated acceptance criteria consistent with the purpose of the calculation, and intended use?	☒	☐	☐
62. Are the stated acceptance criteria consistent with the plant's design basis, applicable licensing commitments and industry codes, and standards?	☒	☐	☐
63. Do the calculation results and conclusions meet the stated acceptance criteria?	☒	☐	☐
64. Are the results represented in the proper units with an appropriate tolerance, if applicable?	☒	☐	☐
65. Are the calculation results and conclusions reasonable when considered against the stated inputs and objectives?	☒	☐	☐
66. Is sufficient conservatism applied to the outputs and conclusions?	☒	☐	☐
67. Do the calculation results and conclusions affect any other calculations?	☐	☒	☐
68. If so, have the affected calculations been revised?	☐	☐	☒
69. Does the calculation contain any conceptual, unconfirmed or open assumptions requiring later confirmation?	☐	☒	☐
70. If so, are they properly identified?	☐	☐	☒
DESIGN REVIEW			
71. Have alternate calculation methods been used to verify calculation results?	☐	☐	☒
No, a Design Review was performed.			

Note:

1. Where required, provide clarification/justification for answers to the questions in the space provided below each question. An explanation is required for any questions answered as "No' or "N/A".

Originator: Aaron Holloway — 12/12/17

Print Name and Sign — Date

BREAKPOINTS FOR PRIMARY CONTAINMENT PRESSURE CONTROL

Pressure (psig)	Item of Interest	Significance
53 (Torus)	Primary Containment Pressure Limit (PCPL)	When PCPL is reached, containment venting is required.
~21.4 (Torus)	Pressure Suppression	Pressure Suppression Pressure exceeded for normal torus level
>11 (Torus) (11.15)	Drywell Sprays	Drywell sprays may be initiated if drywell parameters are within the Drywell Spray Initiation Limit and torus level is less than 13.5 feet
11.4 (Drywell)	Drywell Spray Initiation Limit (DWSIL) Break Point	Above 11.4 psig drywell pressure, drywell spray initiation is unrestricted by the DWSIL.
<11 (Torus) (11.15)	Torus Spray Initiation Pressure	Start torus sprays prior to 11 psig, if possible. If pressure is exceeded before torus sprays are initiated - initiate them anyway
2 (Drywell)	Drywell High Pressure Scram Setpoint	ECCS Initiation, Isolations and RPS defeats may be needed, EOP 1 and EOP 2 entry
1 (Drywell)	Drywell N2 Makeup Isolation	Drywell N2 makeup supply isolates if drywell pressure exceeds 1 psig

EOP 3
SECONDARY CONTAINMENT CONTROL

Fuel Pool Exhaust RIS 4131A[B] HI HI Rad Levels above 8 mR/hr

RB Vent Shaft Rad Monitor RIM 7606A[B] Rad Levels above 8 mR/hr

Area temperature above Max Normal Operating Limit (Table 6)

Area radiation level above Max Normal Operating Limit (Table 6)

Area water level above Max Normal Operating Limit (Table 6)

SC-1 ☐ Refer to EPIP 1.1 for EAL assessment.

While in this procedure:

IF	THEN
Fuel Pool Exhaust RIS 4131A[B] Radiation Level is above 8 mR/hr OR RB Vent Shaft RIM 7606A[B] Radiation Level is above 8 mR/hr OR Offgas Vent Pipe RM 4116A[B] is above HI-HI Rad Trip setpoint	• Verify isolation of Reactor Building HVAC. • Verify initiation of SBGT.
All the following conditions apply: • Reactor Building HVAC is isolated • Fuel Pool Exhaust RIS 4131A[B] Radiation Level is below 8 mR/hr • RB Vent Shaft RIM 7606A[B] Radiation Level is below 8 mR/hr • Offgas Vent Pipe RM 4116A[B] is below HI-HI Rad Trip setpoint	Restart Reactor Building HVAC. If necessary, bypass high drywell pressure and low RPV level isolation interlocks (Defeat 9).

Monitor and control SC temperature, radiation, and water levels concurrently. (10)

SC/T

SC/T-1 ☐ Operate available coolers in the affected area.

NOTE
Reactor Building HVAC isolations:
Fuel Pool Exhaust 8 mR/hr
RB Vent Shaft 8 mR/hr
Offgas Vent Pipe HI-HI Trip

Are reactor building exhaust radiation levels below isolation setpoints?
SC/T-2 ☐ No
SC/T-2 ☐ Yes

SC/T-3 ☐ Operate available main plant supply and exhaust fans.

SC/R

SC/L

SC/L-1 ☐ Operate available sump pumps to restore and maintain water level below the Max Normal Operating Limit (Table 6).

WAIT UNTIL
Any parameter is above its Max Normal Operating Limit (Table 6)
SC-2 ☐

SC-3 ☐ Isolate all systems discharging into the area except systems:
- Required to be operated by EOPs
  OR
- Required for damage control

Will RPV pressure reduction decrease leakage into secondary containment?
SC-4 ☐ No
SC-4 ☐ Yes

BEFORE
SC-5 ☐ any parameter reaches its Max Safe Operating Limit (Table 6)

EOP 1 ①

WAIT UNTIL
the same parameter exceeds its Max Safe Operating Limit in 2 or more areas (Table 6)
SC-6 ☐

SC-7 ☐ Emergency RPV Depressurization is Required

WAIT UNTIL
the same parameter exceeds its Max Safe Operating Limit in 2 or more areas (Table 6)
SC-8 ☐

SC-9 ☐ Begin reactor shutdown per IPOI 3, 4, or 5, as appropriate.

IPOI 3/4/5

CAUTIONS

10 — High spent fuel pool temperature or low spent fuel pool water level may affect conditions within secondary containment.

Table 6 — Secondary Containment Limits

	Parameter and Areas		Max Normal Operating Limit	Max Safe Operating Limit	Value/Trend
Temperature	Area/Location	Indicator	°F	°F	
	A RHR—SE Corner Room Area				
	RHR SE CORNER ROOM AMBIENT	TR/TDR 2000A Ch 1	130	140	
	RHR SE CORNER ROOM DIFFERENTIAL	TR/TDR 2000A Ch 2	50	N/A	
	B RHR—NW Corner Room Area				
	RHR NW CORNER ROOM AMBIENT	TR/TDR 2000B Ch 1	130	140	
	RHR NW CORNER ROOM DIFFERENTIAL	TR/TDR 2000B Ch 2	50	N/A	
	HPCI Room Area				
	HPCI EMER COOLER AMBIENT	TR/TDR 2225A[B] Ch 1	175	310	
	HPCI ROOM AMBIENT	TR/TDR 2225A Ch 2	175	310	
	HPCI ROOM DIFFERENTIAL	TR/TDR 2225A[B] Ch 4[3]	50	N/A	
	RCIC Room Area				
	RCIC EMER COOLER AMBIENT	TR/TDR 2425A[B] Ch 1	175	300	
	RCIC ROOM AMBIENT	TR/TDR 2425A Ch 2	175	300	
	RCIC ROOM DIFFERENTIAL	TR/TDR 2425A[B] Ch 4	50	N/A	
	Torus Area				
	TORUS CATWALK NORTH AMBIENT	TR/TDR 2425A Ch 3	150	165	
	TORUS CATWALK WEST AMBIENT	TR/TDR 2425B Ch 2	150	165	
	TORUS CATWALK SOUTH AMBIENT	TR/TDR 2225A Ch 3	150	165	
	TORUS CATWALK EAST AMBIENT	TR/TDR 2225B Ch 2	150	165	
	TORUS CATWALK EAST DIFF	TR/TDR 2425A Ch 5	50	N/A	
	TORUS CATWALK WEST DIFF	TR/TDR 2425B Ch 5	50	N/A	
	TORUS CATWALK SOUTHWEST DIFF	TR/TDR 2225A Ch 5	50	N/A	
	TORUS CATWALK SOUTH DIFF	TR/TDR 2225B Ch 4	50	N/A	
	RB 786' South Area				
	RWCU PUMP ROOM AMBIENT	TR/TDR 2700A[B] Ch 1	130	212	
	RWCU HX ROOM AMBIENT	TR/TDR 2700A[B] Ch 2,3	130	212	
	RB 757' South Area				
	RWCU ABOVE TIP ROOM AMBIENT	TR/TDR 2700A[B] Ch 4,5	111.5	150	
	Steam Tunnel Area				
	STEAM TUNNEL AMBIENT	TR/TDR 2425B Ch 3	160	300	
	STEAM TUNNEL DIFFERENTIAL	TR/TDR 2225B Ch 5	70	N/A	
Radiation	Area/Location	Indicator	mR/hr	mR/hr	
	RB 757' South Area				
	RB RAILROAD ACCESS AREA	RI 9167	10	100	
	SOUTH CRD MODULE AREA	RI 9169	10	100	
	TIP ROOM	RI 9176	60	600	
	RB 757' North Area				
	NORTH CRD MODULE	RI 9168	10	100	
	CRD REPAIR ROOM	RI 9170	15	150	
	RB 786' North Area				
	RWCU SPENT RESIN ROOM	RI 9173	100	10^3	
	RWCU PHASE SEP TANK ROOM	RI 9177	20	200	
	RB 786' South Area				
	RWCU PUMP ROOM	RI 9156	10^3	10^4	
	RCWU HX ROOM	RI 9157	10^3	10^4	
	RB 812' North Area				
	MAIN PLANT EXHAUST FAN ROOM	RI 9171	60	600	
	JUNGLE ROOM	RI 9155	60	600	
	Refuel Floor Area				
	NEW FUEL VAULT AREA	RI 9153	10	100	
	NORTH REFUEL FLOOR	RI 9163	10	100	
	SOUTH REFUEL FLOOR	RI 9164	10	100	
	SPENT FUEL POOL AREA	RI 9178	100	10^3	
Water Level	Area/Location	Indicator	inches	inches	
	HPCI ROOM	LI 3768	2	6	
	RCIC ROOM	LI 3769	3	6	
	"A" RHR & CS SECR	LI 3770	2	10	
	"B" RHR & CS NWCR	LI 3771	2	10	
	TORUS AREA	LI 3772	2	12	

EOP 4
RADIOACTIVITY RELEASE CONTROL

Offsite radioactivity release rate above the offsite release rate which requires an Alert

NOTE
The Alert EAL for Offsite Rad Release is **RA1** (refer to EPIP 1.1 for EAL assessment).

While in this procedure:

IF	THEN
Turbine Building HVAC shuts down	1. Stop turbine building supply fans. 2. Restart turbine building exhaust fans.
Group 3 isolation exists AND Reactor Building Kaman Monitors have a HI-HI Alarm	1. Stop main plant exhaust fans. 2. Stop turbine building supply fans. 3. Verify at least **one** turbine building exhaust fan operating in high speed. 4. Verify Group 3 isolation complete.

RR-1 ☐ Isolate all primary systems discharging radioactivity outside the primary and secondary containments except for systems required to be operated by EOPs.

NOTE
To be a primary system, leakage though an unisolated break in the system decreases as RPV pressure is reduced.

WAIT UNTIL
any primary system is discharging radioactivity into areas outside the primary and secondary containments
RR-2 ☐

RR-3 ☐ Begin reactor shutdown per IPOI 3, 4, or 5, as appropriate.

IPOI 3/4/5

BEFORE
RR-4 ☐ offsite radioactivity release rate reaches a **General Emergency**

NOTE
The General Emergency EAL for Offsite Rad Release is **RG1** (refer to EPIP 1.1 for EAL assessment).

EOP 1 ①

RR-5 ☐ Emergency RPV Depressurization is Required

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EOP 3 – SECONDARY CONTAINMENT CONTROL EOP 4 – RADIOACTIVITY RELEASE CONTROL	REV. 21

BREAKPOINTS FOR PRIMARY CONTAINMENT PRESSURE CONTROL

Pressure (psig)	Item of Interest	Significance
53 (Torus)	Primary Containment Pressure Limit (PCPL)	When PCPL is reached, containment venting is required.
~21.4 (Torus)	Pressure Suppression	Pressure Suppression Pressure exceeded for normal torus level
>11 (Torus) (11.15)	Drywell Sprays	Drywell sprays may be initiated if drywell parameters are within the Drywell Spray Initiation Limit and torus level is less than 13.5 feet
11.4 (Drywell)	Drywell Spray Initiation Limit (DWSIL) Break Point	Above 11.4 psig drywell pressure, drywell spray initiation is unrestricted by the DWSIL.
<11 (Torus) (11.15)	Torus Spray Initiation Pressure	Start torus sprays prior to 11 psig, if possible. If pressure is exceeded before torus sprays are initiated - initiate them anyway
2 (Drywell)	Drywell High Pressure Scram Setpoint	ECCS Initiation, Isolations and RPS defeats may be needed, EOP 1 and EOP 2 entry
1 (Drywell)	Drywell N2 Makeup Isolation	Drywell N2 makeup supply isolates if drywell pressure exceeds 1 psig

EOP 2 – PRIMARY CONTAINMENT CONTROL

Entry conditions:

- Torus water level below 10.1 ft
- Torus water level above 10.4 ft
- Average torus water temperature above 95°F
- Average drywell temperature above 150°F
- Drywell pressure above 2.0 psig

PC-1 ☐ Refer to EPIP 1.1 for EAL assessment.

Monitor and control all primary containment parameters concurrently.

T/L

T/L-1 ☐ Control torus water level between 10.1 ft and 10.4 ft using only those systems not required for adequate core cooling.

- Raise torus water level using one or more of the following:
  - Core Spray (OI 151)
  - HPCI (OI 152)
  - Torus spray from external sources
- Lower torus water level using RHR (OI 149)
  - If necessary, bypass radwaste isolation Defeat 13).

IF		THEN
High Level:	Torus water level cannot be maintained below 10.4 ft	Go to (22)
Low Level:	Torus water level cannot be maintained above 10.1 ft	Go to (21)

Low Level (Below 10.1 ft) (21)

T/L-2 ☐ Maintain torus water level above 7.1 ft.

T/L-3 ☐ WAIT UNTIL torus water level cannot be maintained above 7.1 ft

EOP 1 (1)

T/L-4 ☐ BEFORE torus water level drops to 7.1 ft

T/L-5 ☐ Emergency RPV Depressurization is Required

T/L-6 ☐ Maintain torus water level above 5.8 ft.

T/L-7 ☐ WAIT UNTIL torus water level cannot be maintained above 5.8 ft

T/L-8 ☐ Secure HPCI if not required for adequate core cooling. — (11)

High Level (Above 10.4 ft) (22)

T/L-9 ☐ Maintain torus water level below 13.5 ft.

T/L-10 ☐ WAIT UNTIL torus water level cannot be maintained below 13.5 ft

EOP 1 (1)

T/L-11 ☐ Terminate drywell sprays.

T/L-12 ☐ Terminate RPV injection from sources external to the primary containment not required for adequate core cooling or to shut down the reactor.

T/L-13 ☐ WAIT UNTIL torus water level cannot be restored and maintained below 13.8 ft

T/L-14 ☐ Emergency RPV Depressurization is Required

T/T

T/T-1 ☐ Control torus water temperature below 95°F using RHR in torus cooling (OI 149). — (3)

T/T-2 ☐ WAIT UNTIL torus water temperature cannot be maintained below 95°F

T/T-3 ☐ Operate all available torus cooling using only RHR pumps not required for adequate core cooling.

T/T-4 ☐ BEFORE torus water temperature reaches 110°F.

EOP 1 (1)

T/T-5 ☐ Maintain torus water temperature below the Heat Capacity Limit (Graph 4).

T/T-6 ☐ WAIT UNTIL torus water temperature and RPV pressure cannot be maintained below the Heat Capacity Limit (Graph 4)

T/T-7 ☐ Emergency RPV Depressurization is Required

DW/T

DW/T-1 ☐ Control drywell temperature below 150°F using available drywell cooling systems. — (1)

DW/T-2 ☐ WAIT UNTIL drywell temperature cannot be maintained below 150°F

DW/T-3 ☐ Maximize drywell cooling.

- If necessary, bypass drywell cooling isolation and fan speed interlock (Defeat 4).

IF	THEN
Drywell pressure drops below 2.0 psig	Verify containment sprays isolate.

DW/T-4 ☐ BEFORE drywell temperature reaches 280°F

EOP 1 (1)

DW/T-5 ☐ IF torus water level is below 13.5 ft, AND drywell temperature is below the Drywell Spray Initiation Limit (Graph 7), THEN 1. Shut down recirc pumps. 2. Initiate drywell sprays using only pumps not required for adequate core cooling (Table 9). — (5)

DW/T-6 ☐ WAIT UNTIL drywell temperature cannot be restored and maintained below 340°F

DW/T-7 ☐ Emergency RPV Depressurization is Required

PC/P

PC/P-1 ☐ Control primary containment pressure below 2.0 psig using Containment Atmosphere Control System (OI 573).

PC/P-2 ☐ WAIT UNTIL primary containment pressure cannot be maintained below 2.0 psig

IF	THEN
Torus pressure drops below 2.0 psig	Terminate torus sprays.
Drywell pressure drops below 2.0 psig	Verify containment sprays isolate.
Primary containment pressure reduction is required to: • Restore and maintain adequate core cooling OR • Reduce the total offsite radiation dose	Vent the primary containment: OK to exceed offsite rad release limits. • IF torus water level is below 16 ft THEN vent the torus (SEP 301.1). • IF the torus cannot be vented OR torus water level is at or above 16 ft THEN vent the drywell (SEP 301.2).

PC/P-3 ☐ BEFORE torus pressure reaches 11 psig

PC/P-4 ☐ Initiate torus sprays using only pumps not required for adequate core cooling (Table 9). — (5)

- OK to use sources external to the primary containment if primary containment water level can be restored and maintained below 95 ft.

PC/P-5 ☐ WAIT UNTIL torus pressure exceeds 11 psig

PC/P-6 ☐ IF torus water level is below 13.5 ft, AND drywell temperature is below the Drywell Spray Initiation Limit (Graph 7), THEN 1. Shut down recirc pumps 2. Initiate drywell sprays using only pumps not required for adequate core cooling (Table 9). — (5)

PC/P-7 ☐ WAIT UNTIL torus pressure cannot be maintained below the Pressure Suppression Pressure (Graph 5)

PC/P-8 ☐ Emergency RPV Depressurization is Required

PC/P-9 ☐ WAIT UNTIL torus pressure cannot be maintained below 53 psig

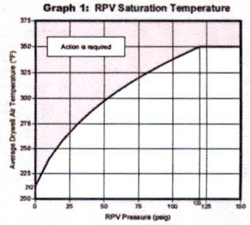
PC/P-10 ☐ Vent the primary containment to control torus pressure below 53 psig:

- OK to exceed offsite rad release limits.
- IF torus water level is below 16 ft THEN vent the torus (SEP 301.1).
- IF the torus cannot be vented OR torus water level is at or above 16 ft THEN vent the drywell (SEP 301.2).

Table 9	Spray Sources			
Systems	Pressure Range (psig)	Capacity (gpm)	Procedure	Status
RHR	0-260	0-4800	OI 149	
RHR Service Water	0-270	0-2400	AIP 401	
Fire System	0-125	0-2500	AIP 404	
Well Water A/C B/D	0-125	0-750 0-1200	AIP 403	
GSW	0-125	0-5100	AIP 403	
ESW	0-110	0-1200	AIP 402	
Condensate Service Water	0-150	0-100	AIP 405	

(3)

Graph 1: RPV Saturation Temperature



CAUTIONS

(1) The following restrictions apply to RPV level instruments:

1. If drywell air temperature is above the RPV Saturation Temperature (Graph 1), water in the instrument legs may boil. If boiling is suspected:
   a. Subtract 23 inches from Fuel Zone and Narrow Range GEMAC indications.
   b. Do not use Floodup and Wide Range Yarway instruments.
2. Floodup and Wide Range instruments may not be used below the Minimum Indicated Level for the indicated drywell temperature.

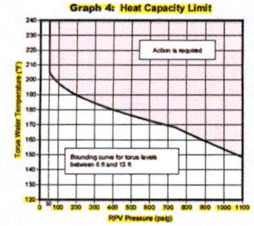
Level Instrument	Temperature Instrument	Drywell Temp (°F)	MIL (in.)
Wide Range Yarway LI-4539 (+8 to +218 in.)	TR-4383A Channel 1 (red)	100-150 151-200 201-250 251-300 301-350 Upscale	+9 +12 +16 +21 +25 +47
Wide Range Yarway LI-4540 (+8 to +218 in.)	TR-4383B Channel 2 (green)		
Floodup Range LI-4541 (+158 to +458 in.)	TR-4383A Channel 1 (red)	100-150 151-200 201-250 251-300 301-350 Upscale	+169 +175 +182 +190 +199 +256

(3) Operation of HPCI, RCIC, Core Spray, or RHR with suction from the torus and pump flow above the NPSH or vortex limit may damage equipment.

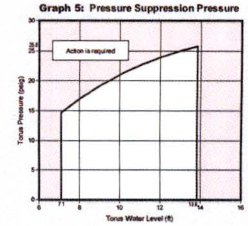
(5) Reducing primary containment pressure will reduce NPSH for pumps taking suction from the torus.

(11) Operation of HPCI with the turbine exhaust opening uncovered (5.8 ft) will increase torus pressure and may challenge primary containment limits.

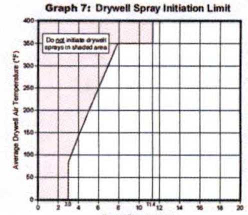
Graph 4: Heat Capacity Limit



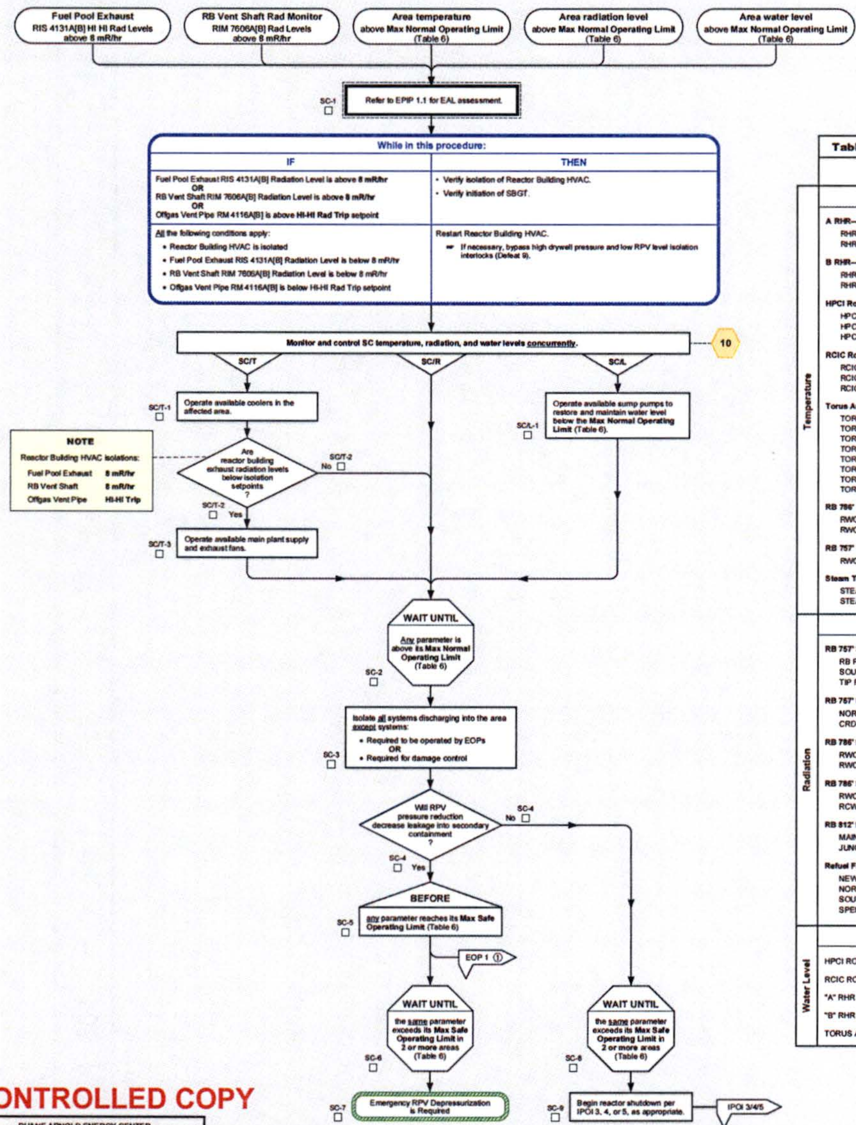
Graph 5: Pressure Suppression Pressure



Graph 7: Drywell Spray Initiation Limit



EOP 3
SECONDARY CONTAINMENT CONTROL

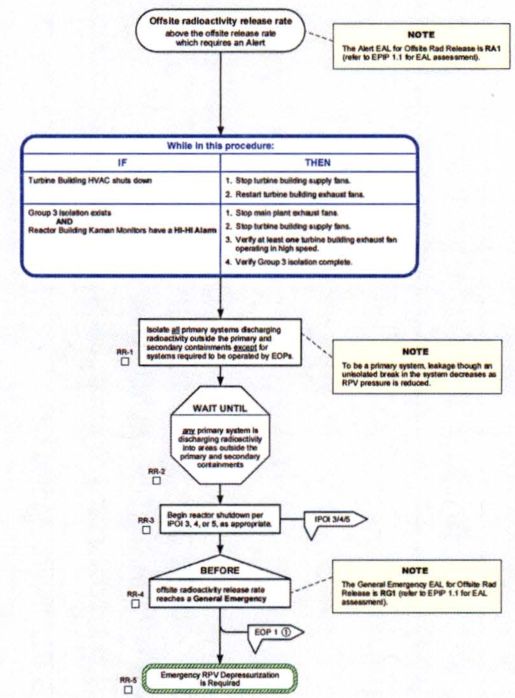


CAUTIONS

10 — High spent fuel pool temperature or low spent fuel pool water level may affect conditions within secondary containment.

Table 6	Secondary Containment Limits				
	Parameter and Areas		Max Normal Operating Limit	Max Safe Operating Limit	Value/Trend
Temperature	Area/Location	Indicator	°F	°F	
	A RHR—SE Corner Room Area				
	RHR SE CORNER ROOM AMBIENT	TR/TDR 2000A Ch 1	130	140	
	RHR SE CORNER ROOM DIFFERENTIAL	TR/TDR 2000A Ch 2	50	N/A	
	B RHR—NW Corner Room Area				
	RHR NW CORNER ROOM AMBIENT	TR/TDR 2000B Ch 1	130	140	
	RHR NW CORNER ROOM DIFFERENTIAL	TR/TDR 2000B Ch 2	50	N/A	
	HPCI Room Area				
	HPCI EMER COOLER AMBIENT	TR/TDR 2225A[B] Ch 1	175	310	
	HPCI ROOM AMBIENT	TR/TDR 2225A Ch 2	175	310	
	HPCI ROOM DIFFERENTIAL	TR/TDR 2225A[B] Ch 4[3]	50	N/A	
	RCIC Room Area				
	RCIC EMER COOLER AMBIENT	TR/TDR 2425A[B] Ch 1	175	300	
	RCIC ROOM AMBIENT	TR/TDR 2425A Ch 2	175	300	
	RCIC ROOM DIFFERENTIAL	TR/TDR 2425A[B] Ch 4	50	N/A	
	Torus Area				
	TORUS CATWALK NORTH AMBIENT	TR/TDR 2425A Ch 3	150	165	
	TORUS CATWALK WEST AMBIENT	TR/TDR 2425B Ch 2	150	165	
	TORUS CATWALK SOUTH AMBIENT	TR/TDR 2225A Ch 3	150	165	
	TORUS CATWALK EAST AMBIENT	TR/TDR 2225B Ch 2	150	165	
	TORUS CATWALK EAST DIFF	TR/TDR 2425A Ch 5	50	N/A	
	TORUS CATWALK WEST DIFF	TR/TDR 2425B Ch 5	50	N/A	
	TORUS CATWALK SOUTHWEST DIFF	TR/TDR 2225A Ch 5	50	N/A	
	TORUS CATWALK SOUTH DIFF	TR/TDR 2225B Ch 4	50	N/A	
	RB 786' South Area				
	RWCU PUMP ROOM AMBIENT	TR/TDR 2700A[B] Ch 1	130	212	
	RWCU HX ROOM AMBIENT	TR/TDR 2700A[B] Ch 2,3	130	212	
	RB 757' South Area				
	RWCU ABOVE TIP ROOM AMBIENT	TR/TDR 2700A[B] Ch 4,5	111.5	150	
	Steam Tunnel Area				
	STEAM TUNNEL AMBIENT	TR/TDR 2425B Ch 3	160	300	
	STEAM TUNNEL DIFFERENTIAL	TR/TDR 2225B Ch 5	70	N/A	
Radiation	Area/Location	Indicator	mR/hr	mR/hr	
	RB 757' South Area				
	RB RAILROAD ACCESS AREA	RI 9167	10	100	
	SOUTH CRD MODULE AREA	RI 9169	10	100	
	TIP ROOM	RI 9176	60	600	
	RB 757' North Area				
	NORTH CRD MODULE	RI 9168	10	100	
	CRD REPAIR ROOM	RI 9170	15	150	
	RB 786' North Area				
	RWCU SPENT RESIN ROOM	RI 9173	100	10^3	
	RWCU PHASE SEP TANK ROOM	RI 9177	20	200	
	RB 786' South Area				
	RWCU PUMP ROOM	RI 9156	10^3	10^4	
	RCWU HX ROOM	RI 9157	10^3	10^4	
	RB 812' North Area				
	MAIN PLANT EXHAUST FAN ROOM	RI 9171	60	600	
	JUNGLE ROOM	RI 9155	60	600	
	Refuel Floor Area				
	NEW FUEL VAULT AREA	RI 9153	10	100	
	NORTH REFUEL FLOOR	RI 9163	10	100	
	SOUTH REFUEL FLOOR	RI 9164	10	100	
	SPENT FUEL POOL AREA	RI 9178	100	10^3	
Water Level	Area/Location	Indicator	inches	inches	
	HPCI ROOM	LI 3768	2	6	
	RCIC ROOM	LI 3769	3	6	
	"A" RHR & CS SECR	LI 3770	2	10	
	"B" RHR & CS NWCR	LI 3771	2	10	
	TORUS AREA	LI 3772	2	12	

EOP 4
RADIOACTIVITY RELEASE CONTROL



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EOP 3 – SECONDARY CONTAINMENT CONTROL EOP 4 – RADIOACTIVITY RELEASE CONTROL	REV. 21

PROBABLE ANNUNCIATORS

None

PROBABLE INDICATIONS

1C35

- The amber DESIGN BASIS EARTHQUAKE (DBE) light is ON.
- The amber OPERATING BASIS EARTHQUAKE (OBE) light is ON.
- The amber .01G RECORDERS RUNNING light is ON.
- The white CONTINUITY light is OFF.
- The Seismic Wailing Alarm is sounding.
- Building vibration.

A Cooling Tower Valve House

- No power indicating light is operable.

INFORMATION

Earthquake	OBE	DBE
Ground Acceleration	0.06g	0.12g

CONDITIONAL STATEMENTS

IF while performing this procedure:

IF	Control Room access is regained **AND** personnel are available	**THEN**	when directed by the Emergency Response and Recovery Director resume control of unaffected components from the Control Room **AND**	______
			maintain control of Division II components from 1C388 until operability of Control Room instruments, indications and controls has been verified.	______

NOTE

______

- Operations personnel evacuate to the Remote Shutdown Panel except: the STA, Shift Communicator, and on-site personnel not on shift evacuate to the TSC.
- The preferred evacuation route to the Remote Shutdown Panel is out the back door of the Control Room, and down the stairs. Emergency lighting is provided for this path.
- The alternate evacuation route to the Remote Shutdown Panel is out the front door of the Control Room, and down the stairs to access control. Emergency lighting is provided for this path.
- Since fire induced failure in 1C05 could adversely affect manual scram circuits, the initiation of ATWS ARI/RPT provides a redundant and diverse means of control rod insertion.

CAUTION

______

For Control Room evacuation as the result of a fire, transfer of control at panels 1C388, 1C389, 1C390, 1C391, 1C392 **is required to be completed within 20 minutes**.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ 0.2 μCi/gm.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor coolant specific activity > 0.2 μCi/gm and ≤ 2.0 μCi/gm DOSE EQUIVALENT I-131.	-------------NOTE----------- LCO 3.0.4.c is applicable. ---------------------------	
	A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
OR	AND	
Reactor Coolant specific activity > 2.0 μCi/gm DOSE EQUIVALENT I-131.	B.2.1 Isolate all main steam lines.	12 hours
	OR	
		(continued)

2.0 uci/gm chosen as EAL threashold since levels above that activity directly influence continued plant operation.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. ≤ 5 gpm unidentified LEAKAGE;

b. ≤ 25 gpm total LEAKAGE averaged over the previous 24 hour period; and

c. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Unidentified LEAKAGE not within limit. <u>OR</u> Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce unidentified LEAKAGE increase to within limits. <u>OR</u>	4 hours (continued)

Developer Notes:

For EAL #1 leak rate value, entered the higher of 10 gpm or the value specified in DAEC's Technical Specifications for this type of leakage.

- 5 gpm per DAEC Tech Specs, so 10 gpm used in EAL

For EAL #2 enter the higher of 25 gpm or the value specified in DAEC's Technical Specifications for this type of leakage.

- DAEC uses a total leakage (identified + unidentified) spec of 25 gpm averaged over 24 hour period, so 25 gpm used in the EAL

(9) During the approach to criticality, the operator withdrawing control rods should pause long enough between control rod notches to allow neutron count rate and period to stabilize, thus allowing a slow and controlled approach to the critical condition.

(10) When a control rod reaches position 48, perform a coupling check by attempting to withdraw the rod past position 48. If uncoupling should occur, stop control rod withdrawal, notify the CRS, and perform ARP 1C05A, D-7 (ROD OVERTRAVEL OUT).

(11) If criticality occurs significantly earlier or later than expected, notify the CRS.

(12) Approach the power range on a stable period of about 60-150 seconds. Do not achieve a sustained period of less than 50 seconds. If the period becomes too short, insert the notch and monitor for subcriticality.

(13) Each operable IRM channel must be indicating at least 5/40 scale on Range 1 prior to SRM count rate exceeding 10^6 cps with SRMs fully inserted. One IRM recorder on each RPS System should be in second speed (30 s/div) during startups while in the IRM Range. However, during extended stable operation in the IRM Range, it is permissible to shift the recorders to normal speed (30 min/div).

(14) Reactor plant heatup with MO-4629 and MO-4630, A/B RECIRC PUMP DISCH BYP in the closed position may cause bonnet over pressurization, resulting in failure of the valve to open due to pressure lock and damage to valve internals.

(15) Do not establish a vacuum in the main condenser until:

(a) Steam seals are in operation.

(b) Turbine is on turning gear.

(c) Lube Oil Temperature > 80°F.

(16) Do not exceed a reactor pressure of 400 psig unless a reactor feed pump is in operation or the MSIVs are closed and the RCIC or HPCI Systems are operating.

(17) Do not retract IRMs until the MODE SWITCH is in RUN.

(18) Do not operate the mechanical vacuum pump above 10% reactor power to minimize the possibility of a hydrogen explosion or an untreated radioactivity release.

(19) Place the MODE SWITCH in RUN prior to reaching 12% reactor power.

BREAKPOINTS FOR REACTOR LEVEL CONTROL
Page 2 of 2

RPV Level (inches)	Item of Interest	Significance
-25	Minimum Steam Cooling RPV Water Level (MSCRWL)	• No guarantee that fuel cladding temperature can be kept <1500 °F • ED required in EOP 1 before -25 inches • SAG Entry in EOP 1 if cannot restore and maintain level above -25 inches and spray cooling cannot be established • Lower end of level control band in ATWS level/power control • Loss of ACC in ATWS Steam Cooling & SAG Entry
-39	Elevation of top of Jet Pump Suction (~2/3 Core Height)	• RPV water level following DBA LOCA • SAG Entry in EOP 1 if cannot restore and maintain level above -39 inches while spray cooling

ATTACHMENT 5

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATING TO LICENSE AMENDMENT REQUEST TSCR-166

UPDATED DAEC EAL SCHEME WALLBOARDS
[FOR INFORMATION ONLY]

2 pages follow

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
R 1	RG1.1 (Notes 1, 2, 3, 4) Reading on ANY Table R-1 effluent radiation monitor greater than column "GE" for 15 minutes or longer. [illegible]	RS1.1 (Notes 1, 2, 3, 4) Reading on ANY Table R-1 effluent radiation monitor greater than column "SAE" for 15 minutes or longer. [illegible]	[illegible]	[illegible]
R 2	[illegible]	[illegible]	[illegible]	[illegible]
R 3			[illegible]	
E 1				[illegible]
H 1	[illegible]	[illegible]	[illegible]	[illegible]
H 2				[illegible]
H 3				[illegible]
H 4				[illegible]
H 5		[illegible]	[illegible]	
H 6	[illegible]	[illegible]	[illegible]	[illegible]

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

NOTICE

[illegible]

Table H-1 Fire Areas

[illegible]

Table E-1 Cask Dose Rates

[illegible]

Modes: 1 Power Operation | 2 Startup | 3 Hot Shutdown | 4 Cold Shutdown | 5 Refueling | DEF Defueled

Duane Arnold Energy Center
EAL Classification Matrix
HOT CONDITIONS
(Modes 1, 2, 3)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
S 1	[illegible]	[illegible]	[illegible]	[illegible]
S 2	[illegible]	[illegible]		
S 3			[illegible]	[illegible]
S 4				[illegible]
S 5				[illegible]
S 6	[illegible]	[illegible]	[illegible]	[illegible]
S 7				[illegible]
S 8			[illegible]	

Table S-2 Significant Transients

[illegible]

Table S-1 Safety System Parameters

[illegible]

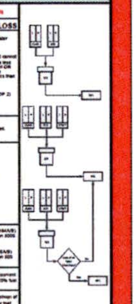
Table S-4 Hazardous Events

[illegible]

F	FG1	FS1	FA1
	Loss of ANY two barriers AND Loss OR Potential Loss of the third barrier (Table F-1)	Loss OR potential loss of ANY two barriers (Table F-1)	ANY Loss OR ANY Potential Loss of EITHER the Fuel Clad OR RCS barrier (Table F-1)

Table F-1 DAEC EAL Fission Product Barrier Table

Category	Fuel Clad (FC) Barrier: LOSS	Fuel Clad (FC) Barrier: POTENTIAL LOSS	Reactor Coolant System (RCS) Barrier: LOSS	Reactor Coolant System (RCS) Barrier: POTENTIAL LOSS	Containment (CNTMT) Barrier: LOSS	Containment (CNTMT) Barrier: POTENTIAL LOSS
1	[illegible]	Not Applicable	[illegible]	Not Applicable	[illegible]	[illegible]
2	[illegible]	[illegible]	[illegible]	Not Applicable	Not Applicable	[illegible]
3	Not Applicable	Not Applicable	[illegible]	[illegible]	[illegible]	Not Applicable
4	[illegible]	Not Applicable	[illegible]	Not Applicable	Not Applicable	[illegible]
5	[illegible]	Not Applicable	Not Applicable	Not Applicable	Not Applicable	[illegible]
6	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]



	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

R — Abnormal Rad Levels / Rad Effluent

1 Rad Effluent

General Emergency

RG1.1 (Notes 1, 2, 3, 4)
Reading on ANY Table R-1 effluent radiation monitor greater than column "GE" for 15 minutes or longer.

RG1.2 (Note 4)
Dose assessment using actual meteorology indicates doses greater than 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]

RG1.3 (Notes 1, 2)
Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.

Site Area Emergency

RS1.1 (Notes 1, 2, 3, 4)
Reading on ANY Table R-1 effluent radiation monitor greater than column "SAE" for 15 minutes or longer.

RS1.2 (Note 4)
Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY. [Preferred]

RS1.3 (Notes 1, 2)
Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.

Alert

RA1.1 (Notes 1, 2, 3, 4)
Reading on ANY Table R-1 effluent radiation monitor greater than column "ALERT" for 15 minutes or longer.

RA1.2 (Note 4)
Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond SITE BOUNDARY. [Preferred]

RA1.3 (Notes 1, 2)
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for one hour of exposure.

RA1.4 (Notes 1, 2)
Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

Unusual Event

RU1.1 (Notes 1, 2, 3)
Reading on ANY Table R-1 effluent radiation monitor greater than column "NOUE" for 60 minutes or longer.

RU1.2 (Notes 1, 2)
Reading on ANY effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

RU1.3 (Notes 1, 2)
Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODAM limits for 60 minutes or longer.

2 Irradiated Fuel Event

General Emergency

RG2.1 (Note 1)
Spent fuel pool level cannot be restored to at least 16.36 feet for 60 minutes or longer.

Site Area Emergency

RS2.1
Lowering of spent fuel pool level to 16.36 feet.

Alert

RA2.1
Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2
Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by a Hi Rad alarm for ANY of the following ARMS:
- Spent Fuel Pool Area, RI-9178
- North Refuel Floor, RI-9163
- New Fuel Vault Area, RI-9153
- South Refuel Floor, RI-9164

OR
Reading greater than 5 R/hr on ANY of the following radiation monitors (In Mode 5 only):
- NW Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

RA2.3
Lowering of spent fuel pool level to 25.17 feet.

Unusual Event

RU2.1
a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following:
- Report to control room (visual observation)
- Fuel pool level indication (LI-3413) less than 36 feet and lowering
- WR GEMAC Floodup indication (LI-4541) coming on scale

AND
b. UNPLANNED rise in area radiation levels as indicated by ANY of the following radiation monitors:
- Spent Fuel Pool Area, RI-9178
- North Refuel Floor, RI-9163
- New Fuel Vault Area, RI-9153
- South Refuel Floor, RI-9164
- NW Drywell Area Hi Range Rad Monitor, RIM-9184A
- South Drywell Area Hi Range Rad Monitor, RIM-9184B

Table R-1 - Effluent Monitor Classification Thresholds

	Monitor	GE	SAE	Alert	NOUE
Gaseous	Reactor Building ventilation rad monitor	[illegible]	[illegible]	[illegible]	[illegible]
	Turbine Building ventilation rad monitor	[illegible]	[illegible]	[illegible]	[illegible]
	Offgas bleed rad monitor	[illegible]	[illegible]	[illegible]	[illegible]
	SJAE rad monitor	--	[illegible]	[illegible]	[illegible]
Liquid	GSW rad monitor	--	--	[illegible]	[illegible]
	RHRSW & ESW rad monitor	--	--	[illegible]	[illegible]
	RHRSW & ESW Rupture Disc rad monitor	--	--	[illegible]	[illegible]

3 Area Radiation Levels

Alert

RA3.1
Dose rate greater than 15 mR/hr in ANY of the following areas:
- Control Room (RM-9162)
- Central Alarm Station (by survey)

E — ISFSI

1 Confinement Boundary

Table E-1 Cask Dose Rates

Unusual Event

E-HU1.1
Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by a radiation reading greater than the values shown on Table E-1 on the spent fuel cask.

H — Hazards

1 Security

General Emergency

HG1.1
a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.

AND
b. EITHER of the following has occurred:
1. ANY of the following safety functions cannot be controlled or maintained
- Reactivity control
- RPV water level
- RCS heat removal

OR
2. Damage to spent fuel has occurred or is IMMINENT

Site Area Emergency

HS1.1
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the DAEC Security Shift Supervision.

Alert

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the DAEC Security Shift Supervision.

HA1.2
A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

Unusual Event

HU1.1
A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by DAEC Security Shift Supervision.

HU1.2
Notification of a credible security threat directed at DAEC.

HU1.3
A validated notification from the NRC providing information of an aircraft threat.

2 Seismic Event

Unusual Event

HU2.1
Seismic event greater than Operating Basis Earthquake (OBE) as indicated by receipt of the Amber Operating Basis Earthquake Light and the sealing seismic alarm on 1C35.

NOTES

- Note 1: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.
- Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- Note 4: The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.
- Note 5: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- Note 6: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.
- Note 7: EAL HU3.4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.
- Note 8: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

3 Natural or Tech. Hazard

Unusual Event

HU3.1
A tornado strike within the PROTECTED AREA

HU3.2
Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.

HU3.3
Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

HU3.4 (Note 7)
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

4 Fire

Table H-1 Fire Areas
- 1G31 DG and Day Tank Rooms
- 1G21 DG and Day Tank Rooms
- Battery Rooms
- Essential Switchgear Rooms
- Cable Spreading Room
- Torus Room
- Intake Structure
- Pumphouse
- Drywell
- Torus
- NE, NW, SE Corner Rooms
- HPCI Room
- RCIC Room
- RHR Valve Room
- North CRD Area
- South CRD Area
- CSTs
- Control Building
- Remote Shutdown Panel 1C388 Area
- Panel 1C55/56 Area
- SBGT Room

Unusual Event

HU4.1
a. A FIRE is NOT extinguished within 15 min. of ANY of the following FIRE detection indications:
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND
b. The FIRE is located within ANY Table H-1 plant rooms or areas

HU4.2
a. Receipt of a single fire alarm with no other indications of a FIRE

AND
b. The FIRE is located within ANY Table H-1 plant rooms or areas

AND
c. The existence of a FIRE is not verified within 30 minutes of alarm receipt.

HU4.3
A FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60 minutes of the initial report, alarm or indication.

HU4.4
A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

5 Control Room Evacuation

Site Area Emergency

HS5.1
a. An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).

AND
b. Control of ANY of the following key safety functions is not reestablished within 20 minutes
- Reactivity control
- RPV water level
- RCS heat removal

Alert

HA5.1
An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (1C388).

6 ED Judgment

General Emergency

HG6.1
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Site Area Emergency

HS6.1
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Alert

HA6.1
Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Unusual Event

HU6.1
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

C — Cold SD/Refueling System Malfunction

1 RCS Level

General Emergency

CG1.1 (Note 1)
a. RPV level less than +15 inches for 30 minutes or longer

AND
b. ANY indication from the Secondary Containment Challenge Table C-1.

CG1.2 (Note 1)
a. RPV level cannot be monitored for 30 minutes or longer

AND
b. Core uncovery is indicated by EITHER of the following:
- Drywell Monitor (9184A/B) reading greater than 5.0 R/hr
- UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery.

AND
c. ANY indication from the Secondary Containment Challenge Table C-1.

Site Area Emergency

CS1.1
a. CONTAINMENT CLOSURE not established.

AND
b. RPV level less than +64 inches.

CS1.2
a. CONTAINMENT CLOSURE established.

AND
b. RPV level less than +15 inches.

CS1.3 (Note 1)
a. RPV level cannot be monitored for 30 minutes or longer.

AND
b. Core uncovery is indicated by EITHER of the following:
- Drywell Monitor (9184A/B) reading greater than 5.0 R/hr.
- UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool of sufficient magnitude to indicate core uncovery.

Alert

CA1.1
Loss of RPV inventory as indicated by level less than 119.5 inches.

CA1.2 (Note 1)
a. RPV level cannot be monitored for 15 minutes or longer

AND
b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool due to a loss of RPV inventory.

Unusual Event

CU1.1 (Note 1)
UNPLANNED loss of reactor coolant results in RPV level less than a required lower limit for 15 minutes or longer.

CU1.2
a. RPV level cannot be monitored

AND
b. UNPLANNED level rise in Drywell/Reactor Building Equipment or Floor Drain sump, or Suppression Pool.

Table C-1 Secondary Containment Challenge
- CONTAINMENT CLOSURE not established *
- Drywell Hydrogen or Torus Hydrogen greater than 6% AND Drywell Oxygen or Torus Oxygen greater than 5%
- UNPLANNED increase in containment pressure
- Secondary containment radiation monitors above max safe operating limits (MSOL) of EOP 3, Table 6

* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

2 Loss of Emergency AC Power

Alert

CA2.1 (Note 1)
Loss of ALL offsite and ALL onsite AC Power to 1A3 and 1A4 buses for 15 minutes or longer.

Unusual Event

CU2.1 (Note 1)
a. AC power capability to 1A3 and 1A4 buses is reduced to a single power source for 15 minutes or longer.

AND
b. Any additional single power source failure will result in loss of ALL AC power to SAFETY SYSTEMS.

3 RCS Temp.

Table C-2 RCS Heat-up Duration Thresholds

RCS Integrity	Containment Closure Status	Heat-up Duration
Intact	Not applicable	60 minutes*
Not intact	Established	20 minutes*
	Not Established	0 minutes

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.

Alert

CA3.1 (Note 1)
UNPLANNED increase in RCS temperature to greater than 212°F for greater than the duration specified in Table C-2.

CA3.2
UNPLANNED RCS pressure increase greater than 10 psig due to a loss of RCS cooling.

Unusual Event

CU3.1
UNPLANNED increase in RCS temperature to greater than 212°F

CU3.2 (Note 1)
Loss of ALL RCS temperature and RPV level indication for 15 minutes or longer.

4 Loss of Vital DC Power

Unusual Event

CU4.1 (Note 1)
Indicated voltage is less than 105 VDC on BOTH Div 1 and Div 2 125 VDC buses for 15 minutes or longer.

5 Loss of Comm.

NOTES

- Note 1: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.
- Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- Note 4: The pre-calculated effluent monitor values presented in EAL 1.1 should only be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.
- Note 5: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- Note 6: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.
- Note 7: EAL HU3.4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.
- Note 8: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

Unusual Event

CU5.1
Loss of ALL of the following onsite communication methods:
- Plant Operations Radio System
- In-Plant Phone System
- Plant Paging System (Gaitronics)

CU5.2
Loss of ALL of the following offsite response organization communications methods:
- DAEC All-Call phone
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system
- FTS Phone system

CU5.3
Loss of ALL of the following NRC communications methods:
- FTS Phone system
- All telephone lines (PBX and commercial)
- Cell Phones (including fixed cell phone system)
- Control Room fixed satellite phone system

6 Hazardous Event Affecting Safety Systems

Table C-3 Hazardous Events
- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the Shift Manager or Emergency Director

Alert

CA6.1 (Notes 5, 6)
a. The occurrence of ANY of the Table C-3 hazardous events.

AND
b. 1. Event damage has caused indications of degraded performance in one train of a SAFETY SYSTEM needed for the current operating mode.

AND
2. EITHER of the following:
- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode.

OR
- The event has resulted in VISIBLE DAMAGE to the second train of a SAFETY SYSTEM needed for the current operating mode.

Modes: 1 Power Operation | 2 Startup | 3 Hot Shutdown | 4 Cold Shutdown | 5 Refueling | DEF Defueled

NEXTera ENERGY RESOURCES

Duane Arnold Energy Center
EAL Classification Matrix
COLD CONDITIONS
(Modes 4, 5, and Defueled)