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2CAN031001

March 31, 2010

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: License Amendment Request
Technical Specification Changes and Analyses Relating to Use of Alternative Source Term
Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

REFERENCE   NRC letter to Kevin T. Walsh, dated October 29, 2009, "Arkansas Nuclear One, Unit 2 – Issuance of Amendment Re: Adoption of Technical Specification Task Force (TSTF) Change Traveler TSTF-448, Revision 3, "Control Room Envelope Habitability" (TAC No. MD7175) (2CNA100902)

Dear Sir or Madam:

10 CFR 50.67(b) states:

> *A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under Sec. 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report.*

In accordance with the provisions of 10 CFR 50.90, Entergy Operations, Inc. (Entergy) is submitting a request for license amendment to implement an alternative source term (AST) for Arkansas Nuclear One, Unit 2 (ANO-2). In support of AST implementation and also in accordance with the provisions of 10 CFR 50.90, Entergy is including a request for an amendment to the Technical Specifications (TS) for ANO-2. The proposed amendment will modify TS requirements related to the use of an AST associated with accident offsite and control room dose consequences. Implementation of AST supports adoption of the control room envelope habitability controls in accordance with Technical Specification Task Force (TSTF)-448, Revision 3 (Reference 1). TSTF-448 related changes were approved by the NRC for ANO-2 on October 29, 2009 and implemented at the site in February 2010. The justification for the proposed TS changes is included in Attachment 1.

A mark-up of affected Technical Specification (TS) pages is included in Attachment 2.

Attachment 3 includes a summary of the results of dose analyses for the events that are expected to produce the most limiting dose consequences, in accordance with Regulatory Guide (RG) 1.183, along with figures and summary tables of dose analyses input using the AST methodology.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that the change involves no significant hazards consideration. The bases for these determinations are included in Attachment 1.

Entergy requests acceptance of the ANO-2 use of AST as discussed in Attachments 1 and 3, and approval of the proposed TS changes by April 1, 2011. Once approved, the amendment shall be implemented within 90 days. Although this request is neither exigent nor emergency, your prompt review is requested.

This letter contains no new commitments.

If you have any questions or require additional information, please contact David Bice at 479-858-5338.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on March 31, 2010.

Sincerely,

Original signed by Kevin T. Walsh

KTW/dbb

Attachments: 1. Analysis of Proposed Change
2. Proposed Technical Specification Changes (mark-up)
3. Summary Results, Figures, and Summary Tables of Dose Analyses Input

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Attachment 1 to

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Analysis of Proposed Change

1.0   DESCRIPTION

This letter is a request to amend Operating License NPF-6 for Arkansas Nuclear One, Unit 2 (ANO-2). The proposed change will implement an alternative source term (AST) for determining accident offsite and control room doses. A license amendment is required for AST implementation in accordance with 10 CFR 50.67(b) which states:

> *A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under Sec. 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report.*

The proposed amendment will modify Technical Specification (TS) definitions, requirements, and terminology related to the use of an AST associated with accident offsite and control room dose consequences. Implementation of AST supports adoption of the control room envelope habitability controls in accordance with Technical Specification Task Force (TSTF)-448, Revision 3 (Reference 8).

A markup of affected TS pages is included in Attachment 2.

The AST methodology as established in Regulatory Guide (RG) 1.183 is used to calculate the offsite and control room radiological consequences for ANO-2 to support an increase in allowed control room unfiltered air inleakage and to remove the current analysis assumption of an Emergency Core Cooling System (ECCS) pump passive seal failure. Attachment 3 includes a summary of the results of dose analyses for the following events that are expected to produce the most limiting dose consequences, along with figures and summary tables of dose analyses input using AST methodology of RG 1.183.

- Loss of Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Control Rod Ejection Accident (CREA)
- Locked Reactor Coolant Pump Rotor Accident (LRA)

2.0   PROPOSED CHANGE

The change to the ANO-2 licensing basis involves the adoption of an AST for calculating accident doses to control room personnel and offsite receptors. The following TS revisions are requested to support assumptions associated with the new AST analyses:

Definitions - The TS definitions for Dose Equivalent I-131 (DEI) and Dose Equivalent X-133 (DEX) (TS definitions 1.18 and 1.19, respectively) are revised to address terminology appropriate for plants with an AST and reference the AST-related federal guidelines.

TS 3.4.8 - The DEX activity limit prescribed in Surveillance Requirement (SR) 4.4.8.1 is revised from 1200 µCi/gm to 3100 µCi/gm as a result of utilizing the AST. The new limit is conservatively assumed in the AST analysis to ensure offsite and control room dose limitations will not be challenged.

TS 6.5.12 - The Control Room Envelope Habitability Program is revised to reflect the AST-related term of Total Effective Dose Equivalent (TEDE).

3.0   BACKGROUND

The AST and methodology described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 6) and in RG 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 7), provide regulatory guidance for the implementation of the AST. Revision of a plant licensing basis from the Technical Information Document (TID)-14844 source term to an AST involves the preparation of dose consequence analyses. Demonstration that the results satisfy the regulatory acceptance criteria and NRC approval of the requested change establishes the acceptability of the use of the AST for ANO-2.

The current ANO-2 license bases limit for unfiltered air leaking into the control room while in the emergency recirculation ventilation mode of operation is 61 cubic feet per minute (cfm). Adoption of the AST will increase this limit to 250 cfm. Attachment 3 of this letter provides relevant background information with regard to the analysis and assumptions supporting adoption of the AST. The proposed changes to the TSs are consistent with the terminology and limits necessary to adopt an AST.

4.0   TECHNICAL ANALYSIS

The justification for full implementation of the AST methodology, as defined in RG 1.183, into the ANO-2 design basis accident analyses is provided in Attachment 3 of this letter. Analysis of the dose consequences of the LOCA, FHA, MSLB, SGTR, CREA, and LRA have been completed using the RG 1.183 methodology. The analyses used assumptions consistent with proposed changes to the ANO-2 licensing basis and the calculated doses have been verified to not exceed the defined acceptance criteria.

Attachment 3 supports a maximum allowable control room unfiltered air inleakage of 250 cfm. The use of 250 cfm as a design basis value was verified to be above the unfiltered inleakage value determined through tracer gas testing and analysis consistent with the resolution of issues identified in Nuclear Energy Institute (NEI) 99-03 (Reference 11) and Generic Letter (GL) 2003-01 (Reference 1). The analysis also utilized the proposed new TS DEX limit of 3100 µCi/gm in order to verify control room and offsite dose consequences would remain acceptable relative to this limit. Based on the information provided in the attachments to this letter, Entergy has concluded that the proposed TS changes and the adoption of the AST are acceptable for ANO-2.

5.0   REGULATORY ANALYSIS

5.1   No Significant Hazards Consideration

As provided by 10 CFR 50.67, Entergy Operations, Inc. (Entergy) is implementing the use of an Alternate Source Term (AST) and the dose calculation methodology described in NUREG-1465 and Regulatory Guide (RG) 1.183 to calculate accident doses to control room and offsite receptors following postulated events that result in the release of radioactive material from the reactor fuel at Arkansas Nuclear One, Unit 2 (ANO-2). The AST and associated methodology define the amount, isotopic composition, physical and chemical characteristics, and timing of radioactive material releases following postulated events. Transport of the material to the control room and offsite areas is modeled, and the resulting Total Effective Dose Equivalent (TEDE) is determined. Regulatory acceptance criteria account for the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). In accordance with 10 CFR 50.67(b), licensees wishing to adopt an AST must apply for a license amendment in accordance with 10 CFR 50.90.

In support of the revised analysis applying AST, Technical Specification (TS) limits for Reactor Coolant System (RCS) Dose Equivalent Xenon-133 (DEX) activity is increased as verified acceptable through the AST supporting analysis. Other changes to the TSs involve the adoption of terminology and the regulatory guidance on which AST is based.

Entergy has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1.   Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

    Response: No.

    The use of an AST is recognized in 10 CFR 50.67. RG 1.183 provides guidance for implementation of an AST. The AST involves quantities, isotopic composition, chemical and physical characteristics, and release timing of radioactive material for use as inputs to accident dose analyses. As such, the AST cannot affect the probability of occurrence of a previously evaluated accident. In addition, the increase in the DEX activity limit and the terminology/reference changes proposed for the ANO-2 TSs are unrelated to accident initiators. No facility equipment, procedure, or process changes are required in conjunction with implementing the AST that could increase the likelihood of a previously analyzed accident. The proposed changes in the source term and the methodology for the dose consequence analyses follow the guidance of RG 1.183. As a result, there is no increase in the likelihood of existing event initiators.

    Regarding accident consequences, the increase in the DEX activity limit acts to support the analysis results given the application of an AST. The proposed limit was utilized as an assumption in the AST analysis and determined to be acceptable. The results of accident dose analyses using the AST are compared to TEDE acceptance criteria that account for the sum of deep dose equivalent (for external exposure) and committed effective dose equivalent (for internal exposure). Dose results were previously

compared to separate limits on whole body, thyroid, and skin doses as appropriate for the particular accident analyzed. The results of the revised dose consequences analyses demonstrate that the regulatory acceptance criteria are met for each analyzed event. The proposed TS terminology/reference changes are consistent with the analysis and adoption of an AST. Implementing the AST involves no facility equipment, procedure, or process changes that could affect the radioactive material actually released during an event. Subsequently, no conditions have been created that could significantly increase the consequences of any of the events being evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of any of the events being evaluated.

2.   Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The AST involves quantities, isotopic composition, chemical and physical characteristics, and release timing of radioactive material for use as inputs to accident dose analyses. As such, the AST cannot create the possibility of a new or different kind of accident. In addition, the increased DEX activity limit and proposed terminology/reference changes within the TSs are unrelated to accident initiators and are supported by AST adoption.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3.   Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

Implementing the AST is relevant only to calculated accident dose consequences. The results of the revised dose consequences analyses demonstrate that the regulatory acceptance criteria are met for each analyzed event. In addition, the increased DEX activity limit and proposed terminology/reference changes within the TSs support adoption of the AST methodologies, have been determined to result in acceptable dose consequence and do not result in a significant impact to any margin to safety. The AST does not affect the transient behavior of non-radiological parameters (e.g., RCS pressure, Containment pressure) that are pertinent to a margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2   Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the existing TSs with regard to RCS DEX activity limit and certain changes to terminology and regulatory references in support of AST methodologies. As discussed in Section 2.0 above, Entergy proposes to revise TS Surveillance Requirement (SR) 4.4.8.1, to be consistent with the activities assumed in the AST analyses presented in Attachment 3 of this submittal.

Compliance with General Design Criterion (GDC) 19, Control Room, is demonstrated for the proposed change based on meeting the dose limit to control room personnel of 5 Rem TEDE. The original licensing basis had been established based on the whole body, thyroid, and skin dose limits now described in the ANO-2 Safety Analysis Report (SAR). As stated in Regulatory Guide (RG) 1.183, the applicable acceptance criteria to establish compliance with GDC 19 for facilities licensed to use an AST is the 5 Rem TEDE criterion of 10 CFR 50.67(b)(2)(iii). The analysis demonstrates that ANO-2 complies with this 5 Rem TEDE requirement and, therefore, Entergy Operations, Inc. (Entergy) requests an operating license amendment to implement the new AST licensing basis.

5.3   Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

Concerning the types and amounts of effluents that may be released offsite, the AST involves some changes in assumed quantities and characteristics of radioactive material that are inputs to offsite accident dose calculations. These are changes to calculation assumptions only. No facility equipment, procedure, or process changes are associated with use of the AST that affect actual releases. Therefore, implementation of the AST will not increase the quantities or alter the types of radioactive material actually released if an event were to occur. Implementation of the AST also has no effect on the actual or calculated effluents arising from normal operation.

With respect to occupational doses, the AST only involves a change in accident dose calculation inputs and methodology. Calculated doses meet TEDE criteria. No aspect of implementing the AST involves facility equipment, procedure, or process changes that would increase actual onsite doses if an event were to occur. The AST does not result in actual or calculated changes in the normal radiation levels in the facility, or in the type or quantity of radioactive materials processed during normal operation.

Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this proposed amendment.

6.0   PRECEDENCE

Several plants have been approved for the use of the AST, including Grand Gulf Nuclear Station in March 2001, D. C. Cook in November 2002, and River Bend Station in March 2003. Those submittals included various TS changes which were justified by the change to the AST. An AST was also approved for ANO, Unit 1 (ANO-1) by NRC letter dated October 21, 2009 (Reference 12).

7.0   REFERENCES

1.   NRC letter dated June 12, 2003, *Generic Letter 2003-01, Control Room Habitability* (0CNA060308)

2.   Entergy Letter dated August 28, 2003, *Response to Generic Letter 2003-01* (0CAN080304)

3.   Entergy Letter dated February 14, 2007, *Supplemental Response to Generic Letter 2003-01 Regarding Control Room Habitability* (0CAN020701)

4.   Entergy Letter dated June 30, 2003, *License Amendment Request to Add a New Control Room Emergency Ventilation System Surveillance Requirement* (1CAN060302)

5.   Entergy Letter dated January 10, 1980, *Control Room Toxic Gas Protection* (1CAN018008)

6.   NUREG-1465, *Accident Source Terms for Light Water Nuclear Power Plants*, L. Soffer, et. al., February 1995

7.   Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, USNRC, July 2000

8.   Federal Register Notice dated January 17, 2007, *Notice of Availability of Technical Specification Improvement To Modify Requirements Regarding Control Room Envelope Habitability Using the Consolidated Line Item Improvement Process* (*FRN* Federal Register, Vol. 72, No. 10, page 2022)

9.   Entergy Letter dated June 17, 2004, *Proposed Operating License Amendment for Revised ANO-1 Control Room Habitability Analysis* (1CAN060401)

10.   Entergy letter dated May 6, 2006, *Withdrawal of Proposed Operating License Amendment for Revised ANO-1 Control Room Habitability Analysis* (1CAN050601)

11.   NEI 99-03, *Control Room Habitability Assessment Guidance*, June 2001

12   NRC letter dated October 21, 2009, *Arkansas Nuclear One, Unit No. 1 – Issuance of Amendment Re: Use of Alternate Source Term (TAC No. MD7178)* (1CNA100901)

Attachment 2 to

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Proposed Technical Specification Changes (mark-up)

UNIDENTIFIED LEAKAGE

1.15   UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or controlled leakage.

PRESSURE BOUNDARY LEAKAGE

1.16   PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary to secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

AZIMUTHAL POWER TILT – T_q

1.17   AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

DOSE EQUIVALENT I-131

1.18   DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same committed effective dose equivalent (CEDE)~~when inhaled~~ as the quantity and isotopic mixture~~combined activities~~ of ~~iodine isotopes~~ I-131, I-132, I-133, I-134, and I-135 actually present. The CEDE dose conversion factors used to determine~~ation the~~of DOSE EQUIVALENT I-131 shall be performed using Table 2.1~~III~~ of EPA Federal Guidance Report No. 11, 1988~~TID-14844, AEC, 1962~~, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion~~Calculation of Distance Factors for Power and Test Reactor Sites~~."

DOSE EQUIVALENT XE-133

1.19   DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective~~the average disintegration energies derived from the~~ dose conversion factors for air submersion listed~~provided~~ in Table III.1 of EPA Federal Guidance Report No. 12, 1993~~ICRP Publication 2~~, "External Exposure to Radionuclides in Air, Water, and Soil~~Report of ICRP Committee II on Permissible Dose for Internal Radiation" for non-fuel damage events~~."

STAGGERED TEST BASIS

1.20   A STAGGERED TEST BASIS shall consist of:

   a.   A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

   b.   The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21   The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8   RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity
         shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION

Note: The provisions of Specification 3.0.4.c are applicable to ACTION a and b.

a.   With the DOSE EQUIVALENT I-131 not within limit:

      1.   Verify DOSE EQUIVALENT I-131 ≤ 60 µCi/gm once every 4 hours, and

      2.   Restore DOSE EQUIVALENT I-131 within limit within 48 hours.

b.   With the DOSE EQUIVALENT XE-133 not within limit, restore DOSE EQUIVALENT
      XE-133 within limit within 48 hours.

c.   With the requirements of ACTION a and/or b not met, or with DOSE EQUIVALENT I-131
      > 60 µCi/gm, be in at least HOT STANDBY in 6 hours and in COLD SHUTDOWN within the
      following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1   Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 31200 µCi/gm
           once every 7 days.*

4.4.8.2   Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 µCi/gm:*

           a.   once every 14 days, and

           b.   between 2 and 6 hours after THERMAL POWER change of ≥ 15% RATED
                 THERMAL POWER within a 1 hour period.

* Only required to be performed in MODE 1.

6.5.12   Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem Total Effective Dose Equivalent (TEDE)whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

a.   The definition of the CRE and the CRE boundary.

b.   Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c.   Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d.   Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of one train every 18 months. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e.   The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f.   The provisions of Specification 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

Attachment 3 to

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**Summary Results, Figures, and
Summary Tables of Dose Analyses Input**

Table of Contents

Figures

Tables

1.0 Radiological Consequences Utilizing the Alternative Source Term Methodology

1.1 Introduction

The current Arkansas Nuclear One, Unit 2 (ANO-2), licensing basis for the radiological analyses for accidents discussed in Chapter 15 of the Safety Analysis Report (SAR) is based on methodologies and assumptions that are primarily derived from Technical Information Document (TID)-14844 (Reference 2) and other early guidance.

Regulatory Guide (RG) 1.183 (Reference 6) provides guidance on application of Alternative Source Terms (AST) in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10 CFR 50.67 (Reference 5). Because of advances made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents, 10 CFR 50.67 was issued to allow holders of operating licenses to voluntarily revise the traditional accident source terms used in the design basis accident (DBA) radiological consequence analyses with ASTs.

1.2 Evaluation Overview and Objective

The AST methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for ANO-2 to support an increase in allowed control room unfiltered air inleakage and to remove the current Emergency Core Cooling System (ECCS) pump seal passive failure assumption.

The following limiting SAR (Reference 1) Chapter 15 accidents are analyzed:

- Loss-of-Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Locked Reactor Coolant Pump Rotor Accident (LRA)
- Control Rod Ejection Accident (CREA)

Each accident and the specific input and assumptions are described in Section 2.0 of this attachment. These analyses provide for a bounding allowable control room unfiltered air inleakage of 250 cfm. The use of 250 cfm as a design basis value was established to be above the unfiltered inleakage value determined through testing and analysis consistent with the resolution of issues identified in Nuclear Energy Institute (NEI) 99-03 (Reference 25) and Generic Letter (GL) 2003-01 (Reference 26).

1.3 Proposed Changes to the ANO-2 Licensing Basis

Entergy Operations, Inc. (Entergy) proposes to revise the ANO-2 licensing basis to implement the AST described in RG 1.183 through reanalysis of the radiological consequences of the SAR Chapter 15 accidents listed in Section 1.2 above. As part of this full implementation of AST, the following changes are assumed in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.

- New onsite (control room) atmospheric dispersion factors are used.

- Dose conversion factors for inhalation and submersion are from Federal Guidance Reports (FGR) Nos. 11 and 12, respectively.

- An increased Dose Equivalent Xenon-133 (DEX) value is utilized.

- An increased value for control room unfiltered air inleakage is assumed.

- The passive failure assumption of an ECCS pump seal is eliminated.

- Two entire fuel assemblies are assumed to be damaged following a FHA.

1.4   Compliance with Regulatory Guidelines

The revised ANO-2 accident analyses addressed in this attachment follow the guidance provided in RG 1.183.

1.5   Computer Codes

The following computer codes are used in performing the AST analyses:

Computer Code	Version	Reference	Purpose
ARCON-96	1997	11	Atmospheric Dispersion Factors
ORIGEN-S	2.0	12	Core Fission Product Inventory
SCAP-II	1.0	13	Direct Shine Dose Calculations
DORT	3.2	14	Control Room Attenuation Factors
BUGLE-96	N/A	15	ENDF/B-VI Data Files
RADTRAD	3.0.3	16	Radiological Dose Calculations

1.5.1   ARCON-96 - used to calculate relative concentrations (X/Q factors) in plumes from nuclear power plants at control room intakes in the vicinity of the release point using plant meteorological data.

1.5.2   ORIGEN - used for calculating the buildup, decay, and processing of radioactive materials.

1.5.3   SCAP-II - used to analyze shielding and estimate exposure from gamma radiation.

1.5.4   DORT – used to calculate control room attenuation factors.

1.5.5   BUGLE-96 – provides the gamma ray cross-section library.

1.5.6   RADTRAD - estimates the radiological doses at offsite locations and in the control room of nuclear power plants as consequences of postulated accidents. The code considers the timing, physical form (i.e., vapor or aerosol) and chemical species of the radioactive material released into the environment. The dose conversion factors used by RADTRAD are from Federal Guidance Report Nos. 11 and 12 (FGR-11 and FGR-12).

1.6   Radiological Evaluation Methodology

1.6.1   Analysis Input Assumptions

Common analysis input assumptions include those for the control room ventilation system and dose calculation model (Section 1.6.3), direct shine dose (Section 1.6.5), radiation source terms (Section 1.7), and atmospheric dispersion factors (Section 1.8). Event-specific assumptions are discussed in the event analyses in Section 2.0.

1.6.2   Acceptance Criteria

Offsite and Control Room doses must meet the guidelines of RG 1.183 and requirements of 10 CFR 50.67. The acceptance criteria for specific postulated accidents are provided in Table 6 of RG 1.183.

1.6.3   Control Room Ventilation System Description

The Control Room Emergency Air Conditioning System (CREACS) and Control Room Emergency Ventilation System (CREVS) are designed to assure control room habitability. The overall description of the system is discussed in ANO-2 SAR Section 9.4.1. Following a design basis accident, the control room is pressurized to maintain a positive pressure differential. Makeup air for pressurization is filtered before entering the control room.

The control room is normally air conditioned by one of two 100 percent capacity air conditioning units. One unit is normally running, with the other in standby, isolated from the control room by shutoff dampers. The standby unit is available for manual actuation in the event of failure of the operating unit. Adequate fresh air makeup is supplied via the operating air conditioning unit.

The control room air is continuously monitored for high radiation via redundant monitors located in the inlet ductwork of the control room normal air conditioning system through which outside air is supplied. The control room inlet air radiation monitor system consists of two identical monitor strings each having an auto-ranging digital ratemeter, pre-amplifier, and Beta-Gamma sensitive scintillation detector. These monitors have a minimum detectable level of 1E-5 µCi/cc of Cs-137 with no lead shield. A variable setpoint for the monitor is set slightly above equilibrium background level and alarms are provided for high radiation and circuit failure. The configuration is such that either monitor can isolate the control room on high radiation or circuit failure conditions.

In the event of high radiation, the normal air conditioning system and the normal control room ventilation system is completely isolated from control room within 5 seconds following detector actuation. The actuation level for high radiation is sufficiently below hazardous radiation levels to minimize operator dose during an accident and is sufficiently above normally experienced background levels to minimize spurious actuations. The control room isolation dampers in the supply and return ductwork are spring loaded such that they fail closed upon loss of air or power.

The single supply and single return isolation dampers are each actuated by either of two solenoid valves. Under these conditions control room air is recirculated by the automatically-actuated emergency air filtering system. The emergency air filtering system consists of two redundant filter trains, VSF-9 and 2VSF-9. Due to space limitations, the two trains are designed differently. One filter train (2VSF-9) consists of a fan, roughing filters, HEPA filters, and a 4-inch deep bed charcoal adsorber rated for 2000 cfm. The other train (VSF-9) consists of a fan, one filter unit assembly rated for 2000 cfm and an outside air filter unit rated for 333 cfm, each with the necessary roughing filters, HEPA filters and 2-inch charcoal tray adsorber. Both VSF-9 and 2VSF-9 were originally designed to provide ~333 cfm outside air to minimize unfiltered air inleakage to the combined control room envelope, which was in turn based upon providing greater than or equal to 0.5 volume changes per hour based upon Standard Review Plan 6.4, Revision.2. However, the actual outside air drawn by 2VSF-9 is ~465 cfm, as measured during control room tracer gas testing in November 2001.

Calculations have been performed that indicate that even with the higher 2VSF-9 makeup air flow rate, operation of VSF-9 with 333 cfm makeup air is limiting in terms of control room radiation dose (Reference 27). For either train, outside air will be filtered through four inches of charcoal adsorber and the recirculation air will go through at least two inches of charcoal bed. Fan failure is monitored by a flow switch with an indicating light in the control room. On an indication of fan failure, the standby unit is manually started.

The CREVS design is based on a minimum of three room air changes per hour for the combined control room volume (Reference 28). The filter banks are sized in accordance with manufacturer's recommendations for maximum efficiency. The control room operator has manual control for selecting fan, filter, and air conditioning unit operations in order to ensure satisfactory control room conditions following an accident. The system is designed to perform its safety functions and maintaining a habitable environment in the control room envelope during isolated conditions.

ANO-2 shares a common control room envelope with ANO-1. The net volume of the common envelope serviced by the CREACS and CREVS is 40,000 cubic feet.

The habitability systems (air filtration and ventilation equipment with associated instrumentation, controls and radiation monitoring) are capable of performing the associated design functions assuming a single active component failure coincident with a loss of offsite power. Redundant equipment which is essential to safety is powered from separate safety related electrical buses such that loss of one bus does not prevent the CREACS and CREVS from fulfilling the specified safety function.

1.6.4   Control Room Dose Calculation Model

The control room model includes a recirculation filter model along with filtered air intake, unfiltered air inleakage, and an exhaust path. System performance, sequence, and timing of operational evolutions associated with the CREVS are discussed below. CREVS parameters assumed in the analyses are provided in Table 1.6.3-1. The dispersion factors for use in modeling the control room during each mode of operation are provided in Table 1.8.1-1. Control room occupancy factors and assumed breathing rates are those prescribed in RG 1.183. Figures 1.8.1-1 through 1.8.1-7 provide sketches illustrating the ANO-2 containment and auxiliary building layout, including the location of potential onsite radiological release points with respect to the control room air intakes. The elevations of release points and intakes used in the control room AST dose assessments are also provided in the figures.

The CREVS contains a filtration system for removal of radioactive iodine and particulate material that may enter the control room during the course of the event. Calculation of the dose to operators in the control room requires modeling of various system configurations and operating evolutions of the CREVS during the course of the accident. The control room model defines two concurrent air intake paths representing the defined CREVS air intake and the unfiltered inleakage into the control room. Outside air can enter the control room through the filtration/ventilation system from either of two ventilation intake locations that are located over 40 feet apart on top of the ANO-1 auxiliary building. Due to the diverse locations, these intakes are assigned different dispersion factors for calculating the concentration of radioactive isotopes in the air drawn in through that intake due to the activity released from various locations on the site during an accident. Unfiltered outside air can also enter the control room directly. Modeling of the control room addresses these factors as they apply to the various release locations for each analyzed event. Details of the control room modeling for each event are described in subsequent event analyses sections.

During normal operation, one of two ANO-2 normal air supply fans will be operating supplying fresh air to the control room envelope. The second fan is maintained in standby. In addition, ANO-1 has a separate control room normal ventilation system and one of two of its normal air supply fans will also be operating during normal operation, the second fan being maintained in standby. Since the two ANO units share a common control room envelope, unfiltered outside air due to operation of one fan on each unit has been considered in the ANO-2 AST analyses. The ANO-1 fans are rated at 13,900 cfm each, while the ANO-2 fans are rated at 21,300 cfm each. The supply air from both units' fans is isolated within 5 seconds of receipt of a high radiation signal from any of the supply duct radiation detectors. Thus, for the ANO-2 AST analyses, 35,200 cfm of unfiltered air is conservatively assumed to pass through the control room envelope for 10 seconds following an event.

Isolation of control room normal ventilation also results in actuation of the CREVS. The CREVS recirculates air within the control room through a filtration system to remove contaminates that have already been drawn into or have leaked into the control room. The flow rate of this recirculation air is 1667 cfm. During the course of the event, fresh, filtered air is added to the control room by the CREVS at a rate of 333 cfm (the more limiting of the air intake measurements) in order to maintain positive pressure and air quality. This filtered intake is assumed to continue throughout the remaining duration of the dose calculation.

The control room model uses the applicable dispersion factors for the worst-case air intake location when assessing the dose due to the normal ventilation supply prior to its isolation, due to the filtered makeup air post-isolation and due to unfiltered inleakage.

1.6.5   Direct Shine Dose

The total control room dose also requires the calculation of direct shine contributions from the activities in the primary containment atmosphere and in the radioactive plume in the environment. The direct shine dose contribution from the primary containment atmosphere was calculated for the LOCA event. This direct shine dose to a person in the control room is based on the conservative assumptions of minimum control room wall thickness and an occupancy factor of 1.0 over the 30-day accident duration.

Direct shine dose due to the radioactive plume in the environment (i.e. the external radioactive cloud that envelops the control room) is considered separately and is calculated directly by RADTRAD for each event. The control room attenuation factors calculated for the primary containment atmosphere direct shine dose are utilized in the RADTRAD external radioactive cloud direct shine dose calculations as well. These factors are discussed further below.

Gamma ray dose rates interior to the control room are calculated using the SCAP-II computer code. The SCAP code is based on the point kernel method for calculation of radiation dose in complex source-shield geometries. In this case, the geometry applied in the SCAP calculation includes simulations of the walls and ceiling of the control room as well as of the outer structure of the reactor containment building. The cylindrical portion of the containment is treated as a 116 foot diameter cylindrical shell with a thickness of 3.75 feet of concrete. The height of the cylinder is considered to be 179 feet. The control room is treated as a structure with concrete wall and ceiling thickness of 1.5 feet. The gamma ray source strengths used in the calculations are determined using the ORIGEN-S computer code. Activity releases following the DBA event are based on the AST scenario defined in NUREG-1465 and RG 1.183. Reactor Coolant System (RCS) activity is released during the first 30 seconds after a LOCA followed by a "gap release" phase during which all of the gap activity (3% of the total core inventory of volatile nuclides) is instantaneously released. In addition, for accidents where long-term fuel cooling or core geometry are not maintained, an additional release of 2% of the inventory of volatile core inventories are considered to be released at a constant rate over a 30 minute gap release phase. Volatile species are considered to be noble gases, halogens, and alkali metals. Following the gap release phase, an in-vessel release phase is considered, which endures for 1.3 hours. These releases are assumed to be at a constant rate over the release interval.

The analysis also considers removal of some of the containment airborne activity by the Containment Spray System (CSS). The amount of removal is consistent with the LOCA dose analyses described in Section 2.1. The removal in this evaluation is conservatively assumed to take place at several intervals during the spray period, at which time overall decontamination factors (DFs) are applied.

Control room attenuation factors are calculated for individual isotopes using the DORT discrete ordinates transport code in one-dimensional slab geometry. For each isotope, the calculations are performed using unit source strength and the gamma ray emission spectrum from the ORIGEN nuclear data libraries. As in the case of the direct dose calculation, conservatism is introduced into the attenuation factor determination by using the minimum control room wall thickness in the analysis. All calculations are run using the BUGLE-96 cross-section library, which includes a 20 energy group gamma ray cross-section matrix. The ratio of the calculated dose rate near the interior surface of the wall to that at the exterior surface defines the minimum attenuation factor for each isotope. The control room attenuation factors are then utilized in the RADTRAD analyses when calculating the direct shine dose due to the external radioactive cloud.

1.7   Radiation Source Terms

1.7.1   Fission Product Inventory

The source term data to be used in performing AST analyses for ANO-2 are summarized in the following tables.

     Table 1.7.2-1 - Primary Coolant Source Term

     Table 1.7.3-1 - Secondary Coolant Source Term

     Table 1.7.4-1 – Core Isotopic Inventory for Fuel Failure Events

Note that the source terms provided in the referenced tables do not include any decay before the start of the events. Decay time assumptions are applied in the RADTRAD cases for individual event analysis. For example, the RADTRAD case for the FHA analysis would account for the required decay time before the movement of fuel is allowed (as determined by TSs).

1.7.2   Primary Coolant Source Term

The primary coolant source term for ANO-2 is derived from plant chemistry data. RCS activity levels for various isotopes were determined from actual plant samples collected at periodic intervals. The plant data readings for each isotope were averaged and adjusted to achieve the TS limit of 1.0 µCi/gm DEI, using the TS definition of DEI and dose conversion factors (DCFs) for individual isotopes from ICRP 30, which are equivalent to the rounded thyroid values from FGR-11 for iodine isotopes. The non-iodine species were adjusted to achieve the previous ANO-2 TS limit of 100/E-bar µCi/gm of gross activity.

The DCFs for inhalation and submersion are taken from FGR-11 and FGR-12, respectively. When adjusting the primary coolant isotopic concentrations to achieve TS limits, the relative concentrations of fission products in the primary coolant system are assumed to remain constant. The final adjusted primary coolant source term is presented in Table 1.7.2-1, "Primary Coolant Source Term." Conservatively, assuming the maximum RCS mass and using the FGR-12 DCFs, the noble gas species in Table 1.7.2-1 translate into a DEX value of > 3100 µCi/gm, which supports this license amendment request.

1.7.3   Secondary Coolant Source Term

Secondary coolant system activity is limited to the TS limit of 0.10 µCi/gm DEI. Noble gases entering the secondary coolant system are assumed to be immediately released; that is, the noble gas activity concentration in the secondary coolant system is assumed to be 0.0 µCi/gm. Thus, the secondary coolant iodine activity is 1/10 of the activity given in Table 1.7.2-1.

The secondary source term is presented in Table 1.7.3-1, "Secondary Coolant Source Term."

1.7.4   Core Isotopic Inventory for Fuel Failure Events

Per Section 3.1 of RG 1.183, the inventory of fission products in the ANO-2 reactor core available for release to the containment are based on the maximum full power operation of the core and the approximate current licensed values for fuel enrichment and fuel burnup. The period of irradiation is selected to be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.

The ANO-2 reactor core consists of 177 fuel assemblies. The full core isotopic inventory is determined in accordance with RG 1.183, Regulatory Position 3.1, using the ORIGEN-S isotope generation and depletion computer code to develop the isotopics for the specified burnup, enrichment, and burnup rate (power level).

The assembly source term is based on 102% of rated (licensed) thermal power (3026 MWth x 1.02 = 3086.52 MWt). For non-LOCA events with fuel failures, except the FHA, a bounding radial peaking factor of 1.65 is then applied to conservatively simulate the effect of power level differences across the core that might affect the localized fuel failures for assemblies containing the peak fission product inventory. The FHA AST analysis assumes an even more conservative radial peaking factor of 1.7.

The core inventory release fractions for the gap release and early in-vessel damage phases for the design basis LOCAs were obtained from RG 1.183, Regulatory Position 3.2, Table 2, "PWR Core Inventory Fraction Released into Containment." Non-LOCA events that result in fuel damage, except the CREA, utilized RG 1.183, Regulatory Position 3.2, Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap." The CREA used the gap fractions identified in RG 1.183, Appendix H, which are larger than those listed in Table 3 for noble gases and iodines. However, the RG 1.183 Table 3 gap fraction for alkali metals was added for the CREA analysis. The CREA gap fractions are identified in Table 2.6-1.

The specific parameters used in the ORIGEN-S calculations are listed below:

Core Thermal Power	3086.52 MWt
Cycle Operating Time	520 EFPD
Number of Fuel Assemblies	177
Batch size	88
Average Initial Enrichment	4.0 w/o

Fuel Region	No. of Assemblies	MW per Assembly	Burnup (MWD)
Twice burned	1	16.86	26296
Once burned	88	17.44	18135
Feed batch	88	21.80	11334

A conservative nominal fuel assembly uranium loading of 4.0 w/o U-235 has been applied to all 177 fuel assemblies in the core and radioactive decay of fission products during refueling outages has been conservatively ignored. In addition, in order to address the potential variability of these input parameters for future fuel cycles, the calculated radioactive inventory includes an additional 4% margin for use in the AST analyses. A sensitivity study of the fuel enrichment using ORIGEN-ARP indicated that the added 4% margin bounds enrichments up to 4.6 w/o U-235. The ANO-2 TSs do not contain an enrichment limit, other than a limit on fuel storage in the spent fuel pool. However, the ANO-2 fuel cycle design (reload) processes limit fuel enrichment to 4.55 w/o U-235, plus 0.05 w/o uncertainty. Thus, the sensitivity results show the AST source term will continue to support the current reload analysis limit for enrichment. The final results, including this additional margin term, are provided in Table 1.7.4-1, "Core Isotopic Inventory for Fuel Failure Events."

Per Section 3.1 of RG 1.183, the source term methodology is similar for all events that result in fuel failure, except for DBA events that do not involve the entire core (i.e. non-LOCA events). The fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. The ANO-2 core contains 177 fuel assemblies, with 236 fuel rods in each. Thus, for the FHA, LRA, and CREA, the isotopic inventory in the damaged fuel is determined by adjusting (reducing) the total core inventory presented in Table 1.7.4-1 proportionally to the amount of fuel calculated to be damaged during the event. To account for differences in power level across the core, radial peaking factors are applied to these non-LOCA events, as discussed above, when determining the source term associated with the damaged fuel.

The FHA for ANO-2 is assumed to result in damage to two entire fuel assemblies, as discussed in Section 2.2. The CREA and LRA are assumed to result in failure of 14% of the core as discussed in Sections 2.5 and 2.6, respectively. Therefore, based on the methodology specified in RG 1.183, the FHA source term is derived by applying a factor of 1.7 x (2/177) to the core inventory of Table 1.7.4-1, and the CREA and LRA source terms are derived by applying a factor of 1.65 x 0.14 to the Table 1.7.4-1 core inventory.

1.8   Atmospheric Dispersion (X/Q) Factors

1.8.1   Onsite X/Q Determination

New X/Q factors for onsite release-receptor combinations have been developed using the ARCON-96 computer code (Reference 11). Different combinations are considered in order to provide the limiting release-receptor combination for the various events. Releases may occur from containment (LOCA, CREA, and FHA), the Main Steam Safety Valves (MSSVs) or upstream Atmospheric Dump Valves (ADVs) (MSLB, SGTR, LRA, and CREA), the penetration room ventilation system (PRVS) plenum (LOCA), or the fuel handling area ventilation (FHAV) plenum (FHA). Each of these release points is considered below.

Calculations of the new X/Q factors were completed in 2000. The calculations were performed to support power uprate and steam generator replacement dose calculations for ANO-2. The current licensing basis for ANO-2 only contains the onsite X/Q factors that were required to complete the necessary control room dose calculations to support these licensing activities. However, additional onsite X/Q factors associated with all of the various, potential post-accident release points discussed below were developed as part of the calculations completed in 2000.

Therefore, the calculation of these new, additional onsite ANO-2 X/Q factors used the same methodology and the same meteorological data as was used in calculating currently approved ANO-2 onsite X/Q factors. Information regarding the current ANO-2 dose calculations was submitted to the NRC in References 19 and 20 to support a power uprate amendment request. The ANO-2 power uprate was subsequently approved by the NRC in April 2002 (TS Amendment 244).

In addition, it is noted that the X/Q calculations completed in 2000 included development of new ANO-1 onsite X/Q factors using the same methodology and the same meteorological data as was used for ANO-2's factors. The new ANO-1 onsite X/Q factors were then utilized during performance of new AST analyses for that unit. Entergy submitted a license amendment request to use AST on ANO-1 on October 22, 2007 (Reference 24).

In the course of this calculation, several conservative assumptions were made about the operation of the CREVS and the location of the leakage. These assumptions are described in detail below.

1. The CREVS intake penthouse VPH-1 is centered approximately 10' $3^3/_4$" west of the auxiliary building support column designator "5.9" and 79' $7\frac{1}{2}$" south of the auxiliary building support column designator "A". The base of VPH-1 has an elevation of 447' $10^{15}/_{16}$".

2. The CREVS intake penthouse VPH-2 is centered approximately 10' $3^3/_4$" west of the auxiliary building support column designator "5.9" and 13' 6" south of the auxiliary building support column designator "E". The base of VPH-2 has an elevation of 448' 0".

3. During a MSLB accident, it is postulated that steam will be released with such force that the normal ventilation system will not be capable to control the release. It is therefore assumed that the walls and/or roof of the steam pipe area (Room 2155) will be damaged and allow direct release of all effluents to the environment. This is the worst possible scenario. It is unlikely that all the walls and roof will be completely removed, but it is possible that they will be dislodged from their fastened position. Since the path of the release is from the steam line to the room and then to the environment through the walls and/or roof, the source is conservatively modeled as a horizontal vent release with zero vertical velocity.

4. The stability index for meteorological data is calculated using the methods that are discussed in Reference 3.

5. The MSSVs are postulated to release steam in several accident scenarios. Depending on the accident conditions, as few as two or as many as all ten MSSVs may release steam. To determine X/Q values that would be conservative for use in all cases, instead of a diffuse release from ten valves, it is assumed that the release is from the valve with the lowest pressure setpoint that is closest to the control room intakes.

6. The ADVs are manually controlled by the operators to effect a plant cooldown following various accident scenarios. The dishcarge flow rate varies depending on the steam line pressure. As with the MSSVs, it is assumed that any ADV release is from the valve closest to the control room intakes.

7. Ground level is 354' 0". The height of the containment building is taken as the elevation of the parapet, which is 533' 6".

Site-specific meteorological (MET) data are obtained from the MET tower, which is located approximately 0.51 mile due east of the ANO-2 containment building at an elevation of 360 feet above sea level. The tower collects data at 10 and 57 meters above ground level.

ARCON-96 requires several data inputs to accurately calculate X/Qs. The first of these is the MET data itself, which were obtained from January 1995 to December 1999 and include wind speed and direction for both the 10- and 57-meter heights. A stability index ranging from 1 to 7 that identifies the apparent atmospheric turbulence for each hour of the day over the stated period is also required and was calculated per Assumption 4 above.

The receptor for all cases considered is one of the CREVS intakes, VPH-1 or VPH-2. The configuration of each CREVS intake in relation to the potential points of release is shown in Figures 1.8.1-1 through 1.8.1-7. Information regarding calculation of the X/Q for each configuration follows.

Release from the Atmospheric Dump Valves

The ADVs release steam directly to the atmosphere via exhaust pipes that are located on the auxiliary building as shown in Figure 1.8.1-1. The ADVs are uncapped exhausts located on the auxiliary building roof. The vent height is 30.56 meters above ground level. Since the release from these two stacks is dependent on which secondary loop is damaged, the worst-case release point is used, which is the ADV downstream of Steam Generator (SG) A (2CV-1001).

VPH-1 is 28.62 meters above ground level and is located at a distance of 79.77 meters from 2CV-1001. The direction to the source is approximately 346° relative to North (0°).

VPH-2 is 28.65 meters above ground level and is located at a distance of 92.22 meters from 2CV-1001. The direction to the source is 348°.

Release from the Main Steam Safety Valves

There are ten MSSVs that are designed to relieve pressure in the event of an over-pressurization of the secondary system. Five MSSVs are associated with each SG. The MSSVs are located on the ANO-2 auxiliary building roof as depicted in Figure 1.8.1-2 and Figure 1.8.1-3. Like the ADVs, the MSSVs are uncapped exhausts that vent directly to the auxiliary building roof. Typically, the release of effluents is even more violent than with the ADVs, because the MSSVs release steam only at high pressures. However, thermal and momentum plume height additions are neglected for conservatism. The MSSVs are located at a height of 30.66 meters above ground level.

The MSSVs are postulated to release steam in several accident scenarios. Depending on the accident conditions, as few as two or as many as all ten MSSVs may release steam. To determine X/Q values that would be conservative for use in all cases, the minimum vertical velocity and vent area was used. Therefore, instead of a diffuse release from ten valves, it is assumed that the release is from the closest MSSV with the lowest pressure setpoint. Thus the source is assumed to be located at 2PSV-1002. Since 2PSV-1002 exhausts to two 16" stacks within 5' of each other, the release is represented by a single virtual source.

The effective distance between the source and VPH-1 is approximately 69.23 meters and is in the direction 347°.

The distance between the source and VPH-2 is 81.73 meters and is in the direction 349°.

Release from a Main Steam Pipe

A MSLB outside containment on ANO-2 is assumed to be a double-ended guillotine break that releases steam directly to the environment. It is assumed that the south and east walls and the roof of Room 2155 will be damaged (effectively removed) during a MSLB, leaving large holes in their place. This room opens directly to the ANO-2 auxiliary building roof near the ANO-2 ADVs and the containment building.

Figure 1.8.1-4 shows the location of this room in relation to the CREVS intake structures, and Figure 1.8.1-5 provides the details of the walls that are assumed to be destroyed. Since this release may occur at virtually any point in this room, the release is assumed to be from the center of the nearest wall, which is the east wall. The vertical height of this release is 21.74 meters above ground level. Since the path of the release is from the steam line to the room and then to the environment through the walls and/or roof, the source is conservatively modeled as a horizontal vent release with zero vertical velocity.

The horizontal distance between the nearest Room 2155 wall and VPH-1 is 82.69 meters and the direction from VPH-1 is 350°.

VPH-2 is at a distance of 95.27 meters from the nearest wall of Room 2155. The relative direction towards the source from VPH-2 is 351°.

Release from the Fuel Handling Area Ventilation System

The ventilation air from the fuel handling floor radwaste area is exhausted out a flute that is mounted at azimuthal location 150° on the ANO-2 outer containment wall, where it is released at elevation 533' 3". The fuel handling area flute release height is 54.53 meters. This arrangement is shown in Figure 1.8.1-6.

The FHAV exhaust duct is located horizontally 56.5 meters from VPH-1 and at a relative direction of 332°.

VPH-2 is 68.01 meters from the FHAV exhaust duct. The relative direction to the duct is 337°.

Release from Penetration Room Ventilation System Exhaust

Although no credit is taken for filtration by the PRVS, this system could process and release postulated post-LOCA leakage through two parallel fans (2VEF-38A and 2VEF-38B). The fans exhaust through a pipe that travels up the outside of the containment wall as shown in Figure 1.8.1-7. The pipe is a hooked vent with no vertical velocity. The release height is 55.63 meters, at elevation 536' 6".

The PRVS exhaust outlet is located horizontally 59.5 meters from VPH-1 and at a relative direction of 341°.

VPH-2 is 71.7 meters from the PRVS exhaust outlet. The relative direction from VPH-2 to the outlet is 345°.

Release from Containment

In contrast to the other potential radiological release points, the containment release location is not precisely known. Because the source may actually be composed of many release locations, it is conservatively assumed that the release occurs at the shortest horizontal distance between the containment building surface and the control room intake. For the receptor-to-source direction, the direction from the control room intake to the center of the containment is used.

The containment is located horizontally 56.5 meters from VPH-1 and at a relative direction of 331°.

VPH-2 is 67.91 meters from the containment building. The relative direction from VPH-2 to the building center is 335°.

As previously stated, the ground level release type was used for all ANO-2 X/Q assessments. Vertical velocity, stack radius, and stack flow are not required for ground level release evaluations.

The only building that may affect wind diffusion is the containment, which has a cross-sectional area of 2205 m^2. This area is the licensing basis for ANO-2 and was used to calculate the offsite dose X/Q values. This value is slightly conservative with respect to the calculated cross-sectional area and, therefore, has been used in the ANO-2 onsite X/Q calculations. This area corresponds to the projected area of the outer surface of the containment building and is effectively the area that is exposed to a wind projecting on it from any given direction.

Table 1.8.1-1, "Onsite Atmospheric Dispersion Factors for ANO-2," provides the new ANO-2 onsite X/Q factors for the release-receptor combinations discussed above. These factors are not corrected for occupancy. This table summarizes the X/Q factors for the control room intakes and therefore provides the factors that were used for the ANO-2 AST control room dose analyses. The dose calculations use the worst-case X/Q factor for the applicable release point at each time step.

Table 1.8.1-2, "Release-Receptor Combination Parameters," provides information related to the relative elevations of the release-receptor combinations, the straight-line horizontal distance between the release point and the receptor location, and the direction (azimuth) from the receptor location to the release point (relative to true north).

1.8.2   Offsite X/Q Determination

New ANO-2 offsite atmospheric dispersion factors have not been utilized. The ANO-2 AST dose calculations continue to use the current licensing basis offsite X/Q factors. The factors for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) are presented in Table 1.8.2-1, "Offsite Atmospheric Dispersion Factors for ANO-2." The 0 to 2 hour EAB atmospheric dispersion factor is applied to all time periods in the analyses.

1.8.3   Meteorological Data

As discussed above, MET data over a five-year period (January 1995 through December 1999) were used in the development of the onsite X/Q factors used in the AST analyses. The ANO meteorological program complies with RG 1.23, "Onsite Meteorological Programs," with few exceptions, as discussed in Section 1.3.3 of the ANO-2 SAR. The onsite meteorological program itself is described in Section 2.3 of the ANO-2 SAR. For information purposes, the MET data (common to both ANO units) was forwarded to the NRC on compact disc in support of the NRC's review of the previous ANO-1 AST submittal on March 13, 2008 (1CAN030803).

ARCON-96 analyzes the meteorological data file used and lists the total number of hours of data processed and the number of hours of missing data in the case output. A MET data recovery rate may be determined from this information. Since all of the cases use the same MET data files, all of the cases in this analysis have the same data recovery rate. The ARCON-96 files present the number of hours of data processed as 43,824, and the number of missing data hours as 793. This yields a MET data recovery rate of 98.2%. No regulatory guidance is provided in RG 1.194 and NUREG/CR-6331 related to the valid MET data recovery rate required for use in determining onsite X/Q values. However, Regulatory Position C.5 of RG 1.23 requires a 90% data recovery threshold for measuring and capturing MET data. The 98.2% valid MET data rate for the cases in this analysis exceeds the 90% data recovery limit set forth by RG 1.23. With a data recovery rate of 98.2% and a total of five years worth of data, the contents of the MET data files are representative of the long-term meteorological trends at the ANO site.

2.0   Radiological Consequences - Event Analyses

2.1   <u>Loss of Coolant Accident (LOCA)</u>

2.1.1   Background

This event is assumed to be caused by an abrupt failure of a main reactor coolant pipe where the ECCS fails to prevent the core from experiencing significant degradation. This sequence cannot occur unless there are multiple failures and is, therefore, beyond the typical design basis accident that considers a single active failure. Activity is released into the containment building and then to the environment by means of containment leakage and leakage from the ECCS. This event is described in the Section 15.1.13 of the ANO-2 SAR.

2.1.2   Compliance with RG 1.183 Regulatory Positions

The revised LOCA dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident," as discussed below:

1.   Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided in Table 1.7.4-1. The core inventory release fractions for the gap release and early in-vessel damage phases of the LOCA are consistent with Regulatory Position 3.2 and Table 2 of RG 1.183.

2. Regulatory Position 2 – Per SAR Section 6.2.3.3.1.2, the long-term spray (sump) pH will be controlled at a value greater than 7.0 during recirculation. Therefore, the chemical form of the radioiodine released to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine. Prior to the onset of recirculation while the sump is filling, a buffering agent contained in baskets in the bottom of containment will be dissolving in the rising sump water. During this time and for a short period after the start of recirculation while additional mixing in the sump is occurring, the pH of the sump solution could be below 7.0. Calculations show the minimum sump pH at the start of recirculation could be as low as 6.874, but that a sump pH below 7 would exist for no longer than ~11 minutes. These calculations assume the existence of the TS minimum volume of buffering agent and maximum inventory of borated water at the maximum allowed boron concentration. Based on the calculation results, i.e. to account for a potential sump pH below 7.0, the LOCA dose analysis sets the spray removal coefficient at 0.0 for 11 minutes following the start of recirculation. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form.

3. Regulatory Position 3.1 -The activity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment. The release into the containment is assumed to terminate at the end of the early in-vessel phase.

4. Regulatory Position 3.2 - Reduction of the airborne radioactivity in the containment by natural deposition is not credited.

5. Regulatory Position 3.3 – Reduction in airborne radioactivity by containment spray is credited. Containment spray provides coverage to 78% of the containment. Therefore, the ANO-2 containment building atmosphere is not considered to be a single, well-mixed volume. A mixing rate of 11,880 cfm is assumed. This mixing rate is less than the two turnovers per hour of the unsprayed region that is allowed.

    Consistent with RG 1.183 and the Standard Review Plan (SRP), the elemental iodine spray removal coefficient is set to zero when a DF of 200 for elemental iodine is achieved at 3.6 hours. Likewise, the particulate spray removal coefficient is reduced by a factor of 10 when a DF of 50 is achieved for the aerosol at 3.25 hours. Although no maximum DF limit is defined for particulate iodine removal, a DF limit of 1000 is used. Thus, spray removal of particulates is terminated when a DF of 1000 is achieved at 13 hours. Containment spray is conservatively assumed to actuate at 180 seconds.

6. Regulatory Position 3.4 - Reduction in airborne radioactivity in the containment by filter recirculation systems is not assumed in this analysis.

7. Regulatory Position 3.5 - This position relates to suppression pool scrubbing in Boiling Water Reactors (BWRs), which is not applicable to ANO-2.

8. Regulatory Position 3.6 - This position relates to activity retention in ice condensers, which is not applicable to ANO-2.

9. Regulatory Position 3.7 - A containment leak rate of 0.1% per day of the containment air is assumed for the first 24 hours, consistent with the ANO-2 TS maximum allowable leak rate. After 24 hours, the containment leak rate is reduced to 0.05% per day of the containment air.

10. Regulatory Position 3.8 – This position relates to routine containment purging during power operations, which is not applicable to ANO-2.

11. Regulatory Positions 4.1 - 4.6 – All of Section 4 pertains to a dual containment which is not applicable to ANO-2. In addition, ANO-2 does not credit its penetration room ventilation system (PVRS).

12. Regulatory Position 5.1 – Engineered Safeguards Features (ESF) systems that recirculate water outside the primary containment are assumed to leak during their intended operation. With the exception of noble gases, all fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the containment sump water at the time of release from the core.

13. Regulatory Position 5.2 – ESF system leakage is assumed to be 4120 cc/hr, which is two times the leakage limit of 2060 cc/hr identified in SAR Section 15.1.13.4.1. No credit for filtration of this leakage by the PRVS is taken. The leakage is assumed to start at the time recirculation flow occurs in these systems and continue for the 30-day duration. The ECCS pumps have mini-flow return lines to the Refueling Water Tank (RWT), but each flow path is automatically isolated by redundant valves upon receipt of a recirculation actuation signal. Thus, there is no viable means of leakage of sump fluid to the RWT.

14. Regulatory Position 5.3 – With the exception of iodine, all radioactive materials in the recirculating fluid are assumed to be retained in the liquid phase.

15. Regulatory Position 5.4 – A flashing fraction of 2.29% was calculated based upon the sump temperature at the time of recirculation. However, the flashing fraction used in the analysis is limited based on Regulatory Position 5.5.

16. Regulatory Position 5.5 – Since the calculated flashing fraction is less than 10% (see previous item), the amount of iodine that becomes airborne is conservatively assumed to be 10% of the total iodine activity in the leaked fluid.

17. Regulatory Position 5.6 – For ECCS leakage into the auxiliary building, the form of the released iodine is 97% elemental and 3% organic. No reduction in release activity by dilution or holdup within buildings, or by any ventilation system, is credited.

18. Regulatory Position 6 – This position relates to MSSV leakage in BWRs, which is not applicable to ANO-2.

19. Regulatory Position 7 – Containment purge is not a combustible gas or pressure control measure for ANO-2.

2.1.3   Methodology

For the purposes of the LOCA analyses, a major LOCA is defined as a rupture of the RCS piping, including the double-ended rupture of the largest piping in the RCS, or of any line connected to the RCS up to the first closed valve, that results in fuel failure. Should a major break occur, the RCS will depressurize resulting in a reactor trip signal when the RCS low-pressure trip setpoint is reached. Safety injection will be subsequently actuated when the appropriate setpoint is reached. The following measures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat, and

2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

Release Inputs

The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided as Table 1.7.4-1. The source term accommodates an enveloping initial fuel enrichment.

From TS 6.5.16, the initial leakage rate from containment is 0.1% of the containment air per day. Per RG 1.183, Regulatory Position 3.7, the primary containment leakage rate is reduced by 50% at 24 hours into the LOCA event to 0.05%/day based on the post-LOCA primary containment pressure history. The entire leakage is a direct release to the environment, with no credit taken for cleanup of any portion of the leakage via the PRVS.

The ECCS leakage to the auxiliary building is assumed to be 4120 cc/hr based upon two times the current licensing basis value of 2060 cc/hr. The leakage is conservatively assumed to start at the time of switch to recirculation and continue throughout the 30-day period. This portion of the analysis assumes that 10% of the total iodine is released from the leaked fluid. The form of the released iodine is 97% elemental and 3% organic. Dilution and holdup of the ECCS leakage in the auxiliary building are not credited.

Transport Inputs

All of the containment leakage is assumed to be released directly to the environment as a ground level release. ECCS leakage into the auxiliary building is also modeled as an unfiltered, ground level release. The onsite X/Q factors for a diffuse release from the containment and for a release via the PRVS were compared and the larger factor of the two for each time interval was conservatively applied to all LOCA releases.

For this event, the control room ventilation system cycles through two modes of operation: normal and emergency. Initially, the ventilation system is assumed to be operating in its normal mode supplying 35,200 cfm of unfiltered, fresh, outside air.

After the start of the event, the control room is assumed to be isolated due to a high radiation signal. This signal may be initiated due to containment shine, shine from the approaching radioactive cloud or actual initial entry of radioactive material into the normal ventilation ductwork. A loss of offsite power would also initiate control room isolation. Control room isolation is designed to occur within 5 seconds, but a 10-second delay is assumed in the analysis. During isolation of normal control room ventilation, the CREVS automatically actuates and pressurizes the control room. After isolation of normal control room ventilation, 333 cfm of filtered, outside makeup air is assumed to be supplied by the CREVS. CREVS is also assumed to recirculate and filter 1667 cfm of control room air.

The CREVS filter efficiencies that are applied to the filtered makeup air are 99% for particulate, elemental and organic iodine and to the recirculation flow are 99% for particulate and 95% for elemental and organic iodine.

<u>LOCA Removal Inputs</u>

Containment spray provides coverage to 78% of the containment. Therefore, the ANO-2 containment atmosphere is not considered to be a single, well-mixed volume. A mixing rate of 11,880 cfm between the sprayed and unsprayed containment volumes is assumed. This mixing rate is less than the two turnovers per hour of the unsprayed region that is allowed by RG 1.183.

The elemental spray coefficient is initially assumed to be 20 hr^{-1} (limit per SRP 6.5.2), but is conservatively further reduced to 10 hr^{-1} at the start of sump recirculation. This coefficient is then reduced to 0 when an elemental DF of 200 is reached. Based upon the assumed elemental iodine removal rates, the DF of 200 is conservatively computed to occur at 3.6 hours.

The initial particulate iodine removal rate during injection is 3.97 hr^{-1}. The particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated particulate iodine removal rate during recirculation of 4.24 hr^{-1}, the DF of 50 is conservatively computed to occur at 3.25 hours. The particulate iodine removal rate is reduced to 0 when a DF of 1000 is reached at 13 hours.

2.1.5   Radiological Consequences

The control room X/Qs used for this event are based on a diffuse release from containment and a release via the PRVS. The LOCA X/Q input table was created by choosing the worst-case factor from these two release points for each time interval.

The EAB and LPZ dose consequences are determined using the X/Q factors provided in Table 1.8.2-1 for the appropriate time intervals. These factors are taken from the current ANO-2 licensing basis as reported in SAR Table 15.1.0-5.

The radiological consequences of the design basis LOCA are analyzed using the RADTRAD code and the inputs/assumptions previously discussed. In addition, the SCAP-II code is used to calculate direct shine doses to control room personnel due to the activity in the primary containment atmosphere.

The post accident doses are the result of two distinct activity releases:

1.   Containment leakage directly to atmosphere.

2.   ECCS system leakage into the auxiliary building.

The dose to the control room occupants includes terms for:

1.   Contamination of the control room atmosphere by intake and infiltration of radioactive material from the containment and ECCS leakage.

2. External radioactive plume shine contribution from the containment and ECCS leakage releases. This term takes credit for control room structural shielding.

3. A direct shine dose contribution from the activity contained in the containment. This term takes credit for both containment and control room structural shielding.

As shown in Table 2.1-4, the sum of the results of all dose contributions for EAB dose, LPZ dose, and control room dose are all within the appropriate regulatory acceptance criteria.

2.2   Fuel Handling Accident (FHA)

2.2.1   Background

This event consists of the drop of a single fuel assembly either in the fuel handling area or inside containment. The FHA is described in Section 15.1.23 of the ANO-2 SAR, which specifies that 60 fuel rods in a single fuel assembly are damaged during the event. Conservative re-analysis of fuel damage, with the additional weight of a grapple and Control Element Assembly (CEA) included, indicates that failure of two full assemblies, both the dropped assembly and the impacted assembly, is possible. Therefore, the ANO-2 FHA AST analysis assumes failure of two full irradiated fuel assemblies (472 fuel rods).

This analysis considers both a dropped fuel assembly inside the containment with the equipment hatch open and an assembly drop in the fuel handling area. In both cases, the release is assumed to occur directly to the environment without filtration. The source term released from the overlying water pool is the same for both the fuel handling area and the containment. RG 1.183 imposes the same 2-hour criteria for the direct unfiltered release of the activity to the environment for either location. Since the fuel handling area release X/Q is more limiting than the containment release X/Q, only the dropped fuel assembly in the fuel handling area case is actually analyzed. The results of that case are then bounding for the assembly drop in the containment case.

A minimum water level of 23 feet is maintained above the damaged fuel assemblies for both the containment and fuel handling area release locations.

2.2.2   Compliance with RG 1.183 Regulatory Positions

The FHA dose consequence analysis is consistent with the guidance provided in RG 1.183 Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," as discussed below:

1. Regulatory Position 1.1 – The amount of fuel damage is assumed to be 472 fuel rods, or two full irradiated assemblies. This damage is based on a conservative re-analysis of the fuel damage possible as a result of this accident and represents the most limiting case. The weight of the dropped assembly included the additional weight of an attached grapple and inserted CEA.

2. Regulatory Position 1.2 – The fission product release from the breached fuel is based on Regulatory Position 3.2 of RG 1.183 and the conservative evaluation of the estimated number of fuel rods breached. Section 1.7 provides a discussion of how the FHA source term is developed. All activity is assumed to be released from the fuel rods

instantaneously. Xenons, kryptons, halogens, cesiums and rubidiums are all considered.
However, the alkali metals (cesium and rubidium) are assumed to be in a non-volatile form
and therefore retained in the water pool with no release to the environment (see Regulatory
Position 3 below).

3.   Regulatory Position 1.3 – The chemical form of radioiodine released from the damaged fuel
     into the spent fuel pool is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine,
     and 0.15% organic iodide. The CsI is assumed to completely dissociate in the spent fuel
     pool resulting in a final iodine distribution of 99.85% elemental iodine and 0.15% organic
     iodine.

4.   Regulatory Position 2 – A minimum water depth of 23 feet is maintained above the
     damaged fuel assemblies. Therefore, a DF of 286 is applied to the elemental iodine and a
     DF of 1 is applied to the organic iodine in order to provide an overall effective DF of 200 in
     accordance with this Regulatory Position. As a result, the breakdown of the iodine species
     above the surface of the water is 70% elemental and 30% organic. Use of a DF of 286 for
     elemental iodine is consistent with the guidance provided in NRC Regulatory Issue
     Summary (RIS) 2006-04, "Experience with Implementation of Alternate Source Terms,"
     which provides guidance for the use of 285 for elemental iodine.

5.   Regulatory Position 3 – All of the noble gas released is assumed to exit the pool without
     mitigation. All of the non-iodine particulate nuclides are assumed to be retained by the pool
     water.

6.   Regulatory Position 4.1 – The analysis models the release to the environment over a
     2-hour period.

7.   Regulatory Position 4.2 – No credit is taken for filtration of the release.

8.   Regulatory Position 4.3 – Since all activity is assumed to be released to the environment
     over a 2-hour period per Regulatory Position 4.1, the analysis effectively assumes no credit
     for dilution of the release.

9.   Regulatory Position 5.1 – The containment equipment hatch is assumed to be open at the
     time of the FHA.

10. Regulatory Position 5.2 – No automatic isolation of the containment is assumed for the
     FHA.

11. Regulatory Position 5.3 – The release is assumed to leak to the environment over a two-
     hour period.

12. Regulatory Position 5.4 – No ESF filtration of the containment release is credited.

13. Regulatory Position 5.5 – Although the calculation on which this submittal is based
     assumes a mixing volume of just 10% of the containment net free volume, the analysis
     effectively receives no credit for this mixing because all activity is assumed to be released
     to the environment over a 2-hour period per Regulatory Position 5.3.

2.2.3   Methodology

The input assumptions used in the dose consequence analysis of the FHA are provided in Table 2.2-1. It is assumed that the FHA occurs at 100 hours after shutdown of the reactor. 100% of the gap activity specified in Table 2.2-1 is assumed to be instantaneously released from 472 fuel rods (two full irradiated assemblies) into the fuel pool. A minimum water level of 23 feet is maintained above the damaged fuel for the duration of the event. 100% of the noble gas released from the damaged fuel is assumed to escape from the pool. All of the non-iodine particulates released from the damaged fuel are assumed to be retained by the pool. Iodine released from the damaged fuel is assumed to be composed of 99.85% elemental and 0.15% organic. All activity released from the pool is assumed to leak unfiltered to the environment over a two-hour period. No credit for mixing of the released activity in the containment or fuel pool area was taken.

The FHA source term meets the requirements of Regulatory Position 1 of Appendix B to RG 1.183. Section 1.7 discusses the development of the FHA source term, which is listed in Table 1.7.5-1. The analysis includes a decay time of 100 hours before the beginning of fuel movement. Since the source terms presented in Table 1.7.4-1 do not include this decay time, it is accounted for in the RADTRAD model.

For this event, the control room ventilation system cycles through two modes of operation, normal and emergency. Initially, the ventilation system is assumed to be operating in its normal mode supplying 35,200 cfm of unfiltered, fresh, outside air.

After the start of the event, the control room is assumed to be isolated due to a high radiation signal. This signal may be initiated due to shine from the approaching radioactive cloud or actual initial entry of radioactive material into the normal ventilation ductwork. Control room isolation is designed to occur within 5 seconds, but a 36-second delay is assumed in the analysis. During isolation of normal control room ventilation, the CREVS automatically actuates and pressurizes the control room. After isolation of normal control room ventilation, 333 cfm of filtered, outside makeup air is assumed to be supplied by the CREVS. CREVS is also assumed to recirculate and filter 1667 cfm of control room air.

The CREVS filter efficiencies that are applied to the filtered makeup air are 99% for particulate, elemental iodine, and organic iodine and to the recirculation flow are 99% for particulate and 95% for elemental iodine and organic iodine.

2.2.4   Radiological Consequences

The control room X/Q used for this event is based on the postulated release locations. The release point location is chosen that provides the largest calculated X/Q factor. For the FHA event, the release from fuel handling area to VPH-1 provides the maximum X/Q.

The EAB and LPZ dose consequences are determined using the X/Q factors provided in Table 1.8.2-1 for the appropriate time intervals.

The radiological consequences of the FHA are analyzed using the RADTRAD code and the inputs/assumptions previously discussed. As shown in Table 3-1, the results for EAB dose, LPZ dose, and control room dose are all within the appropriate regulatory acceptance criteria.

2.3   Main Steam Line Break (MSLB)

2.3.1   Background

This event consists of a double-ended break of one main steam line as described in ANO-2 SAR Section 15.1.14. The affected SG rapidly depressurizes and releases its initial contents. Plant cooldown is then achieved via the remaining, unaffected SG. No fuel failure is postulated to occur during this event. The break is assumed to occur in piping located outside containment which provides bounding dose results.

2.3.2   Compliance with RG 1.183 Regulatory Positions

The MSLB dose consequence analysis followed the guidance provided in RG 1.183, Appendix E, "Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident," as discussed below:

1.   Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1. No fuel damage is postulated to occur for the ANO-2 MSLB event.

2.   Regulatory Position 2 – Since no fuel damage is postulated to occur during an ANO-2 MSLB, the maximum coolant activity allowed by the proposed, revised TSs and the two cases of iodine spiking are assumed.

3.   Regulatory Position 3 - The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

4.   Regulatory Position 4 - Iodine releases from the faulted SG and the unaffected SG to the environment (or containment) are assumed to be 97% elemental and 3% organic.

5.   Regulatory Position 5.1 - The accident-induced primary-to-secondary leak rate of 1.0 gpm allowed by TS 6.5.9 is apportioned equally between the SGs (0.5 gpm per SG).

6.   Regulatory Position 5.2 - A cold water density of 62.4 lbm/ft^3 is conservatively used in converting volumetric leak rates to mass leak rates.

7.   Regulatory Position 5.3 - The primary-to-secondary leakage is assumed to continue on the unaffected SG until after the shutdown cooling (SDC) system has been placed in service and on the faulted SG until the temperature of the RCS is less than 212 °F.

8.   Regulatory Position 5.4 - All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

9.   Regulatory Position 5.5.1 - In the faulted SG, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. For the intact SG used for plant cooldown, a portion of the leakage is assumed to flash to vapor based on the thermodynamic conditions in the primary and secondary coolants. To address iodine transport for release from the intact SG, a flashing fraction model was developed and based on conservative calculations, the flashing fraction of the primary-to-secondary leakage was determined to be less than 0.05. For conservatism, the MSLB analysis assumes the flashing fraction is 0.05. The flashed portion of the leakage is assumed to be directly released from the RCS to the atmosphere with no partitioning in the

SG. The remaining portion (95%) of the primary-to-secondary leakage that is discharged as liquid to the intact SG is assumed to be mixed with the SG secondary side liquid inventory and released to the atmosphere with partitioning via steam releases from the bulk fluid in the SG. The SG tubes on the intact SG are assumed to be partially uncovered during the first hour while steaming on the MSSVs. Operator action to open the ADVs is assumed to occur at one hour at the same time SG level is assumed to have recovered. However, no credit for total submergence of the tubes after the first hour has been taken, i.e. 5% flashing continues to be assumed for the duration of the cooldown.

10. Regulatory Position 5.5.2 – As stated above, total submergence of the tubes in the intact SG is expected to occur within one hour, but no credit for scrubbing of the assumed 5% flashed leakage has been taken.

11. Regulatory Position 5.5.3 – Primary-to secondary leakage into the intact steam generator that does not immediately flash is assumed to mix with the bulk water.

12. Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the intact SG is limited by the moisture carryover from the SG, which is 0.1%. Thus, the partition coefficient for alkali metals is 0.001.

13. Regulatory Position 5.6 – Partial uncovery of the tubes in the intact SG has essentially been postulated for the duration of the event. During this period, the 5% fraction of primary-to-secondary leakage that has been conservatively calculated to flash to vapor is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

2.3.3   Other Assumptions

1. The initial RCS activity is assumed to be at the TS limit of 1.0 µCi/gm DEI and the previous TS limit of 100/E-bar gross activity. Since the previous TS limit has been replaced with a DEX limit, a new, bounding DEX limit based on the actual noble gas activity assumed in this analysis is being submitted as part of this amendment request. The initial SG activity is assumed to be at the TS limit of 0.1 µCi/gm DEI.

2. The steam mass release rates from the intact SG are 3.245×10^6 gm/min from 0 to 2 hours, then 1.967×10^6 gm/min from 2 to 8 hours after event initiation. These values are based upon a cooldown rate of 75 °F/hr for the first two hours followed by a cooldown rate of 38.833 °F/hr until the RCS temperature is reduced to 212 °F at 8 hours, stopping further releases through the faulted SG. Although the intact SG could be isolated prior to 8 hours upon placing the SDC system in service, releases from the intact SG are conservatively assumed to continue for 8 hours.

3. Secondary releases from the intact SG are postulated to occur from the MSSVs for the first hour and from the ADVs from 1 to 8 hours, using the most limiting atmospheric dispersion factors for each release pathway. Secondary releases from the faulted SG are postulated to occur from main steam pipe using the most limiting atmospheric dispersion factors.

2.3.4   Methodology

Input assumptions used in the dose consequence analysis of the MSLB are provided in Table 2.3-1. The postulated accident is based upon a double-ended break of one main steam line outside of containment. Upon a MSLB, the affected SG rapidly depressurizes. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. Plant cooldown is achieved via the remaining, unaffected (intact) SG.

The analysis assumes that activity is released as reactor coolant enters the SGs due to primary-to-secondary leakage. All noble gases associated with this leakage are assumed to be released directly to the environment. All primary-to-secondary leakage into the faulted SG is also assumed to directly enter the atmosphere. Primary-to-secondary leakage is assumed to continue until the RCS temperature is reduced to 212 °F at 8 hours.

A portion (5%) of the primary-to-secondary leakage into the intact SG is flashed and released unfiltered to the atmosphere. The remaining activity (95%) is diluted by the contents of the SG and released via steaming through the ADVs (MSSVs during the first hour) until the RCS is cooled to 212 °F at 8 hours, which is lower than the SDC system initiation temperature. The SDC system then continues to remove decay heat without further releases to the environment.

The analysis also assumes that the initial iodine activity in the faulted SG is released directly to the environment. The entire contents of the faulted SG are released immediately (i.e. within one minute) without filtration. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 µCi/gm DE I-131 permitted by the TS and the faulted SG liquid mass is assumed to be at its maximum operating value of 2.72 x 10^7 gm. These release assumptions are consistent with the requirements of RG 1.183.

For this event, the control room ventilation system cycles through two modes of operation: normal and emergency. Initially, the ventilation system is assumed to be operating in its normal mode supplying 35,200 cfm of unfiltered, fresh, outside air.

After the start of the event, the control room is assumed to be isolated due to a high radiation signal. This signal may be initiated due to shine from the approaching radioactive cloud or actual initial entry of radioactive material into the normal ventilation ductwork. A loss of offsite power would also initiate control room isolation. Control room isolation is designed to occur within 5 seconds, but a 10-second delay is assumed in the analysis. During isolation of normal control room ventilation, the CREVS automatically actuates and pressurizes the control room. After isolation of normal control room ventilation, 333 cfm of filtered, outside makeup air is assumed to be supplied by the CREVS. CREVS is also assumed to recirculate and filter 1667 cfm of control room air.

The CREVS filter efficiencies that are applied to the filtered makeup air are 99% for particulate, elemental iodine, and organic iodine and to the recirculation flow are 99% for particulate and 95% for elemental iodine and organic iodine.

2.3.5   Radiological Consequences

The control room X/Qs used for this event are based on the postulated release locations. The release-receptor point locations are chosen to minimize the distance from the release point to the control room air intakes.

For the MSLB event, secondary releases from the intact SG are postulated to occur from the ADVs (MSSVs for the first hour) using the most limiting atmospheric dispersion factors. Secondary releases from the faulted SG are postulated to occur from main steam pipe using the most limiting atmospheric dispersion factors.

The EAB and LPZ dose consequences are determined using the X/Q factors provided Table 1.8.2-1 for the appropriate time intervals.

The radiological consequences of the MSLB Accident are analyzed using the RADTRAD code and the inputs/assumptions previously discussed. Two activity release cases corresponding to the RCS maximum pre-existing iodine spike and the accident-initiated iodine spike are analyzed. As shown in Table 3-1, the results of both cases for EAB dose, LPZ dose and control room dose are within the appropriate regulatory acceptance criteria.

2.4   Steam Generator Tube Rupture (SGTR)

2.4.1   Background

This event is assumed to be caused by the instantaneous rupture of a SG tube that relieves to the lower pressure secondary system. No melt or clad breach is postulated for the ANO-2 SGTR event. This event is described in Section 15.1.18 of the ANO-2 SAR.

Following a SGTR, the ANO-2 reactor will trip very early in the event. Since higher releases occur after the trip, the trip is assumed to occur at time zero (0.00 hours). A loss of offsite power (LOOP) is assumed to occur coincident with the reactor trip causing the MSSVs to open and provide steam relief. The release of activity via the ruptured SG and its associated MSSVs continues until the SG is assumed to be isolated at 60 minutes. At this point, only the intact SG is used to continue cooldown to SDC entry conditions and the release point then becomes the ADV with the worst X/Q values to the control room ventilation intakes. It is conservatively assumed that it will take 8 hours to reach SDC entry conditions and isolate the intact SG, thus terminating the release. Primary-to-secondary SG leakage is assumed to be at the ANO-2 TS maximum limit of 150 gpd per SG throughout the event.

2.4.2   Compliance with RG 1.183 Regulatory Positions

The revised SGTR dose consequence analysis follows the guidance provided in RG 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," as discussed below:

1.   Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1. No fuel damage is postulated to occur for the ANO-2 SGTR event.

2.   Regulatory Position 2 – Since no fuel damage is postulated to occur for the ANO-2 SGTR event, the maximum coolant activity allowed by the proposed, revised TSs and the two cases of iodine spiking are assumed.

3. Regulatory Position 2.1 - One case assumes a reactor transient prior to the postulated SGTR that raises the primary coolant iodine concentration to the maximum allowed by TS 3.4.8, 60.0 µCi/gm DE I-131. This is the pre-existing spike case.

4. Regulatory Position 2.2 – The other case assumes the transient associated with the SGTR causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the TS 3.4.8 value of 1.0 µCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-initiated spike case.

5. Regulatory Position 3 – No activity is assumed to be released from the fuel, since no fuel damage is postulated.

6. Regulatory Position 4 - Iodine releases from the SGs to the environment are assumed to be 100% elemental. Selection of different iodine fractions, e.g. 97% elemental and 3% organic, would not impact the overall results because the CREVS filter efficiencies are the same for both forms.

7. Regulatory Position 5.1 - The primary-to-secondary leak rate is 150 gpd per SG as specified by ANO-2 TS 3.4.6.2.

8. Regulatory Position 5.2 - A cold water density of 62.4 lbm/ft^3 is conservatively used in converting volumetric leak rates to mass leak rates.

9. Regulatory Position 5.3 - The primary-to-secondary leakage is assumed to continue until after the SDC system has been placed in service and both SGs have been isolated. The affected SG is conservatively assumed to be isolated by operator action after one hour. This isolation terminates releases from the affected SG, while primary-to-secondary leakage continues to provide activity for release from the unaffected SG via its ADV until SDC is placed in service, conservatively assumed to be at 8 hours.

10. Regulatory Position 5.4 - The release of activity from the secondary system is evaluated with the assumption of a LOOP at the time of reactor trip.

11. Regulatory Position 5.5 - All noble gases released from the primary system are assumed to be released to the environment without reduction or mitigation.

12. Regulatory Position 5.6 - Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the SGs is as follows:

    A portion of the primary-to-secondary leakage and ruptured tube flow is assumed to flash to vapor based on the thermodynamic conditions in the primary and secondary coolants. To address iodine transport for release from the ANO-2 SGs, a flashing fraction model was developed and a conservative upper bound flashing fraction of 0.05 was calculated. The flashed portion of the primary-to-secondary leakage and ruptured tube flow is assumed to be directly released from the RCS to the atmosphere with no partitioning in the SG. The remaining 95% of the activity in the primary-to-secondary leakage and ruptured tube flow that is discharged as liquid is assumed to be mixed with the SG secondary side liquid inventory and released to the atmosphere with partitioning via steam releases from the bulk

fluid in the SG. The SG tubes are assumed to be partially uncovered during the first hour while steaming on the MSSVs. Operator action to isolate the affected SG and open the ADV on the intact SG is assumed to occur at one hour at the same time SG level is conservatively assumed to have recovered. However, no credit for total submergence of the tubes after the first hour has been taken, i.e. 5% flashing continues to be assumed for the duration of the cooldown. The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs, which is a maximum of 0.06%. Thus, the partition coefficient for alkali metals is conservatively chosen to be 0.001.

2.4.3 Other Assumptions

1. The initial RCS activity is assumed to be at the TS limit of 1.0 µCi/gm DEI and the previous TS limit of 100/E-bar gross activity. Since the previous TS limit has been replaced with a DEX limit, a new, bounding DEX limit based on the actual noble gas activity assumed in this analysis is being submitted as part of this amendment request.

2. The steam mass release rates from the intact SG are 1.728 x10^6 gm/min from 0 to 1 hour (an additional 1.728 x10^6 gm/min is released from the affected SG during this time period), 3.456 x10^6 gm/min from 1 to 2 hours, and 1.956 x 10^6 gm/min from 2 to 8 hours after event initiation. These values are based upon a cooldown rate of 75 °F/hr for the first two hours followed by a cooldown rate of 32.5 °F/hr until the SDC system is placed in service and the intact SG is isolated, conservatively assumed to be at 8 hours.

3. Releases from the SGs are postulated to occur from the MSSVs for the first hour and from the ADVs from 1 to 720 hours, using the most limiting atmospheric dispersion factors for each release pathway.

2.4.4 Methodology

Input assumptions used in the dose consequence analysis of the SGTR event are provided in Table 2.4-1. This event is assumed to be caused by the instantaneous rupture of a SG tube releasing primary coolant to the lower pressure secondary system.

A LOOP at time zero (0.00 hours) is assumed to occur coincident with a reactor trip. The LOOP results in a loss of the condenser causing the MSSVs to open and provide steam relief. At 1 hour, the operators isolate the affected SG and initiate emergency cooldown of the RCS. At this point, only the intact SG is used to continue cooldown to SDC entry conditions and the release point then becomes the ADV with the worst X/Q values to the control room ventilation intakes. The analysis conservatively assumes that it will take 8 hours to initiate SDC and isolate the intact SG, thus terminating the release. Primary-to-secondary SG leakage is assumed to be at the ANO-2 TS maximum limit of 150 gpd per SG throughout the event.

A thermal-hydraulic analysis is performed to determine a conservative maximum break flow, break flashing flow, and steam release inventory through the faulted SG relief valves. Additional activity, based on the current TS limit for primary-to-secondary leakage, is released to the SG until the RCS is cooled to SDC system entry conditions.

No fuel failure is postulated for the SGTR event.

Consistent with RG 1.183 Appendix F, Regulatory Position 2, if no, or minimal, fuel damage is postulated for the limiting event, the activity release is assumed as the maximum allowed by TSs for two cases of iodine spiking: (1) maximum pre-existing iodine spike and (2) maximum accident-initiated, or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated SGTR event. The primary coolant iodine concentration is increased to the maximum value of 60 µCi/gm DE I-131 permitted by TS 3.4.8. Primary coolant is released into the ruptured SG by the tube rupture and by the allowable primary-to-secondary leakage.

Activity is released to the environment from the ruptured SG via direct flashing of a fraction of the released primary coolant from the tube rupture and also via steaming from the ruptured SG until the ruptured SG is isolated at 60 minutes. The intact SG is then used to continue the cooldown of the plant. Primary-to-secondary tube leakage is also postulated into the intact SG. Activity is released via steaming from the intact SG ADV until the RCS is cooled to SDC system entry conditions. These release assumptions are consistent with the requirements of RG 1.183.

For the case of the accident-initiated iodine spike, the postulated STGR event induces a concurrent iodine spike. The RCS activity is initially assumed to be 1.0 µCi/gm DE 1-131 as allowed by TS 3.4.8. Iodine is released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. All other release assumptions for this case are identical to those for the pre-existing spike case.

For this event, the control room ventilation system cycles through two modes of operation: normal and emergency. Initially, the ventilation system is assumed to be operating in its normal mode supplying 35,200 cfm of unfiltered, fresh, outside air.

The control room is conservatively assumed to be isolated 36 seconds after the beginning of the event. Control room isolation is initiated on LOOP and is designed to occur within 5 seconds of receipt of the isolation signal. With isolation of normal control room ventilation, the CREVS actuates and pressurizes the control room. After isolation of normal control room ventilation, 333 cfm of filtered, outside makeup air is assumed to be supplied by the CREVS. CREVS is also assumed to recirculate and filter 1667 cfm of control room air.

The CREVS filter efficiencies that are applied to the filtered makeup air are 99% for particulate, elemental iodine, and organic iodine and to the recirculation flow are 99% for particulate and 95% for elemental iodine and organic iodine.

2.4.5   Radiological Consequences

The control room X/Qs used for this event are based on the postulated release locations. The release-receptor point locations are chosen to minimize the distance from the release point to the control room air intakes. The releases from both the intact and faulted SGs are assumed to occur from the MSSVs using the worst-case X/Q values until the operator isolates the affected SG at 60 minutes. Releases during the remainder of the cooldown will be from the ADV with the worst-case X/Q value at each time step.

The EAB and LPZ dose consequences are determined using the X/Q factors provided Table 1.8.2-1 for the appropriate time intervals.

The radiological consequences of the SGTR accident are analyzed using the RADTRAD code and the inputs/assumptions previously discussed. Two activity release cases corresponding to the RCS maximum pre-existing iodine spike and the accident-initiated iodine spike are analyzed. As shown in Table 3-1, the radiological consequences of the ANO-2 SGTR event for EAB dose, LPZ dose, and control room dose are all within the appropriate regulatory acceptance criteria.

2.5   Locked Rotor Accident (LRA)

2.5.1   Background

This accident is postulated to occur as a consequence of an instantaneous seizure of a primary reactor coolant pump rotor. Reactor coolant flow is then rapidly reduced to the 3-pump value. Since a rapid reduction in coolant flow results in a rapid reduction in the margin to Departure from Nucleate Boiling (DNB), a low DNBR trip will occur. Fuel damage may occur as a result of this accident. This event is described in Section 15.1.5 of the ANO-2 SAR.

2.5.2   Compliance with RG 1.183 Regulatory Positions

The LRA dose consequence analysis followed the guidance provided in RG 1.183 Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," as discussed below:

1.   Regulatory Position 1 – The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 1.7.4-1. The inventory provided in Table 1.7.4-1 is then adjusted based on the RG 1.183 Table 3 gap fractions and the conservatively assumed number of fuel rods breached. A radial peaking factor of 1.65 is also applied. ANO-2 has processes in place to ensure the restrictions in RG 1.183, Footnote 11, are met.

2.   Regulatory Position 2 – Fuel damage is assumed for this event.

3.   Regulatory Position 3 – The activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the primary coolant.

4.   Regulatory Position 4 – The chemical form of radioiodine released from the damaged fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the SGs to the environment is assumed to be 97% elemental and 3% organic.

5.   Regulatory Position 5.1 – The primary-to-secondary leak rate is 300 gpd (150 gpd per SG as specified by ANO-2 TS 3.4.6.2). This leak rate is conservatively held constant throughout the event.

6.   Regulatory Position 5.2 – A cold water density of 62.4 lbm/ft^3 is conservatively used in converting volumetric leak rates to mass leak rates.

7. Regulatory Position 5.3 – Primary-to-secondary leakage and releases of radioactivity are assumed to continue until the SDC system has been placed in service and both SGs have been isolated. This isolation is conservatively assumed to be at 8 hours and terminates releases from the SGs.

8. Regulatory Position 5.4 – The LRA is evaluated assuming a coincident loss of offsite power.

9. Regulatory Position 5.5 – All noble gas radionuclides released to the primary system are assumed to be released directly to the environment without reduction or mitigation.

10. Regulatory Position 5.6 – Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the SGs is as follows:

    Both SGs are used for plant cooldown. A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the primary and secondary coolants. To address iodine transport for release from the ANO-2 SGs, a flashing fraction model was developed and a conservative upper bound flashing fraction of 0.05 was calculated. The flashed portion of the primary-to-secondary leakage is assumed to be directly released from the RCS to the atmosphere with no credit for partitioning in the SG. The remaining 95% of the activity in the primary-to-secondary leakage that is discharged as liquid is assumed to be mixed with the SG secondary side liquid inventory and released to the atmosphere with partitioning via steam releases from the bulk fluid in the SG. The SG tubes are assumed to be partially uncovered during the first hour. Operator action at 30 minutes to open the ADVs is assumed and SG level is conservatively assumed to recover within 1 hour of event initiation. However, no credit for total submergence of the tubes after the first hour has been taken, i.e. 5% flashing continues to be assumed for the duration of the cooldown. The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs, which is less than 1%. Thus, the partition coefficient for alkali metals is conservatively chosen to be 0.001.

2.5.3   Other Assumptions

1. The initial RCS and secondary activities are neglected due to the activity associated with the failed fuel being much larger in comparison.

2. The initial steam mass release rate is based upon a cooldown rate of 75 °F/hr for 2 hours. The cooldown is then conservatively continued at a rate of 32.5 °F/hr for 6 hours when the SDC system is assumed to be placed into service and the SGs isolated.

3. Following the LRA, 14% of the fuel is assumed to fail as a result of DNB.

4. All secondary releases are postulated to occur for the first 30 minutes from the MSSV with the most limiting atmospheric dispersion factors. After 30 minutes, operator action is assumed and all secondary releases are postulated to occur from the ADV with the most limiting atmospheric dispersion factors.

2.5.4   Methodology

Input assumptions used in the dose consequence analysis of the LRA are provided in
Table 2.5-1. This analysis assumes that 100% of the activity released from the damaged fuel is
completely dissolved in the primary coolant and is available for release to the secondary
system.

Primary coolant activity is assumed to be released into the SGs by leakage across the SG
tubes. The activity on the secondary side is then released via steaming from the MSSVs or
ADVs until the SDC system is assumed to be placed into service and the SGs isolated at
8 hours into the event. All noble gases associated with this leakage are assumed to be
released directly to the environment.

The LRA is evaluated with the assumption that 14% of the fuel experiences DNB. The activity
released from the damaged fuel corresponds to the requirements set out in Regulatory
Position 1 of Appendix G to RG 1.183. A radial peaking factor of 1.65 is applied in the
development of the source terms.

For this event, the control room ventilation system cycles through two modes of operation:
normal and emergency. Initially, the ventilation system is assumed to be operating in its normal
mode supplying 35,200 cfm of unfiltered, fresh, outside air.

After the start of the event, the control room is assumed to be isolated due to a high radiation
signal. This signal may be initiated due to containment shine, shine from the approaching
radioactive cloud, or actual initial entry of radioactive material into the normal ventilation
ductwork. A LOOP would also initiate control room isolation. Control room isolation is designed
to occur within 5 seconds, but a 36-second delay is assumed in the analysis. During isolation of
normal control room ventilation, the CREVS automatically actuates and pressurizes the control
room. After isolation of normal control room ventilation, 333 cfm of filtered, outside makeup air
is assumed to be supplied by the CREVS. CREVS is also assumed to recirculate and filter
1667 cfm of control room air.

The CREVS filter efficiencies that are applied to the filtered makeup air are 99% for particulate,
elemental iodine, and organic iodine and to the recirculation flow are 99% for particulate and
95% for elemental iodine and organic iodine.

2.5.5   Radiological Consequences

The control room X/Qs used for this event are based on the postulated release locations. The
release-receptor point locations are chosen to provide the largest calculated X/Q factors. For
the LRA, all releases are from the MSSVs or ADVs utilizing the most limiting atmospheric
dispersion factors for each release pathway.

The EAB and LPZ dose consequences are determined using the X/Q factors provided in
Table 1.8.2-1 for the appropriate time intervals.

The radiological consequences of the LRA are analyzed using the RADTRAD code and the
inputs/assumptions previously discussed. As shown in Table 3-1, the results for EAB dose,
LPZ dose, and control room dose are all within the appropriate regulatory acceptance criteria.

2.6   Control Rod Ejection Accident (CREA)

2.6.1   Background

This event consists of an uncontrolled withdrawal of a single control rod. The CREA results in a reactivity insertion that leads to a core power level increase and subsequent reactor trip. Following the reactor trip, plant cooldown is performed using steam release from the SG MSSVs and ADVs. Two CREA cases are considered. The first case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere. The second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system. This event is described in ANO-2 SAR Section 15.1.20.

2.6.2   Compliance with RG 1.183 Regulatory Positions

The CREA dose consequence analysis followed the guidance provided in RG 1.183 Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," as discussed below:

1.   Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 1.7.4-1. The inventory provided in Table 1.7.4-1 is adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The release fractions provided in RG 1.183 Table 3 are adjusted to comply with the specific RG 1.183 Appendix H release requirements. For both the containment and secondary release cases, 10% of the noble gas and iodine inventory and 12% of the alkali metal inventory is assumed to be in the fuel gap. No ANO-2 fuel experiences fuel centerline melt during a CREA.

2.   Regulatory Position 2 - Fuel damage is assumed for this event.

3.   Regulatory Position 3 - For the containment release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the containment atmosphere. For the secondary release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the primary coolant and be available for leakage to the secondary side of the SGs.

4.   Regulatory Position 4 - The chemical form of radioiodine released from the damaged fuel to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. No credit for containment spray actuation is taken.

5.   Regulatory Position 5 - The chemical form of radioiodine released from the SGs to the environment is assumed to be 97% elemental iodine, and 3% organic iodide.

6.   Regulatory Position 6.1 - For the containment leakage case, sedimentation of particulates in the containment is credited. Containment spray and PRVS are not credited in the CREA analysis.

7. Regulatory Position 6.2 - The containment is assumed to leak at the TS maximum allowable rate of 0.1% per day for the first 24 hours and 0.05% per day for the remainder of the event.

8. Regulatory Position 7.1 - The primary-to-secondary leak rate is 300 gpd (150 gpd per SG as specified by ANO-2 TS 3.4.6.2). This leak rate is conservatively held constant throughout the event.

9. Regulatory Position 7.2 – A cold water density of 62.4 lbm/ft^3 is conservatively used in converting volumetric leak rates to mass leak rates.

10. Regulatory Position 7.3 - All of the noble gas released to the secondary side is assumed to be released directly to the environment without reduction or mitigation.

11. Regulatory Position 7.4 - Regulatory Position 7.4 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the SGs is as follows:

    For the secondary release case, both SGs are used for plant cooldown. A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the primary and secondary coolants. To address iodine transport for release from the ANO-2 SGs, a flashing fraction model was developed and a conservative upper bound flashing fraction of 0.05 was calculated. The flashed portion of the primary-to-secondary leakage is assumed to be directly released from the RCS to the atmosphere with no credit for partitioning in the SG. The remaining 95% of the activity in the primary-to-secondary leakage that is discharged as liquid is assumed to be mixed with the SG secondary side liquid inventory and released to the atmosphere with partitioning via steam releases from the bulk fluid in the SG. The SG tubes are assumed to be partially uncovered during the first hour. Operator action at 30 minutes to open the ADVs is assumed and SG level is conservatively assumed to recover within 1 hour of event initiation. However, no credit for total submergence of the tubes after the first hour has been taken, i.e. 5% flashing continues to be assumed for the duration of the cooldown. The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs, which is less than 1%. Thus, the partition coefficient for alkali metals is conservatively chosen to be 0.001.

2.6.3   Other Assumptions

1. The initial RCS and secondary activities are neglected due to the activity associated with the failed fuel being much larger in comparison.

2. The initial steam mass release rate is based upon a cooldown rate of 75 °F/hr for 2 hours. The cooldown is then conservatively continued at a rate of 32.5 °F/hr for 6 hours when the SDC system is assumed to be placed into service and the SGs isolated.

3. Following the CREA, 14% of the fuel is assumed to fail as a result of DNB.

4. All secondary releases are postulated to occur for the first 30 minutes from the MSSV with the most limiting atmospheric dispersion factors. After 30 minutes, operator action is assumed and all secondary releases are postulated to occur from the ADV with the most limiting atmospheric dispersion factors.

5. The initial leakage rate from containment is 0.1% of the containment air per day. This leak rate is reduced by 50% after 24 hours to 0.05%/day. No credit for releases from the containment through the PRVS has been taken. All containment releases are released unfiltered to the environment via a ground level release.

6. For the release inside containment, sedimentation of particulates is credited. A conservative sedimentation coefficient of 0.1/hr is used until a DF of 1000 is reached at approximately 69 hours. Containment sprays are not credited.

2.6.4   Methodology

Input assumptions used in the dose consequence analysis of the CREA are provided in Table 2.6-1. The postulated accident consists of two cases. One case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere, and the second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system.

For the containment release case, 100% of the activity is released instantaneously to the containment. The releases from the containment are released unfiltered to the environment via a ground level release using the limiting containment release point X/Q. Sedimentation of particulates inside containment is credited. Removal of activity via containment spray is not credited.

For the secondary release case, primary coolant activity is released into the SGs by leakage across the SG tubes. The activity on the secondary side is then released via steaming from the MSSVs or ADVs until the SDC system is assumed to be placed into service and the SGs isolated at 8 hours into the event. All noble gases associated with this leakage are assumed to be released directly to the environment.

The CREA is evaluated with the assumption that 14% of the fuel experiences DNB. The activity released from the damaged fuel corresponds to the requirements set out in Regulatory Position 1of Appendix H to RG 1.183. A radial peaking factor of 1.65 is applied in the development of the source terms.

For this event, the control room ventilation system cycles through two modes of operation: normal and emergency. Initially, the ventilation system is assumed to be operating in its normal mode supplying 35,200 cfm of unfiltered, fresh, outside air.

After the start of the event, the control room is assumed to be isolated due to a high radiation signal. This signal may be initiated due to containment shine, shine from the approaching radioactive cloud or actual initial entry of radioactive material into the normal ventilation ductwork. A LOOP would also initiate control room isolation. Control room isolation is designed to occur within 5 seconds, but a 10-second delay is assumed in the analysis. During isolation of

normal control room ventilation, the CREVS automatically actuates and pressurizes the control room. After isolation of normal control room ventilation, 333 cfm of filtered, outside makeup air is assumed to be supplied by the CREVS. CREVS is also assumed to recirculate and filter 1667 cfm of control room air.

The CREVS filter efficiencies that are applied to the filtered makeup air are 99% for particulate, elemental iodine, and organic iodine and to the recirculation flow are 99% for particulate and 95% for elemental iodine and organic iodine.

2.6.5   Radiological Consequences

The control room atmospheric dispersion factors (X/Qs) used for this event are based on the postulated release locations. The release-receptor point locations are chosen to provide the largest calculated X/Q factors. For the CREA, all secondary releases are from the MSSVs or ADVs utilizing the most limiting X/Qs for each release pathway. X/Qs for containment releases are based on the worst-case containment release X/Qs.

The EAB and LPZ dose consequences are determined using the X/Q factors provided in Table 1.8.2-1 for the appropriate time intervals.

The radiological consequences of the CREA are analyzed using the RADTRAD code and the inputs/assumptions previously discussed. As shown in Table 3-1, the results of both cases for EAB dose, LPZ dose, and control room dose are all within the appropriate regulatory acceptance criteria.

2.7   Environmental Qualification (EQ)

RG 1.183, Regulatory Position 6, allows the licensee to use either the AST or TID-14844 assumptions for performing the required EQ analyses until such time as a generic issue related to the effect of increased cesium releases on EQ doses is resolved. The ANO-2 EQ analyses will continue to be based on TID-14844 assumptions.

3.0   Summary of Results

Results of the ANO-2 radiological consequence analyses using the AST methodology and the corresponding allowable control room unfiltered inleakage are summarized on Table 3-1.

4.0   Conclusion

Full implementation of the AST methodology, as defined in Regulatory Guide 1.183, into the ANO-2 design basis accident analyses has been made. Analysis of the dose consequences of the LOCA, FHA, MSLB, SGTR, LRA, and CREA have been made using the RG 1.183 methodology. The analyses used assumptions consistent with proposed changes in the ANO-2 licensing basis and the calculated doses do not exceed the defined acceptance criteria.

This report supports a maximum allowable control room unfiltered air inleakage of 250 cfm.
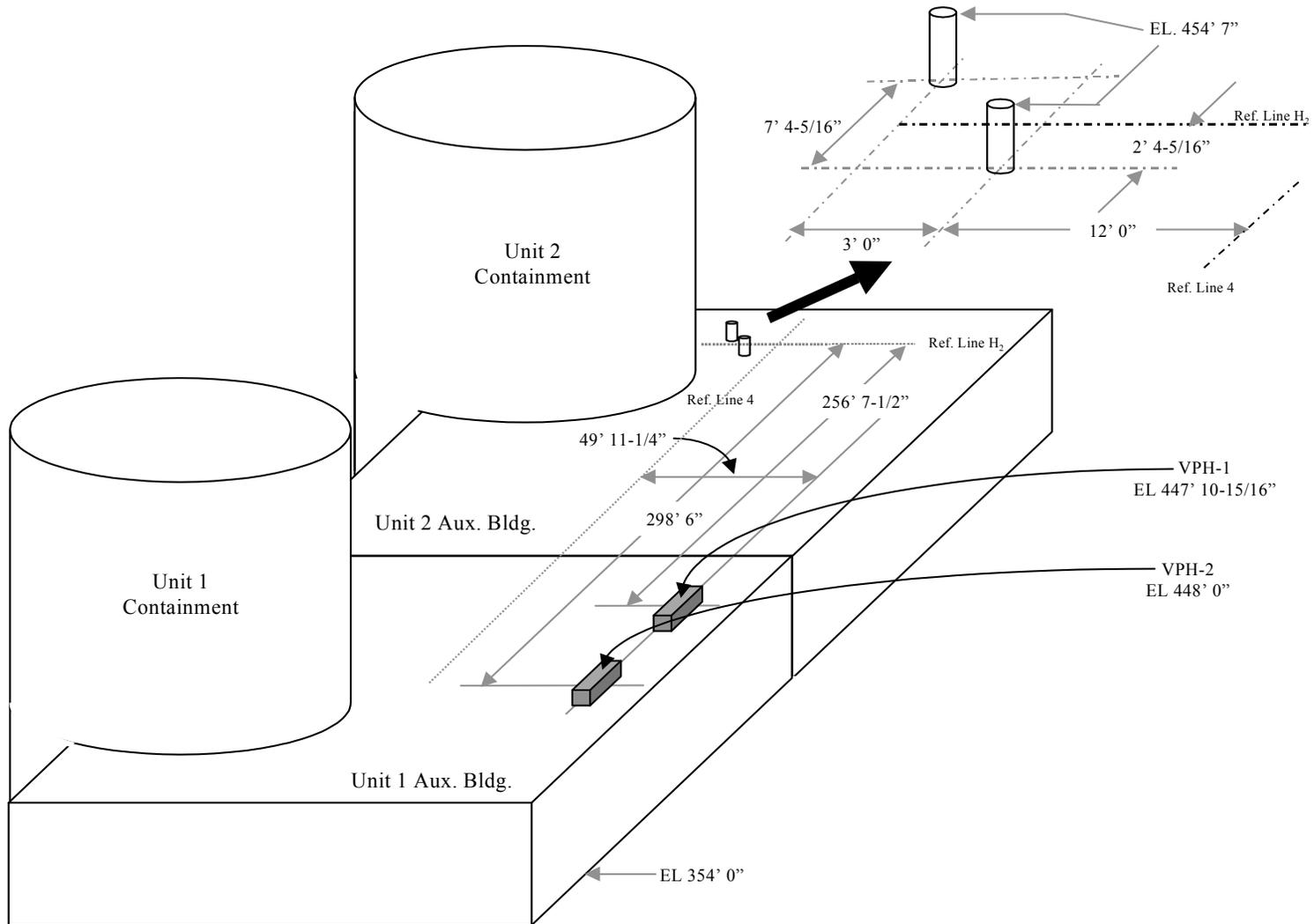
5.0   References

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20. Entergy letter 2CAN070103 to the NRC, "Radiological Dose Consequence Calculations to Support ANO-2 Power Uprate," dated July 3, 2001.

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Figure 1.8.1-1

Position of the ANO-2 ADVs in Relation to the Control Room Intake Structures VPH-1 and VPH-2

EL. 454' 7"

Ref. Line H₂

7' 4-5/16"

2' 4-5/16"

3' 0"

12' 0"

Ref. Line 4

Unit 2
Containment

Ref. Line H₂

Ref. Line 4

256' 7-1/2"

49' 11-1/4"

VPH-1
EL 447' 10-15/16"

Unit 2 Aux. Bldg.

298' 6"

VPH-2
EL 448' 0"

Unit 1
Containment

Unit 1 Aux. Bldg.

EL 354' 0"

(centerline of ANO-1 and ANO-2 auxiliary buildings runs north-south; ANO-2 is directly north of ANO-1)

Figure 1.8.1-2

Position of the ANO-2 MSSVs in Relation to the Control Room Intake Structures VPH-1 and VPH-2
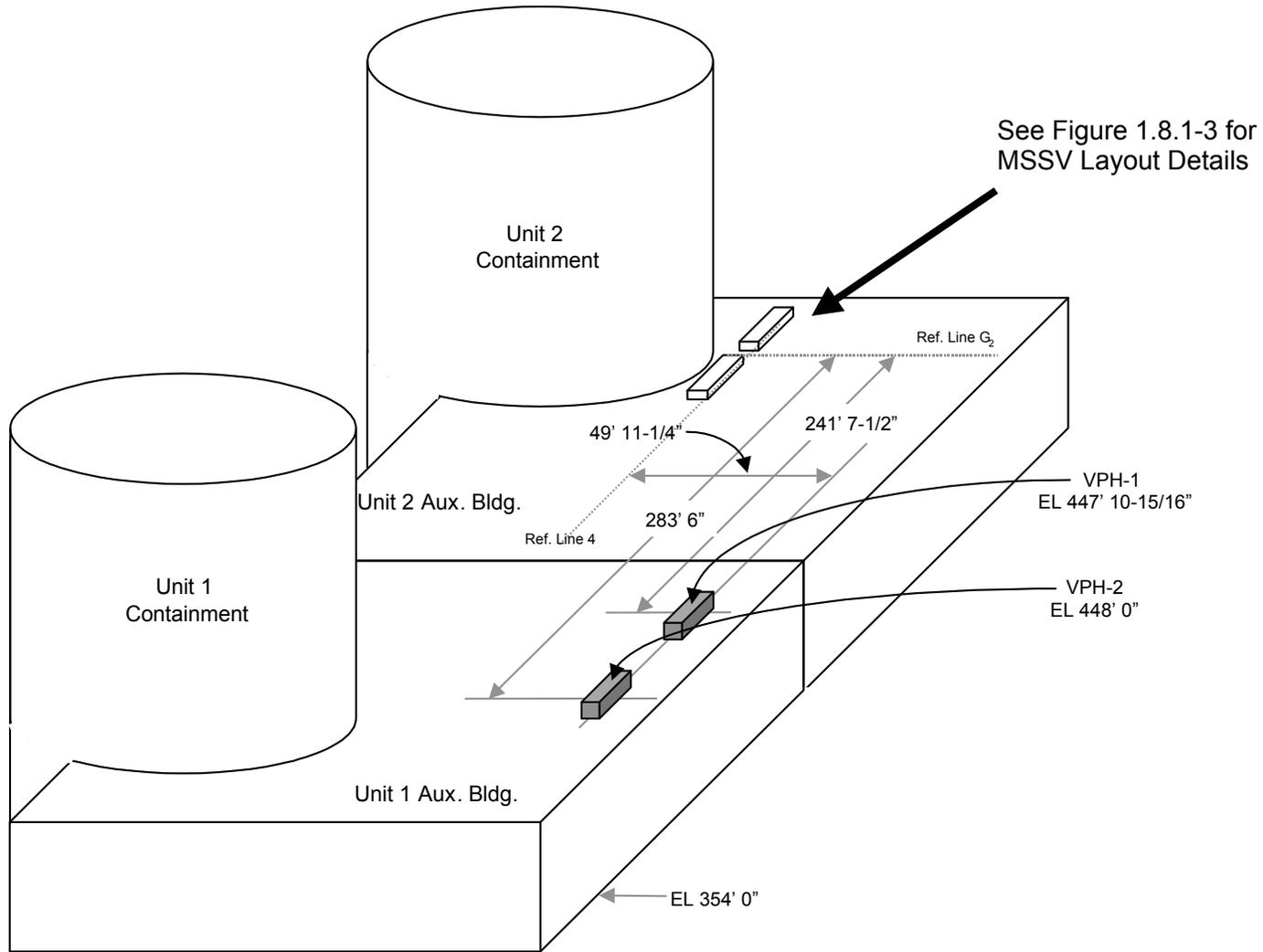
(centerline of ANO-1 and ANO-2 auxiliary buildings runs north-south; ANO-2 is directly north of ANO-1)

Figure 1.8.1-3

Layout of MSSV Exhausts Relative to Column Designators "G₂" and "4", used in Figure 1.8.1-2

Ref. Line (G₂)

4' 1-1/4"

4' 9" (typ)

60 deg (typ)

2PSV-1002

3' 4-7/8" (typ)

3' 6-3/16"

1' 11-5/8"
(typ)

Ref. Line (4)

(true north toward top of figure)

Figure 1.8.1-4

Position of the ANO-2 Main Steam Pipe Area (Room 2155) in Relation to the Control Room Intake Structures VPH-1 and VPH-2

North Wall Hole

See Figure 1.8.1-5 for Details of Room 2155

Roof Hole

East Wall Hole

Unit 2 Containment

EL 404' 0"

49' 11-1/4"

Ref. Line H

Ref. Line 4

256' 7-1/2"

VPH-1
EL 447' 10-15/16"

Unit 2 Aux Bldg

298' 6"

Unit 1 Containment

VPH-2
EL 448' 0"

Unit 1 Aux Bldg

EL 354' 0"

(centerline of ANO-1 and ANO-2 auxiliary buildings runs north-south; ANO-2 is directly north of ANO-1)

Figure 1.8.1-5

Details of the Wall Locations for the ANO-2 Main Steam Pipe Area (Room 2155)

1' 5"

Room 2155

17' 7"

Ref. Line (H$_2$)

1' 6"

Ref. Line (4)
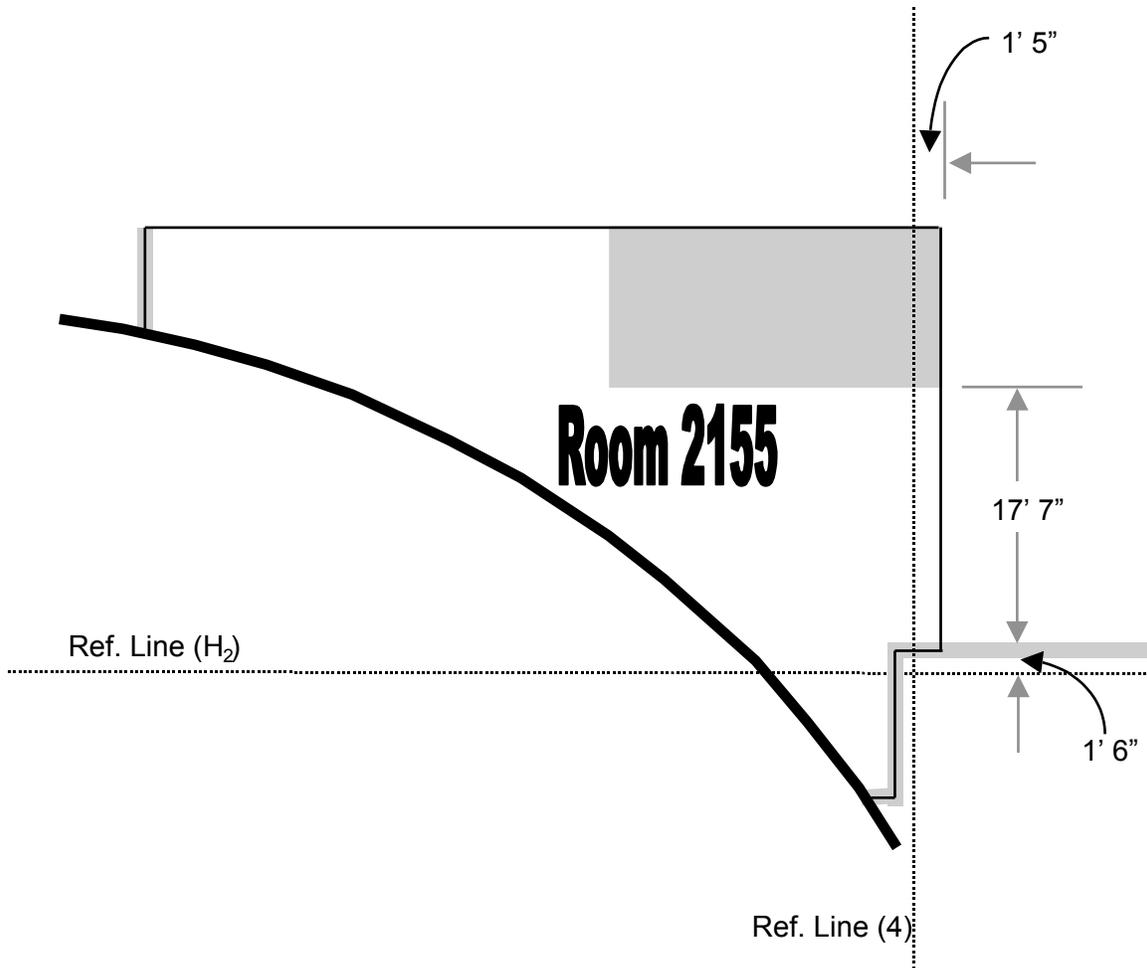
Figure 1.8.1-6

Arrangement of ANO-2 Fuel Handling Area Ventilation Exhaust Duct Relative to VPH-1 and VPH-2
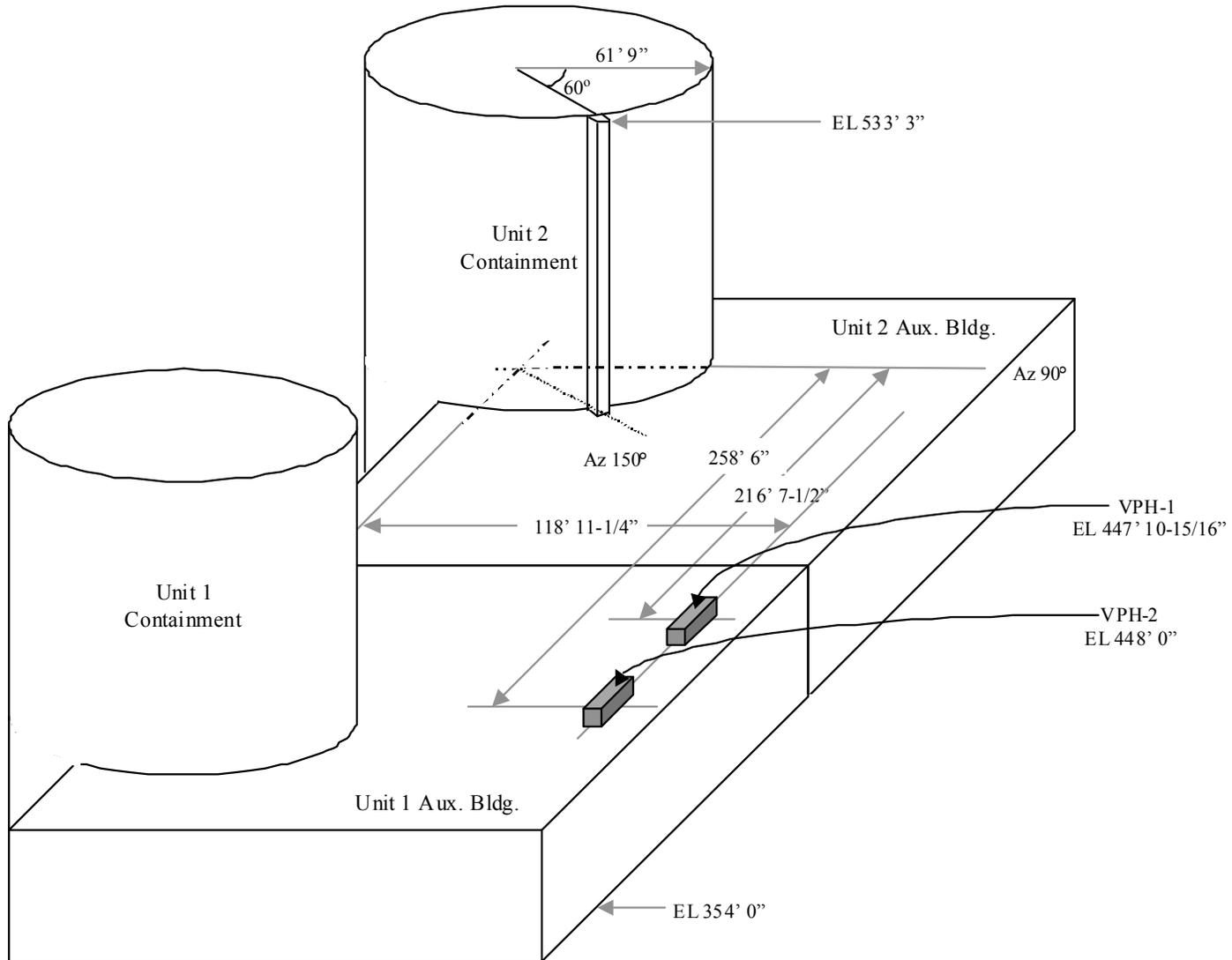
(centerline of ANO-1 and ANO-2 auxiliary buildings runs north-south; ANO-2 is directly north of ANO-1)

Figure 1.8.1-7

Arrangement of ANO-2 Containment and PRVS Exhaust Relative to VPH-1 and VPH-2
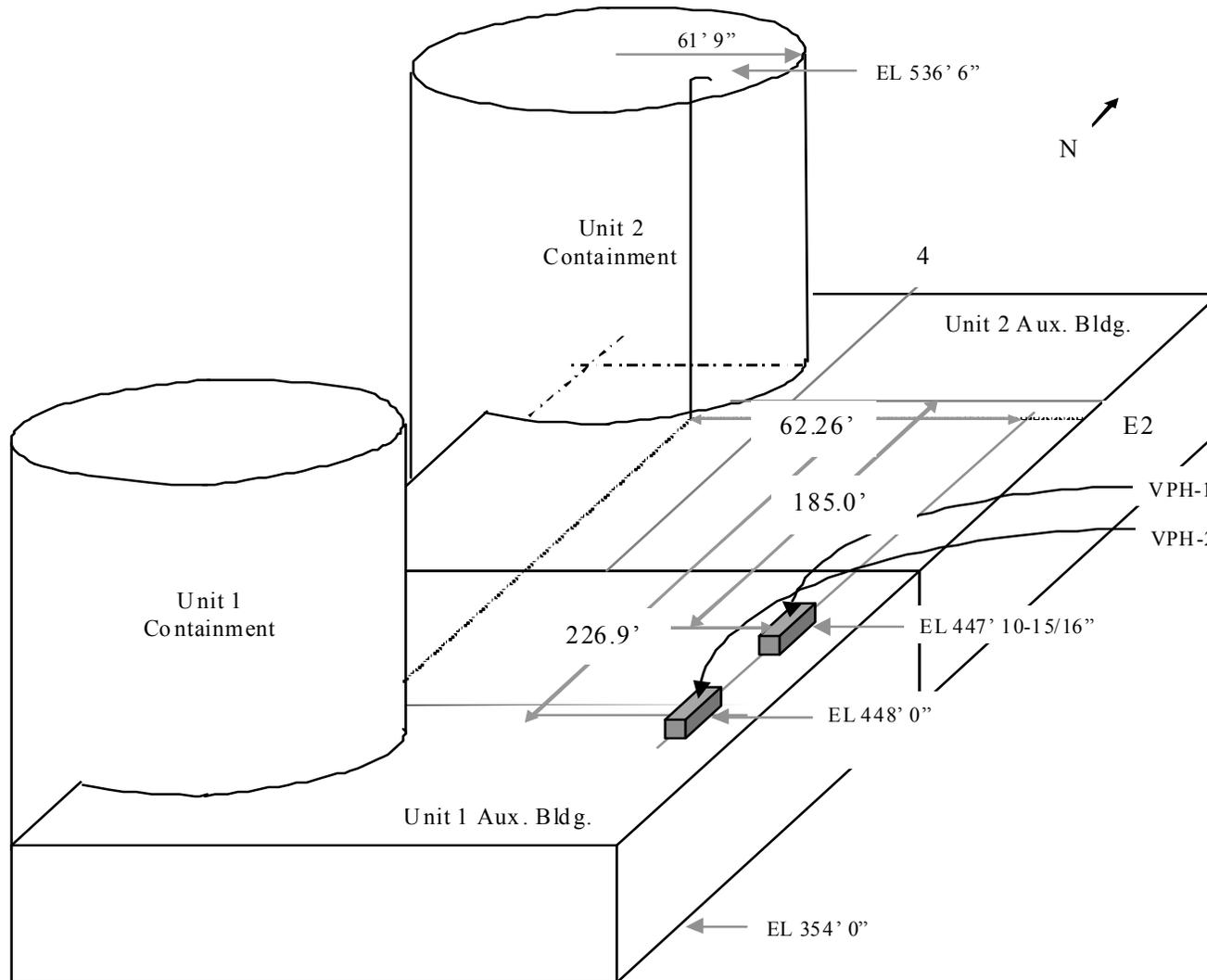
(centerline of ANO-1 and ANO-2 auxiliary buildings runs north-south; ANO-2 is directly north of ANO-1)

ACRONYMS FOR TABLES
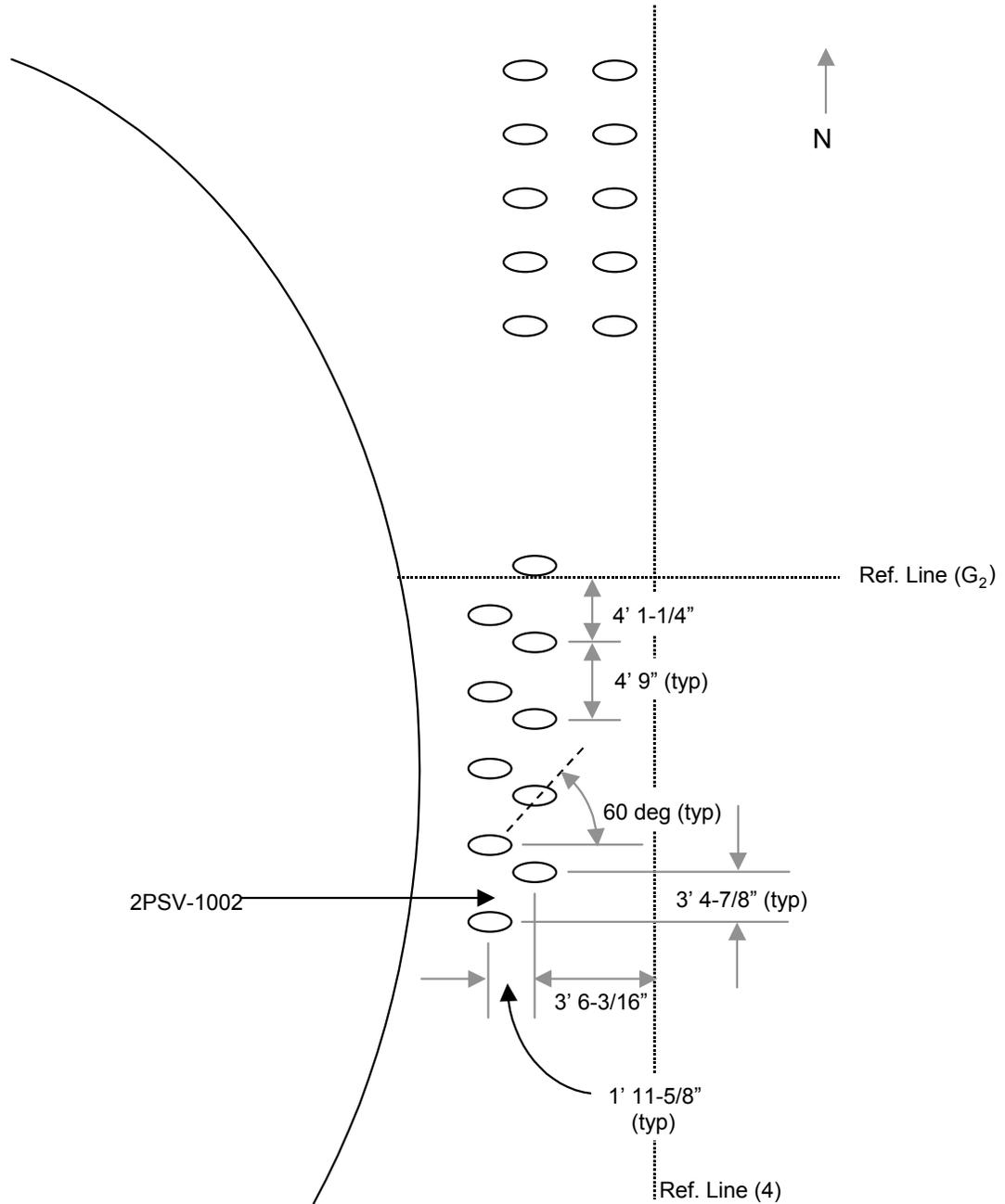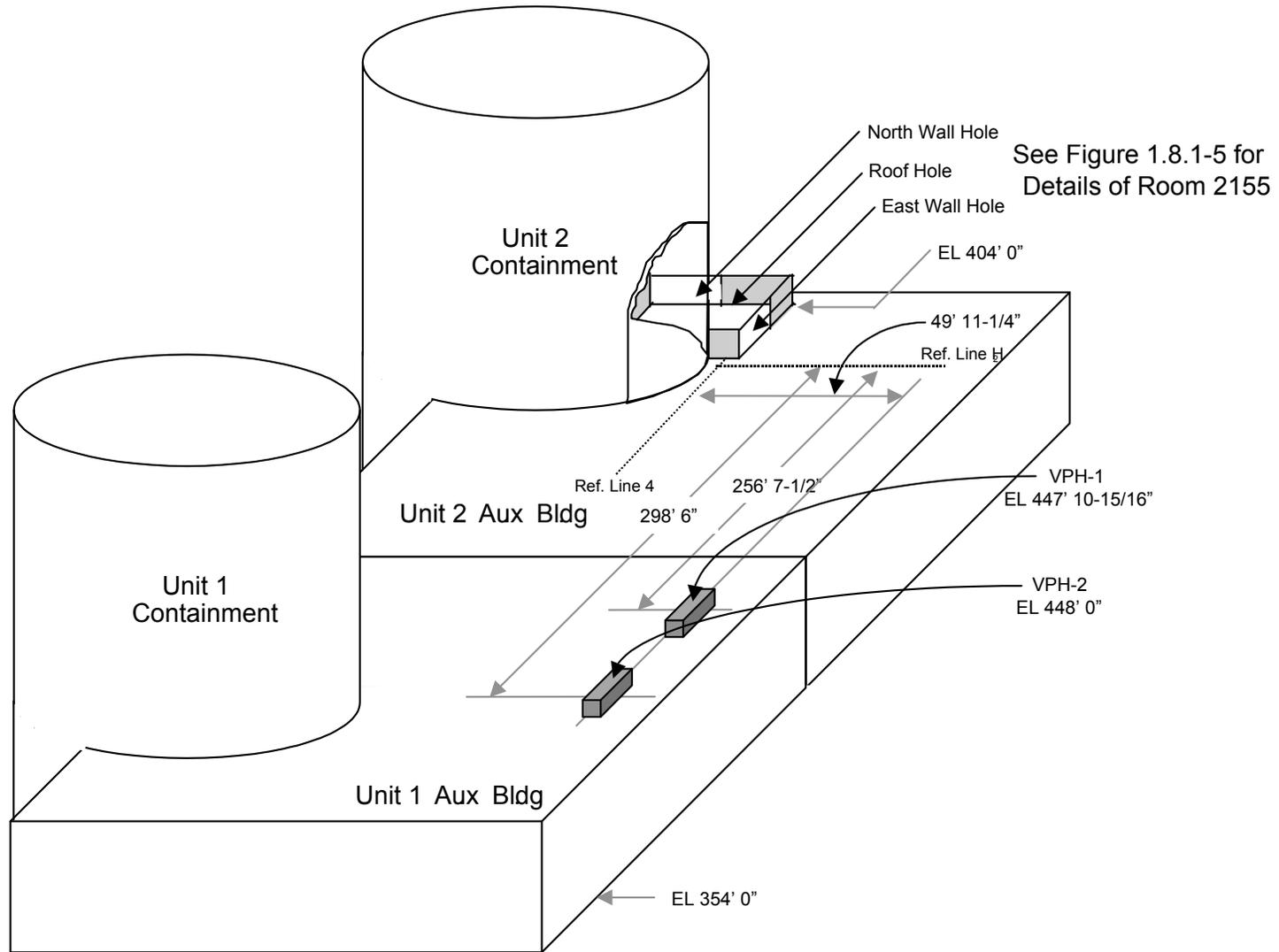
ADV   Atmospheric Dump Valve (Main Steam)

AOR   Analysis of Record

AST   Alternate Source Term Analyses

CLB   Current Licensing Basis

CR   Control Room

CREA   Control Rod Ejection Accident

DEX   Dose Equivalent Xenon-133

ESF   Engineered Safeguards Features

FHA   Fuel Handling Accident

LOCA   Loss of Coolant Accident

LRA   Locked Rotor Accident

MS   Main Steam

MSLB   Main Steam Line Break

MSSV   Main Steam Safety Valve

PRVS   Penetration Room Ventilation System

RG   Regulatory Guide

RP   Regulatory Position

SGTR   Steam Generator Tube Rupture

TS   Technical Specification

Table 1.6.3-1
Control Room Ventilation System Parameters
(Conforms to RG 1.183, Main Section, RP 4.2.4)

Parameter	Value	Basis
Control Room Volume	40,000 ft^3	Conservative internal calculation; Consistent with AOR
Normal Operation		
Filtered Make-up Flow Rate	0 cfm	Conservative assumption; Particulate filters exist
Filtered Recirculation Flow Rate	0 cfm	Conservative assumption; Particulate filters exist
Unfiltered Make-up Flow Rate	35,200 cfm	Total design flow rate for combined ANO-1&2 CR
Emergency Operation		
Filtered Make-up Flow Rate	333 cfm	Consistent with AOR and CLB
Filtered Recirculation Flow Rate	1667 cfm	Consistent with AOR and CLB
Unfiltered Make-up Flow Rate	0 cfm	System design does not support unfiltered makeup flow
Unfiltered Inleakage[1]	250 cfm	Consistent with proposed new TS limit
Filter Efficiencies		
Make-up Flow (two 2" filters):		
- Particulate	99%	Conservative assumption bounded by current TS 6.5.11
- Elemental	99%	Conservative assumption bounded by current TS 6.5.11
- Organic	99%	Conservative assumption bounded by current TS 6.5.11
Recirculation Flow (one 2" filter):		
- Particulate	99%	Conservative assumption bounded by current TS 6.5.11
- Elemental	95%	Conservative assumption bounded by current TS 6.5.11
- Organic	95%	Conservative assumption bounded by current TS 6.5.11

(1)   MSLB sensitivity review indicates 0 cfm unfiltered inleakage is limiting for pre-existing iodine spike case.

Table 1.7.2-1
Primary Coolant Source Term
(Conforms to RG 1.183, Appendix E, RP 2 and Appendix F, RP 2)

Nuclide	Nuclide Group	Activity (Ci)
Kr-85m	1 – Noble Gas	1.88E+03
Kr-85	1 – Noble Gas	9.23E+02
Kr-87	1 – Noble Gas	2.26E+03
Kr-88	1 – Noble Gas	3.80E+03
Xe-131m	1 – Noble Gas	4.80E+02
Xe-133m	1 – Noble Gas	1.31E+03
Xe-133	1 – Noble Gas	6.10E+04
Xe-135m	1 – Noble Gas	1.18E+03
Xe-135	1 – Noble Gas	1.35E+04
Xe-138	1 – Noble Gas	3.82E+03
I-130	2 – Iodine	7.65E+02
I-131	2 – Iodine	1.19E+02
I-132	2 – Iodine	3.12E+02
I-133	2 – Iodine	4.14E+02
I-134	2 – Iodine	4.81E+02
I-135	2 – Iodine	4.93E+02
Cs-134	3 – Alkali Metal	4.96E+02
Cs-136	3 – Alkali Metal	5.56E+01
Cs-137	3 – Alkali Metal	4.84E+02
Cs-138	3 – Alkali Metal	7.06E+03
Rb-86	3 – Alkali Metal	1.75E+02

Table 1.7.3-1
Secondary Coolant Source Term
(No RG 1.183 guidance; Values consistent with TS limit)

Nuclide	Nuclide Group	Activity (Ci)
I-130	2 – Iodine	5.60E+01
I-131	2 – Iodine	8.72E+00
I-132	2 – Iodine	2.28E+01
I-133	2 – Iodine	3.03E+01
I-134	2 – Iodine	3.52E+01
I-135	2 – Iodine	3.61E+01
Cs-134	3 – Alkali Metal	3.63E+01
Cs-136	3 – Alkali Metal	4.07E+00
Cs-137	3 – Alkali Metal	3.54E+01
Cs-138	3 – Alkali Metal	5.17E+02
Rb-86	3 – Alkali Metal	1.28E+01

Table 1.7.4-1
Core Isotopic Inventory for Fuel Failure Events
(Conforms to RG 1.183, Main Section, RPs 3.1 and 3.4)

Isotope	Core Inventory [Curies]	Isotope	Core Inventory [Curies]	Isotope	Core Inventory [Curies]
Kr-83m	1.032E+07	Sb-129	2.347E+07	Ce-141	1.429E+08
Kr-85	8.801E+05	Sb-131	6.632E+07	Ce-143	1.315E+08
Kr-85m	2.234E+07	Te-127	7.511E+06	Ce-144	1.093E+08
Kr-87	4.401E+07	Te-127m	1.250E+06	Np-239	1.621E+09
Kr-88	5.912E+07	Te-129	2.199E+07	Pu-238	1.451E+05
Xe-131m	9.144E+05	Te-129m	4.237E+06	Pu-239	2.411E+04
Xe-133	1.715E+08	Te-131	7.140E+07	Pu-240	3.669E+04
Xe-133m	5.381E+06	Te-131m	1.642E+07	Pu-241	8.901E+06
Xe-135	3.544E+07	Te-132	1.201E+08	Am-241	8.025E+03
Xe-135m	3.617E+07	Te-133	9.225E+07	Cm-242	2.157E+06
Xe-138	1.480E+08	Te-133m	8.217E+07	Cm-244	1.255E+05
I-130	1.459E+06	Sr-89	8.498E+07	La-140	1.552E+08
I-131	8.430E+07	Sr-90	6.786E+06	La-142	1.353E+08
I-132	1.229E+08	Sr-91	1.034E+08	Nb-95	1.500E+08
I-133	1.735E+08	Sr-92	1.106E+08	Nd-147	5.463E+07
I-134	1.950E+08	Ba-139	1.543E+08	Pr-143	1.300E+08
I-135	1.649E+08	Ba-140	1.500E+08	Y-90	7.079E+06
Cs-134	1.217E+07	Mo-99	1.575E+08	Y-91	1.108E+08
Cs-136	3.045E+06	Rh-105	8.327E+07	Y-92	1.120E+08
Cs-137	8.819E+06	Ru-103	1.304E+08	Y-93	1.261E+08
Cs-138	1.619E+08	Ru-105	8.854E+07	Zr-95	1.477E+08
Rb-86	1.389E+05	Ru-106	3.914E+07	Zr-97	1.442E+08
Sb-127	7.667E+06	Tc-99m	1.386E+08		

Table 1.8.1-1
Onsite Atmospheric Dispersion Factors for ANO-2
(Conforms to RG 1.183, Main Section, RP 5.3)

Time Period	Calculated X/Q Value	Basis
ADV Releases to VPH-1		
0 to 2 hrs	6.31×10^{-4} sec/m^3	New ARCON96 analysis
2 to 8 hrs	3.65×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.64×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	1.09×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	8.26×10^{-5} sec/m^3	New ARCON96 analysis
ADV Releases to VPH-2		
0 to 2 hrs	4.78×10^{-4} sec/m^3	New ARCON96 analysis
2 to 8 hrs	2.75×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.23×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	8.61×10^{-5} sec/m^3	New ARCON96 analysis
4 to 30 days	6.32×10^{-5} sec/m^3	New ARCON96 analysis
MSSV Releases to VPH-1		
0 to 2 hrs	8.05×10^{-4} sec/m^3	New ARCON96 analysis
2 to 8 hrs	4.64×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	2.08×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	1.42×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	1.06×10^{-4} sec/m^3	New ARCON96 analysis
MSSV Releases to VPH-2		
0 to 2 hrs	5.91×10^{-4} sec/m^3	New ARCON96 analysis
2 to 8 hrs	3.37×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.50×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	1.06×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	7.74×10^{-5} sec/m^3	New ARCON96 analysis
MS Pipe Release to VPH-1		
0 to 2 hrs	5.48×10^{-4} sec/m^3	New ARCON96 analysis
2 to 8 hrs	3.23×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.43×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	1.01×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	7.45×10^{-5} sec/m^3	New ARCON96 analysis

Table 1.8.1-1 (continued)

Time Period	Calculated X/Q Value	Basis
MS Pipe Release to VPH-2		
0 to 2 hrs	4.22×10^{-4} sec/m^3	New ARCON96 analysis
2 to 8 hrs	2.51×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.08×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	7.87×10^{-5} sec/m^3	New ARCON96 analysis
4 to 30 days	5.78×10^{-5} sec/m^3	New ARCON96 analysis
Fuel Handling Area Releases to VPH-1		
0 to 2 hrs	1.20×10^{-3} sec/m^3	Consistent with CLB
2 to 8 hrs	6.72×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	3.12×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	2.00×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	1.54×10^{-4} sec/m^3	New ARCON96 analysis
Fuel Handling Area Releases to VPH-2		
0 to 2 hrs	8.27×10^{-4} sec/m^3	New ARCON96 analysis
2 to 8 hrs	4.80×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	2.17×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	1.41×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	1.06×10^{-4} sec/m^3	New ARCON96 analysis
PRVS Releases to VPH-1		
0 to 2 hrs	9.77×10^{-4} sec/m^3	Consistent with CLB
2 to 8 hrs	5.76×10^{-4} sec/m^3	Consistent with CLB
8 to 24 hrs	2.56×10^{-4} sec/m^3	Consistent with CLB
1 to 4 days	1.68×10^{-4} sec/m^3	Consistent with CLB
4 to 30 days	1.25×10^{-4} sec/m^3	Consistent with CLB
PRVS Releases to VPH-2		
0 to 2 hrs	7.29×10^{-4} sec/m^3	New ARCON96 analysis
2 to 8 hrs	4.17×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.86×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	1.24×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	9.35×10^{-5} sec/m^3	New ARCON96 analysis

Table 1.8.1-1 (continued)

Time Period	Calculated X/Q Value	Basis
Containment Releases to VPH-1		
0 to 2 hrs	8.49×10^{-4} sec/m^3	New ARCON96 analysis
2 to 8 hrs	5.58×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	2.29×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	1.69×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	1.28×10^{-4} sec/m^3	New ARCON96 analysis
Containment Releases to VPH-2		
0 to 2 hrs	6.42×10^{-4} sec/m^3	New ARCON96 analysis
2 to 8 hrs	4.15×10^{-4} sec/m^3	New ARCON96 analysis
8 to 24 hrs	1.69×10^{-4} sec/m^3	New ARCON96 analysis
1 to 4 days	1.22×10^{-4} sec/m^3	New ARCON96 analysis
4 to 30 days	9.39×10^{-5} sec/m^3	New ARCON96 analysis

Table 1.8.1-2
Release-Receptor Combination Parameters
(Basis: Plant Design Drawings)

Release Point	Receptor Point	Release Height (m)	Receptor Height (m)	Distance (m)	Direction with respect to true north
ADV	N CR Intake	30.56	28.62	79.77	346
ADV	S CR Intake	30.56	28.65	92.22	348
MSSV	N CR Intake	30.66	28.62	69.23	347
MSSV	S CR Intake	30.66	28.65	81.73	349
Steam Pipe	N CR Intake	21.74	28.62	82.69	350
Steam Pipe	S CR Intake	21.74	28.65	95.27	351
Fuel Handling Area	N CR Intake	54.53	28.62	56.50	332
Fuel Handling Area	S CR Intake	54.53	28.65	68.01	337
PRVS Exhaust	N CR Intake	55.63	28.62	59.50	341
PRVS Exhaust	S CR Intake	55.63	28.65	71.70	345
Containment	N CR Intake	4.22[1]	28.62	56.50	331
Containment	S CR Intake	4.22[1]	28.65	67.91	335

(1) Initial vertical diffusion coefficient; containment assumed to be a diffuse area with a height of 83 feet (above the auxiliary building). The diffuse area width (containment diameter) is 123.5 feet.

Table 1.8.2-1
Offsite Atmospheric Dispersion Factors for ANO-2

Time Period	Licensed X/Q Value	Basis
Exclusion Area Boundary		
0 to 30 days	6.5×10^{-4} s/m^3	Conforms to RG 1.183, Main Section, RP 5.3; Consistent with CLB
Low Population Zone		
0 to 8 hrs	3.1×10^{-5} sec/m^3	Conforms to RG 1.183, Main Section, RP 5.3; Consistent with CLB
8 to 24 hrs	3.6×10^{-6} sec/m^3	Conforms to RG 1.183, Main Section, RP 5.3; Consistent with CLB
1 to 4 days	2.3×10^{-6} sec/m^3	Conforms to RG 1.183, Main Section, RP 5.3; Consistent with CLB
4 to 30 days	1.4×10^{-6} sec/m^3	Conforms to RG 1.183, Main Section, RP 5.3; Consistent with CLB

Table 2.1-1
LOCA Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 3026 MWt)	3086.52 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 3026 MWt
Fuel Enrichment	4.0 w/o	Conforms to RG 1.183, Main Section, RP 3.1
Margin Added to ORIGEN Source Term Results	4%	Conservative addition to accommodate potential future variations in fuel enrichment[1]
Core Fission Product Inventory	Table 1.7.4-1	Conforms to RG 1.183, Main Section, RPs 3.1 and 3.4
Gap Release Phase	30 sec – 0.5 hrs	Conforms to RG 1.183, Main Section, RP 3.3, Table 4
Early In-Vessel Release Phase	0.5 – 1.8 hrs	Conforms to RG 1.183, Main Section, RP 3.3, Table 4
Gap Release Fraction	0.05 for noble gases, halogens, and alkali metals only	Conforms to RG 1.183, Main Section, RP 3.2, Table 2
Early In-Vessel Release Fractions	0.95 noble gases 0.35 halogens 0.25 alkali metals 0.05 tellurium metals 0.02 strontium and barium 0.0025 noble metals 0.0005 cerium group 0.0002 lanthanides	Conforms to RG 1.183, Main Section, RP 3.2, Table 2
Iodine species distribution (%)	95.00 particulate 4.85 elemental 0.15 organic	Conforms to RG 1.183, Main Section, RP 3.5 and Appendix A, RP 2; Long-term sump pH controlled to value greater than 7
Containment Net Free Volume	1.78×10^6 ft^3	Conservative internal calculation
Containment Leak Rates	0.1%/day ≤ 24 hrs 0.05%/day > 24 hrs	Conforms to RG 1.183, Appendix A, RP 3.7

Table 2.1-1 (continued)

Parameter	Input Value	Basis
Unsprayed Containment Volume	3.92×10^5 ft^3	Conservative internal calculation
Sump Volume	62,898 ft^3	Conservative internal calculation
Sprayed Containment Volume	1.388×10^6 ft^3	Conservative internal calculation
Containment Sprayed Fractions	0.22 unsprayed 0.78 sprayed	Conservative internal calculation
Containment Mixing Rates	11,880 cfm unsprayed to sprayed 11,880 cfm sprayed to unsprayed	Bounds RG 1.183, Appendix A, RP 3.3; RG would allow 13,066 cfm
Spray Removal Rates Elemental	20 hr^{-1} during injection, 0.0 during first 11 minutes of recirculation, then 10 hr^{-1} until DF=200, then 0; DF=200 at 3.6 hours	Bounds RG 1.183, Appendix A, RP 3.3; Reduction during recirculation not required
Organic	No removal	Bounds RG 1.183, Appendix A, RP 3.3; Removal credit, same as particulate, allowed
Particulate	3.97 hr^{-1} during injection, 4.24 hr^{-1} during recirculation until DF=50 at 3.25 hours, then 0.424 hr^{-1} until DF=1000 at 13 hours, then 0	Bounds RG 1.183, Appendix A, RP 3.3; Reduction to 0 when DF = 1000 not required by RG
Spray Initiation Time (no termination)	180 sec	No termination conforms to RG 1.183, Appendix A, RP 5.2
Natural Deposition in Unsprayed Region	No credit taken	Conforms to RG 1.183, Appendix A, RP 3.2; RADTRAD used
Penetration Room Ventilation System	No credit taken	Consistent with CLB
Offsite X/Q	Table 1.8.2-1	Consistent with CLB

Table 2.1-1 (continued)

Parameter	Input Value	Basis
Offsite Breathing Rates	3.5E-4 m^3/sec 0-8 hrs 1.8E-4 m^3/sec 8-24 hrs 2.3E-4 m^3/sec > 24 hrs	Conforms to RG 1.183, Main Section, RP 4.1.3
CR X/Q (used limiting worst-case combined containment and PRVS exhaust release)	9.77E-4 s/m^3 0-2 hrs 5.76E-4 s/m^3 2-8 hrs 2.56E-4 s/m^3 8-24 hrs 1.69E-4 s/m^3 24-96 hrs 1.28E-4 s/m^3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
CR Breathing Rate	3.5E-4 m^3/sec 0-720 hrs	Conforms to RG 1.183, Main Section, RP 4.2.6
CR Occupancy Factors	100% 0-24 hrs 60% 24-96 hrs 40% > 96 hrs	Conforms to RG 1.183, Main Section, RP 4.2.6
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2
Time of CR Isolation	10 seconds	Conservative value assumed; System design = 5 sec
CR Unfiltered Inleakage	250 cfm	Consistent with proposed new TS limit

(1)   A sensitivity study of the fuel enrichment using ORIGEN-ARP showed that the added 4% margin produced results that bound enrichments up to 4.6 w/o U-235.

Table 2.1-1 (continued)

ESF Leakage Input for LOCA Analyses

Parameter	Input Value	Basis
ESF Leakage Rate	4120 cc/hr	Conforms to RG 1.183, Appendix A, RP 5.2
Fraction of Released Iodine in Sump Solution	1.0	Conforms to RG 1.183, Appendix A, RP 5.1
Iodine Species Distribution in Sump	0.97 elemental 0.03 organic	Conforms to RG 1.183, Appendix A, RP 5.6
Time to Recirculation	1560 sec (0.434 hr)	Conforms to RG 1.183, Appendix A, RP 5.2
Iodine Partition Coefficient for ESF Leakage (Flashing Fraction)	Calculated – 2.29% Used in analysis – 10%	Conforms to RG 1.183, Appendix A, RPs 5.4 and 5.5
Release Filtration Assumed	None	Bounds RG 1.183, Appendix A, RP 5.6

Table 2.1-2
LOCA Dose Results Summary (rem TEDE)

	EAB (worst 2-hour)	LPZ (30 days)	CR (250 cfm, 30 days)
Containment Leakage	8.91	0.618	1.676
ECCS Leakage	0.174	0.055	0.274
Cloud Shine	0	0	0.006
Containment Shine	0	0	0.106
Total TEDE Dose	9.084	0.673	2.062

Table 2.2-1
FHA Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 3026 MWt)	3087 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 3026 MWt
Average Fuel Enrichment	4.0 w/o	Conforms to RG 1.183, Main Section, RP 3.1
Margin Added to ORIGEN Source Term Results	4%	Conservative addition to accommodate potential future variations in fuel enrichment[1]
Peaking Factor	1.7	Consistent with CLB
Number of Fuel Assemblies in Core	177	Consistent with CLB
Number of Damaged Rods	472 (2 full assemblies)	Conforms to RG 1.183, Appendix B, RP 1.1; Conservative internal analysis
Fuel Rod Pressure Limit	1500 psig	Consistent with NRC SER on WCAP-16072-P (Reference 1)
Water Level Above Damaged Fuel	23 feet minimum	Conforms to RG 1.183, Appendix B, RP 2
Delay Before Fuel Movement	100 hrs	Consistent with CLB
Core Fission Product Inventory	Table 1.7.4-1	Conforms to RG 1.183, Main Section, RPs 3.1 and 3.4
Fission Product Gap Fractions (RG 1.183, Table 3)	0.08 I-131 0.10 Kr-85 0.05 other noble gases and iodines	Conforms to RG 1.183, Main Section, RP 3.2, Table 3, and RG 1.183, Appendix B, RP 1.2
Iodine Form in Pool Elemental Organic	99.85% 0.15%	Conforms to RG 1.183, Appendix B, RP 1.3
Iodine Form Above Pool Elemental Organic	70% 30%	Bounds RG 1.183, Appendix B, RP 2

Table 2.2-1 (continued)

Parameter	Input Value	Basis
Pool Decontamination Factors		
Elemental Iodine	286 (limited to provide overall DF = 200)	Bounds RG 1.183, Appendix B, RP 2
Organic Iodine and Noble Gases	1	Conforms to RG 1.183, Appendix B, RP 2
Offsite and CR Breathing Rate (duration of event)	3.5×10^{-4} m^3/s	Conforms to RG 1.183, Main Section, RPs 4.1.3 and 4.2.6
Offsite X/Q (duration of event)	Table 1.8.2-1	Consistent with CLB
Control Room X/Q (fuel handling area ventilation more limiting than containment)	1.20E-3 s/m^3 0-2 hrs 6.72E-4 s/m^3 2-8 hrs 3.12E-4 s/m^3 8-24 hrs 2.00E-4 s/m^3 24-96 hrs 1.54E-4 s/m^3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2
Time of CR Isolation	36 seconds	Conservative value assumed; System design = 5 sec
CR Unfiltered Inleakage	250 cfm	Consistent with proposed new TS limit

(1)   A sensitivity study of the fuel enrichment using ORIGEN-ARP showed that the added 4% margin produced results that bound enrichments up to 4.6 w/o U-235.

Table 2.3-1
MSLB Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 3026 MWt)	3086.52 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 3026 MWt
Initial Primary Coolant Activity	1.0 µCi/gm DE I-131 and 100/E-bar gross activity (Table 1.7.2-1)	Conforms to RG 1.183, Appendix E, RP 2; Gross activity consistent with proposed, revised TS DEX limit
Activity with Pre-existing Iodine Spike	60 µCi/g DE I-131	Conforms to RG 1.183, Appendix E, RP 2.1
Initial Secondary Coolant Activity	0.1 µCi/g I-131 (Table 1.7.3-1)	No guidance in RG 1.183; Value consistent with TS limit
Accident-Initiated Iodine Spike Factor	500	Conforms to RG 1.183, Appendix E, RP 2.2
Accident-Initiated Iodine Spike Duration	8 hrs	Conforms to RG 1.183, Appendix E, RP 2.2
Primary-to-Secondary Leak Rate	0.5 gpm/SG	Bounds RG 1.183, Appendix E, RP 5.1; TS 3.4.6.2 allows 150 gpd/SG
Time to Begin Cooldown (operator action)	60 min	No guidance in RG 1.183; Conservative value assumed
Time to Isolation of Intact SG (initiation of SDC) and to Reach 212 F (terminate steam release)	8 hrs	Conforms to RG 1.183, Appendix E, RP 5.3; Conservative value assumed consistent with CLB
Flashing Fraction in Intact SG	0.05	Conservative internal calculation based on flashing model; SG level recovered and tubes re-submerged in approximately one hour, but no credit taken
Partition Coefficient (faulted SG via flashing and vaporization; intact SG via flashing)	1.0	Conforms to RG 1.183, Appendix E, RP 5.5.1
Partition Coefficients (intact SG via steaming)	0.01 iodines 0.001 alkali metals	Conforms to RG 1.183, Appendix E, RP 5.5.4

Table 2.3-1 (continued)

Parameter	Input Value	Basis
RCS Mass	Maximum – 4.54 x 10^5 lbm Minimum – 4.32 x 10^5 lbm Maximum to produce largest equilibrium appearance rate; minimum to maximize activity concentration	Conservative internal calculation
SG Secondary Mass	Maximum - 1.936 x 10^5 lbm Minimum – 1.382 x 10^5 lbm Maximum in faulted SG to maximize release; minimum in intact SG to maximize activity concentration	Conservative internal calculation
Iodine Form of Secondary Release 　　Particulate 　　Elemental 　　Organic	 0% 97% 3%	Conforms to RG 1.183, Appendix E, RP 4
Offsite X/Q	Table 1.8.2-1	Consistent with CLB
Offsite Breathing Rates	3.5E-4 m^3/sec 0-8 hrs 1.8E-4 m^3/sec 8-24 hrs 2.3E-4 m^3/sec >24 hrs	Conforms to RG 1.183, Main Section, RP 4.1.3
CR X/Q (faulted SG – used worst-case main steam line)	5.48E-4 s/m^3 0-2 hrs 3.23E-4 s/m^3 2-8 hrs 1.43E-4 s/m^3 8-24 hrs 1.01E-4 s/m^3 24-96 hrs 7.45E-5 s/m^3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3

Table 2.3-1 (continued)

Parameter	Input Value	Basis
CR X/Q (intact SG – used worst-case MSSV for first 1 hr, then used worst-case ADV for each time step)	8.05E-4 s/m^3 0-1 hr 6.31E-4 s/m^3 1—2 hrs 3.65E-4 s/m^3 2-8 hrs 1.64E-4 s/m^3 8-24 hrs 1.09E-4 s/m^3 24—96 hrs 8.26E-5 s/m^3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
CR Breathing Rate	3.5E-4 m^3/sec 0-720 hrs	Conforms to RG 1.183, Main Section, RP 4.2.6
CR Occupancy Factors	100% 0-24 hrs 60% 24-96 hrs 40% > 96 hrs	Conforms to RG 1.183, Main Section, RP 4.2.6
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2
Time of CR Isolation	10 seconds	Conservative value assumed; System design = 5 sec
CR Unfiltered Inleakage (pre-existing iodine spike case)	0 cfm	Value determined to be most limiting for this event
CR Unfiltered Inleakage (accident-initiated iodine spike case)	250 cfm	Bounds proposed new TS limit

Table 2.4-1
SGTR Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 3026 MWt)	3086.52 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 3026 MWt
Initial Primary Coolant Activity	1.0 µCi/gm DE I-131 and 100/E-bar gross activity (Table 1.7.2-1)	Conforms to RG 1.183, Appendix F, RP 2; Gross activity consistent with proposed, revised TS DEX limit
Activity with Pre-existing Iodine Spike	60 µCi/g I-131	Conforms to RG 1.183, Appendix F, RP 2.1
Initial Secondary Coolant Activity	0.1 µCi/g I-131 (Table 1.7.3-1)	No guidance in RG 1.183; Value consistent with TS limit
Accident-Initiated Iodine Spike Factor	335	Conforms to RG 1.183, Appendix F, RP 2.2
Accident-Initiated Iodine Spike Duration	8 hrs	Conforms to RG 1.183, Appendix F, RP 2.2
Initial Ruptured SG Tube Leak Rate	240 gpm	Conservative internal calculation, consistent with AOR
Primary-to-Secondary Leak Rate	150 gpd per SG	Conforms to RG 1.183, Appendix F, RP 5.1
Time of Reactor Trip/LOOP	0 sec	Conservative value assumed; Conforms to RG 1.183, Appendix F, RP 5.4
Time to Isolation of Faulted SG (operator action)	60 min	No guidance in RG 1.183; Conservative value assumed
Time to Isolation of Intact SG (initiation of SDC)	8 hrs	Conforms to RG 1.183, Appendix F, RP 5.3; Conservative value assumed consistent with CLB
Flashing Fraction in SGs	0.05	Conservative internal calculation based on flashing model; SG level recovered and tubes re-submerged in approximately one hour, but no credit taken
Partition Coefficient (via flashing)	1.0	Conforms to RG 1.183, Appendix F, RP 5.6

Table 2.4-1 (continued)

Parameter	Input Value	Basis
Partition Coefficients (via steaming)	0.01 iodines 0.001 alkali metals	Conforms to RG 1.183, Appendix F, RP 5.6
RCS Mass	Maximum – 4.54×10^5 lbm Minimum – 4.32×10^5 lbm Maximum to produce largest equilibrium appearance rate; minimum to maximize activity concentration	Conservative internal calculation
SG Secondary Mass	1.382×10^5 lbm Minimum used to maximize activity concentration	Conservative internal calculation
Offsite X/Q	Table 1.8.2-1	Consistent with CLB
Offsite Breathing Rates	3.5E-4 m^3/sec 0-8 hrs 1.8E-4 m^3/sec 8-24 hrs 2.3E-4 m^3/sec > 24 hrs	Conforms to RG 1.183, Main Section, RP 4.1.3
CR X/Q (worst-case MSSV until operator action at 1 hr, then worst-case ADV for event duration)	8.05E-4 s/m^3 0-1 hrs 6.31E-4 s/m^3 1-2 hrs 3.65E-4 s/m^3 2-8 hrs 1.64E-4 s/m^3 8-24 hrs 1.09E-4 s/m^3 24-96 hrs 8.26E-5 s/m^3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
CR Breathing Rate	3.5E-4 m^3/sec 0-720 hrs	Conforms to RG 1.183, Main Section, RP 4.2.6
CR Occupancy Factors	100% 0-24 hrs 60% 24-96 hrs 40% > 96 hrs	Conforms to RG 1.183, Main Section, RP 4.2.6

Table 2.4-1 (continued)

Parameter	Input Value	Basis
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2
Time of CR Isolation	36 seconds	Conservative value assumed; System design = 5 sec
CR Unfiltered Inleakage	250 cfm	Bounds proposed new TS limit

Table 2.5-1
LRA Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 3026 MWt)	3086.52 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 3026 MWt
Fuel Enrichment	4.0 w/o U-235	Conforms to RG 1.183, Main Section, RP 3.1
Margin Added to ORIGEN Source Term Results	4%	Conservative addition to accommodate potential future variations in fuel enrichment[1]
Core Fission Product Inventory	Table 1.7.4-1	Conforms to RG 1.183, Main Section, RPs 3.1 and 3.4
Fuel Failure (rods in DNB)	14%	Consistent with CLB
Peaking Factor	1.65	Consistent with AOR
Fission Product Gap Fractions (RG 1.183, Table 3)	0.08 I-131 0.10 Kr-85 0.05 other noble gases and iodines 0.12 alkali metals	Conforms to RG 1.183, Appendix G, RP 1
Secondary Release Iodine Species Distribution	0% particulate 97% elemental 3% organic	Conforms to RG 1.183, Appendix G, RP 4
Primary-to-Secondary (P-S) Leak Rate	150 gpd/SG	Conforms to RG 1.183, Appendix G, RP 5.1
Duration of Steam Release (switch to SDC system)	8 hrs	Conforms to RG 1.183, Appendix G, RP 5.1; Conservative value assumed consistent with CLB
Flashing Fraction of P-S Leakage during Cooldown	0.05	Conservative internal calculation based on flashing model; SG level recovered and tubes re-submerged in approximately one hour, but no credit taken
Partition Coefficients	0.01 iodines 0.001 alkali metals	Conforms to RG 1.183, Appendix G, RP 5.6

Table 2.5-1 (continued)

Parameter	Input Value	Basis
SG Secondary Mass	1.253×10^8 gm Minimum for 2 SGs used to maximize activity concentration	Conservative internal calculation
Offsite X/Q	Table 1.8.2-1	Consistent with CLB
Offsite Breathing Rates	3.5E-4 m³/sec 0-8 hrs 1.8E-4 m³/sec 8-24 hrs 2.3E-4 m³/sec > 24 hrs	Conforms to RG 1.183, Main Section, RP 4.1.3
CR X/Q (used worst-case MSSV for first 30 min, then used worst-case ADV for each time step)	8.05E-4 s/m³ 0-0.5 hrs 6.31E-4 s/m³ 0.5-2 hrs 3.65E-4 s/m³ 2-8 hrs 1.64E-4 s/m³ 8-24 hrs 1.09E-4 s/m³ 24-96 hrs 8.26E-5 s/m³ > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
CR Breathing Rate	3.5E-4 m³/sec 0-720 hrs	Conforms to RG 1.183, Main Section, RP 4.2.6
CR Occupancy Factors	100% 0-24 hrs 60% 24-96 hrs 40% > 96 hrs	Conforms to RG 1.183, Main Section, RP 4.2.6
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2
Time of CR Isolation	36 seconds	Conservative value assumed; System design = 5 sec
CR Unfiltered Inleakage	250 cfm	Consistent with proposed new TS limit

(1)   A sensitivity study of the fuel enrichment using ORIGEN-ARP showed that the added 4% margin produced results that bound enrichments up to 4.6 w/o U-235.

Table 2.6-1
CREA Input Parameters

Parameter	Input Value	Basis
Power level for analyses (102% of 3026 MWt)	3086.52 MWt	Conforms to RG 1.183, Main Section, RP 3.1; Licensed power = 3026 MWt
Fuel Enrichment	4.0 w/o	Conforms to RG 1.183, Main Section, RP 3.1
Margin Added to ORIGEN Source Term Results	4%	Conservative addition to accommodate potential future variations in fuel enrichment[1]
Core Fission Product Inventory	Table 1.7.4-1	Conforms to RG 1.183, Main Section, RPs 3.1 and 3.4
Fuel Failure (rods in DNB)	14%	Consistent with CLB
Peaking Factor	1.65	Consistent with CLB
Fission Product Gap Fractions (RG 1.183, Appendix H, Section 1)	0.10 noble gases and iodines 0.12 alkali metals	Conforms to RG 1.183, Appendix H, RP 1
Containment Release Iodine Species Distribution	95% particulate 4.85% elemental 0.15% organic	Conforms to RG 1.183, Appendix H, RP 4
Secondary Release Iodine Species Distribution	0% particulate 97% elemental 3% organic	Conforms to RG 1.183, Appendix H, RP 5
Primary-to-Secondary (P-S) Leak Rate (secondary release model)	300 gpd	Conforms to RG 1.183, Appendix H, RP 7.1
Duration of Secondary Release Event (switch to SDC system)	8 hrs	Conforms to RG 1.183, Appendix H, RP 7.1; Conservative value assumed consistent with CLB
Flashing Fraction of P-S Leakage during Cooldown	0.05	Conservative internal calculation based on flashing model; SG level recovered and tubes re-submerged in approximately one hour, but no credit taken

Table 2.6-1 (continued)

Parameter	Input Value	Basis
Containment Net Free Volume	1.78×10^6 ft^3	Conservative internal calculation
Containment Leak Rates	0.1%/day ≤ 24 hrs 0.05%/day > 24 hrs	Conforms to RG 1.183, Appendix H, RP 6.2
Sedimentation Coefficient (Particulates only)	0.1/hr until DF = 1000, then 0	Conforms to RG 1.183, Appendix H, RP 6.1; Value consistent with Reference 2
Containment Spray	No credit taken	Bounds RG 1.183, Appendix H, RP 6.1; Credit allowed
Penetration Room Ventilation System	No credit taken	Consistent with CLB
Partition Coefficients	0.01 iodines 0.001 alkali metals	Conforms to RG 1.183, Appendix H, RP 7.4
RCS Mass	1.962×10^8 gm Minimum used to maximize activity concentration	Conservative internal calculation
SG Secondary Mass	1.2538×10^8 gm Minimum for 2 SGs used to maximize activity concentration	Conservative internal calculation
Offsite X/Q	Table 1.8.2-1	Consistent with CLB
Offsite Breathing Rates	3.5E-4 m^3/sec 0-8 hrs 1.8E-4 m^3/sec 8-24 hrs 2.3E-4 m^3/sec > 24 hrs	Conforms to RG 1.183, Main Section, RP 4.1.3
CR X/Q (containment release)	8.49E-4 s/m^3 0-2 hrs 5.58E-4 s/m^3 2-8 hrs 2.29E-4 s/m^3 8-24 hrs 1.69E-4 s/m^3 24-96 hrs 1.28E-4 s/m^3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3

Table 2.6-1 (continued)

Parameter	Input Value	Basis
CR X/Q (secondary release) (used worst-case MSSV for first 30 min, then used worst-case ADV for each time step)	8.05E-4 s/m^3 0-0.5 hrs 6.31E-4 s/m^3 0.5-2 hrs 3.65E-4 s/m^3 2-8 hrs 1.64E-4 s/m^3 8-24 hrs 1.09E-4 s/m^3 24-96 hrs 8.26E-5 s/m^3 > 96 hrs	Conforms to RG 1.183, Main Section, RPs 4.2.2 and 4.2.3
CR Breathing Rate	3.5E-4 m^3/sec 0-720 hrs	Conforms to RG 1.183, Main Section, RP 4.2.6
CR Occupancy Factors	100% 0-24 hrs 60% 24-96 hrs 40% > 96 hrs	Conforms to RG 1.183, Main Section, RP 4.2.6
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE Federal Guidance Report 12 EDE	Conforms to RG 1.183, Main Section, RPs 4.1.2 and 4.1.4
Control Room Ventilation System	Table 1.6.3-1	Conforms to RG 1.183, Main Section, RPs 4.2.4 and 5.1.2
Time of CR Isolation	10 seconds	Conservative value assumed; System design = 5 sec
CR Unfiltered Inleakage	250 cfm	Consistent with proposed new TS limit

(1)   A sensitivity study of the fuel enrichment using ORIGEN-ARP showed that the added 4% margin produced results that bound enrichments up to 4.6 w/o U-235.

Table 3-1
Arkansas Nuclear One, Unit No. 2
Summary of Alternative Source Term Analysis Results

CASE	Assumed Unfiltered CR Inleakage (cfm)	EAB Dose[1] (rem TEDE)	LPZ Dose[2] (rem TEDE)	CR Dose[2] (rem TEDE)
LOCA	250	9.084	0.673	2.062
SGTR Pre-existing Iodine Spike	250	2.80	0.14	0.59
MSLB Pre-existing Iodine Spike	250	0.815	0.064	1.008
Acceptance Criteria		**25.0[3]**	**25.0[3]**	**5.0[4]**
SGTR Accident-initiated Iodine Spike	250	2.07	0.11	0.22
MSLB Accident-initiated Iodine Spike	250	2.252	0.225	1.143
LRA	250	1.01	0.16	0.27
Acceptance Criteria		**2.5[3]**	**2.5[3]**	**5.0[4]**
FHA 100-hr decay	250	4.01	0.19	1.19
CREA Containment Release	250	2.265	0.327	0.95
CREA Secondary Release	250	1.422	0.217	0.321
Acceptance Criteria		**6.3[3]**	**6.3[3]**	**5.0[4]**

(1) Worst 2-hour dose   (3) RG 1.183, Table 6
(2) Integrated 30-day dose   (4) 10 CFR 50.67

TABLE REFERENCES

(1) NRC Letter of May 6, 2004, Final Safety Evaluation for Topical Report WCAP-16072- P, Revision 00, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs" TAC NO MB8721.

(2) Westinghouse Calculation CN-CRA-05-62, Rev. 0, "Ringhals Unit 3 LOCA Doses for Uprate Project (GREAT) – Conservative Case," U. Bachrach, January 31, 2006.