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SCE&G®
A SCANA COMPANY

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RC-09-0004

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

ATTN: R. E. Martin

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
LICENSE AMENDMENT REQUEST - LAR 04-02911
LICENSE AMENDMENT AND RELATED TECHNICAL SPECIFICATION CHANGES TO IMPLEMENT FULL-SCOPE ALTERNATIVE SOURCE TERM IN ACCORDANCE WITH 10 CFR 50.67

South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests an amendment to the VCSNS licensing basis and Technical Specifications (TS) that supports a full implementation application of an Alternative Source Term (AST) methodology with the following exceptions. The exceptions are that the current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than Control Room Habitability Envelope (CRHE) doses, and FSAR accidents not included in Regulatory Guide (RG) 1.183.

10 CFR 50.67, "Accident Source Term," provides a mechanism for currently licensed nuclear power reactors to replace the traditional source term used in design basis accident analyses with an alternative source term. Under this provision, licensees who seek to revise the accident source term in design basis radiological consequence analyses must apply for a license amendment under 10 CFR 50.90.

Full implementation AST analyses were performed by SCE&G in accordance with the guidance in Regulatory Guide 1.183, and Section 15.0.1 of the Standard Review Plan (SRP). SCE&G performed AST analyses for the six PWR design basis accidents identified in Regulatory Guide 1.183 that could potentially result in significant Control Room and offsite doses. These include the loss of coolant accident, the main steam line break accident, the refueling accident, the steam generator tube rupture, reactor coolant pump locked rotor and the control rod ejection accident. The analyses demonstrate that using AST methodologies, post-accident Control Room and offsite doses remain within regulatory acceptance limits.

SCE&G proposes implementation of this proposed change through a change to the VCSNS licensing basis, including the TS and associated Bases. Upon approval, conforming changes will be made to the VCSNS Final Safety Analysis Report (FSAR) and subsequently submitted to the NRC staff in accordance with 10 CFR 50.71(e) as part of the regular FSAR update process.

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Proposed changes in the licensing basis for VCSNS resulting from application of the AST include the following:

- New Control Room atmospheric dispersion factors (χ/Qs) based on site specific meteorological data collected between 2002 and 2006.
- Revise the CRHE unfiltered inleakage from 10 scfm to 210 scfm.
- New AST analyses performed in accordance with the guidance in Regulatory Guide 1.183 for the six design basis accidents: loss of coolant accident, the main steam line break accident, the refueling accident, steam generator tube rupture, reactor coolant pump locked rotor and the rod ejection accident.
- Revise the TS to address Improved Standard Technical Specifications Change Traveler (TSTF-51, Revision 2) which permits removal of the Technical Specification requirements for ESF features to be OPERABLE after sufficient radioactive decay has occurred to ensure off-site doses remain below the SRP limits.
- Other TS revisions to reflect the update of the accident source term and associated design basis accidents utilizing the guidance provided in Regulatory Guide 1.183 and the associated Control Room and offsite dose requirements of 10 CFR 50.67.

Table 5-1 of Attachment 5 provides a description of each proposed TS and TS Bases change.

The use of an AST results in changes in the design basis accident radiological consequences; however, the AST methodology has no direct impact on the probability or initiation of the evaluated design basis accidents. Application of AST methodology and the other changes requested by this application for a license amendment do not impact the core damage frequency or the large early release frequency. Therefore, this request for a revision to the VCSNS licensing basis is not being submitted as a "risk-informed approach" using the guidelines in Regulatory Guide 1.174.

Several domestic pressurized water reactors (Byron, Calvert Cliffs, Catawba, Millstone, and Seabrook) have previously provided justification for the use of AST methodology utilizing a similar approach. These applications of AST methodology have been approved by the NRC.

Attachment 1 to this letter contains the overall description and summary of the proposed changes. Attachment 2 provides the detailed AST Safety Assessment Report supporting the proposed AST license basis change. Attachments 3 and 4 are compliance tables addressing SCE&G's method of conforming to the regulatory guidance of Regulatory Guides 1.183 and 1.194 respectively. Attachment 5 contains the safety assessment for the proposed TS and Bases changes and their justification. Attachment 6 provides a mark-up of the current TS. The associated marked up TS Bases changes are provided in Attachment 7 for information only and will be implemented in accordance with the VCSNS Technical Specification Bases Control Program. Attachment 8 contains a list of procedure changes to be completed before AST implementation to support analysis assumptions. Attachment 9 includes the No Significant Hazards Consideration Determination and Environmental Consideration for the proposed changes. Attachment 10 contains non-proprietary calculations that support the Safety Assessment and the updated meteorological data used to calculate the new Control Room χ /Qs. Attachment 11 provides the proposed TS changes in final typed format. Attachment 12 delineates that there are no commitments contained in this submittal.

SCE&G has concluded that the proposed changes do not involve a significant hazards consideration. SCE&G has also determined that the proposed changes satisfy the criteria for a categorical exclusion in accordance with 10 CFR 51.22(c)(9) and does not require an environmental review. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared for these changes.

Implementation of the AST is desired by mid-2010. To support this schedule, SCE&G requests approval of the proposed License Amendment by April, 2010, with the amendment conditioned to be effective within 90 days.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated South Carolina Official.

The proposed change has been reviewed and approved by both the Plant Safety Review Committee and the Nuclear Safety Review Committee.

If you should have any questions regarding this submittal, please contact Mr. Bruce L. Thompson at (803) 931-5042.

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

2/17/09
Executed on

Jeffrey B. Archie

JHW/JBA/wm

Attachments: (12)

- Attachment 1 - Description for the Alternate Source Term License Amendment
- Attachment 2 - AST Safety Assessment Report
- Attachment 3 - Regulatory Guide 1.183 Compliance Table
- Attachment 4 - Regulatory Guide 1.194 Compliance Table
- Attachment 5 - Safety Assessment for the Proposed Technical Specification and Bases Changes
- Attachment 6 - Proposed Technical Specification Changes (Mark-ups)
- Attachment 7 - For Information - Proposed Technical Specification Bases Changes (Mark-ups)
- Attachment 8 - Procedure Changes to be Completed Before AST Implementation
- Attachment 9 - No Significant Hazards Consideration Determination and Environmental Consideration for the Proposed Changes
- Attachment 10 - CD of Non-Proprietary Versions of Supporting Calculations (Six Design Basis Accident Calculations and One Control Room Meteorological Calculation) and Meteorological Data Used to Determine New Control Room χ/Qs
- Attachment 11 - Proposed Technical Specification Changes (Re-Typed)
- Attachment 12 –Regulatory Commitments

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Attachment 1

Description for the Alternative Source Term License Amendment

DESCRIPTION AND SUMMARY OF PROPOSED CHANGES

1.0 INTRODUCTION

South Carolina Gas and Electric Company (SCE&G) hereby proposes to amend the licensing basis of the Virgil C. Summer Nuclear Station (VCSNS) through the full implementation application of an Alternative Source Term (AST) methodology. The exceptions are that the current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than Control Room Habitability Envelope (CRHE) doses, and Final Safety Analysis Report (FSAR) accidents not included in Regulatory Guide 1.183. Applicable, proposed Technical Specification (TS) and TS Bases changes, which are justified by the AST analyses, are included in this application for a license amendment.

This full implementation of AST analyses will modify the VCSNS licensing bases by adopting the AST methodology which replaces the current accident source term with an alternative source term as prescribed in 10 CFR 50.67 and establishes the 10 CFR 50.67 total effective dose equivalent (TEDE) dose limits as a new acceptance criterion. The AST is characterized by the composition and magnitude of the radioactive material, the chemical and physical form of the radionuclides, and the timing of the releases of these radionuclides. The current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than CRHE doses and radiological consequences for FSAR accidents not included in Regulatory Guide 1.183.

The use of an AST results in changes in the design basis accident (DBA) radiological consequences; however, the AST methodology has no direct impact on the probability or initiation of the evaluated design basis accidents. Application of AST methodology and the other changes requested by this application for a license amendment do not increase the core damage frequency or the large early release frequency. Therefore, this request for a revision to the VCSNS licensing basis is not being submitted as a "risk-informed approach" using the guidelines in Regulatory Guide 1.174.

Several domestic pressurized water reactors (Byron, Calvert Cliffs, Catawba, Millstone, and Seabrook) have previously provided justification for the use of AST methodology utilizing a similar approach. These applications of AST methodology have been approved by NRC.

Regulatory Guide 1.183 recommends that changes to the FSAR that reflect the revised analyses or the actual calculation documentation be submitted to the NRC staff.

Upon issuance of a license amendment, conforming FSAR changes will be completed as required by VCSNS procedures and submitted to the NRC staff in accordance with the regular FSAR update process as required by 10 CFR 50.71(e). In lieu of providing the NRC staff with proposed FSAR changes at this time, the supporting DBA calculations are being provided in Attachment 10.

The license amendment would revise the following VCSNS licensing bases:

- New Control Room atmospheric dispersion factors (χ/Qs) based on site specific meteorological data collected between 2002 and 2006, and Regulatory Guides1.194 revised methodology.
- Revise the CRHE unfiltered inleakage from 10 scfm to 210 scfm.
- New AST analyses performed in accordance with the guidance in Regulatory Guide 1.183 for the six design basis accidents: loss of coolant accident, the main steam line break accident, the refueling accident, steam generator tube rupture, reactor coolant pump locked rotor and the control rod ejection accident.
- Revise the TS to address Improved Standard Technical Specifications Change Traveler (TSTF-51, Revision 2) which permits removal of the Technical Specification requirements for ESF features to be OPERABLE after sufficient radioactive decay has occurred to ensure off-site doses remain below the Standard Review Plan (SRP) limits.
- Other TS revisions reflect the update of the accident source term and associated design basis accidents utilizing the guidance provided in USNRC Regulatory Guide 1.183 and the associated control room and offsite dose requirements of 10 CFR 50.67.

Implementation of the AST is desired by mid-2010. To support this schedule, SCE&G requests approval of the proposed License Amendment by April, 2010, with the amendment conditioned to be effective within 90 days.

2.0 REGULATORY BACKGROUND

The current VCSNS licensing basis for design basis accident (DBA) analysis source terms is U.S. Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962. This is consistent with 10 CFR Part 100, Section 11 (10 CFR 100.11), "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," for reactor siting, which contains offsite dose limits in terms of whole body and thyroid dose and further makes reference to TID-14844.

In December 1999, the Nuclear Regulatory Commission (NRC) issued 10 CFR 50.67, "Accident Source Term," which provides a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an AST. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." 10 CFR 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs.

As part of the implementation of the AST, 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 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19, "Control Room," for a loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, fuel handling accident (FHA), steam generator tube rupture (SGTR), reactor coolant pump (RCP) locked rotor accident (LRA) and the control rod ejection accident (CREA).

The accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large break LOCA. As a result of significant core damage, fission products are available for release into the containment environment. The proposed AST is an accident source term that is different from the accident source term used in the original design and licensing of VCSNS. 10 CFR 50.67, as implemented in accordance with RG 1.183, identifies an AST that is acceptable to the NRC staff for use at operating reactors.

The following regulatory requirements and guidance are also considered within this proposed license amendment:

- GDC 19, "Control Room," of Appendix A to 10 CFR Part 50, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of allowable values.

- NUREG-0800, SRP 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms, Revision 0," provides guidance for the safety review of the radiological consequences of DBAs associated with implementing an AST. SRP 15.0.1 supports the guidance outlined in RG 1.183.

- NRC Generic letter 2003-01, "Control Room Habitability," requests addressees to submit information that demonstrates that the Control Room at each of their respective facilities complies with the current licensing and design bases and applicable regulatory requirements, and that suitable design, maintenance and testing control measures are in place for maintaining this compliance.

- USNRC RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," provides guidance on determining atmospheric relative concentration (χ/Q) values in support of design basis Control Room radiological habitability assessments at nuclear power plants. This document describes methods acceptable to the NRC staff for determining χ/Q values that will be used in Control Room radiological habitability assessments performed in support of applications for licenses and license amendment requests. Many of the regulatory positions presented in this guide represent substantial changes from procedures previously used to determine atmospheric relative concentrations for assessing the potential Control Room radiological consequences for a range of postulated accidental releases of radioactive material to the atmosphere. These revised procedures are largely based on the NRC sponsored computer code, ARCON96.

3.0 SAFETY ASSESSMENT

SCE&G has performed analyses to support a full implementation of the AST as defined in RG 1.183 with the exception that the current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than CRHE doses and radiological consequences for FSAR accidents not included in Regulatory Guide 1.183. A detailed description of the AST analyses, including a safety assessment, is provided in Attachment 2. Copies of AST accident dose calculations and the calculation that determines the atmospheric dispersion coefficients for the Control Room required to support the licensing bases changes for AST are included in Attachment 10.

The detailed Safety Assessment Report associated with AST analyses is provided in Attachment 2. This report is supplemented with RG 1.183 and 1.194 Compliance Tables presented in Attachments 3 and 4.

The basis/safety assessment associated with the proposed changes to the VCSNS Technical Specifications and Bases is included in Table 5-1 of Attachment 5 and supported by the Safety Assessment in Attachment 2 and the calculations in Attachment 10.

A list of procedure changes to be completed before AST implementation is included in Attachment 8.

A No Significant Hazards Consideration Determination and Environmental Consideration for the proposed changes are included in Attachment 9.

4.0 CONCLUSION

In conclusion, based on the considerations discussed above and detailed in the remainder of this submittal, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

A comparison of the calculated AST doses and the RG 1.183 dose limits for the Control Room operator, the exclusion area boundary and the low population zone for the DBA LOCA, MSLB, FHA, SGTR, LRA, and CREA, are provided in the table below. Note that the calculated AST doses provide a conservative estimate of the dose consequences for the postulated event. The conservative models and methodologies that were used to calculate the dose consequences are presented in Attachment 2.

Summary of Design Basis Accident Radiological Consequences for AST Calculated Dose Versus RG 1.183 Dose Criteria			
		AST Calculated Dose (Rem TEDE)	RG 1.183 Dose Criteria (Rem TEDE)
Loss of Coolant Accident (LOCA)	CR Operator Dose	1.01	5
	EAB Dose	4.87	25
	LPZ Dose	0.54	25
Main Steam Line Break (MSLB)	Case 1: Concurrent Iodine Spike		
	CR Operator Dose	0.37	5.0
	EAB Dose	0.78	2.5
	LPZ Dose	0.16	2.5
	Case 2: Pre-existing Iodine Spike		
	CR Operator Dose	1.15	5.0
	EAB Dose	1.96	25
	LPZ Dose	0.13	25
Fuel Handling Accident (FHA)	Case 1: FHA Inside Containment		
	CR Operator Dose	0.76	5.0
	EAB Dose	4.29	6.3
	LPZ Dose	1.06	6.3
	Case 2: FHA Inside Fuel Handling Building		
	CR Operator Dose	0.41	5.0
	EAB Dose	4.29	6.3
	LPZ Dose	1.06	6.3
Steam Generator Tube Rupture (SGTR)	Case 1: Concurrent Iodine Spike		
	CR Operator Dose	0.37	5.0
	EAB Dose	0.71	2.5
	LPZ Dose	0.047	2.5
	Case 2: Pre-existing Iodine Spike		
	CR Operator Dose	1.18	5.0
	EAB Dose	2.08	25
	LPZ Dose	0.12	25
Locked Rotor Accident (LRA)	CR Operator Dose	2.43	5.0
	EAB Dose	2.20	2.5
	LPZ Dose	0.45	2.5
Control Rod Ejection Accident (CREA)	Case 1: Containment Release Path		
	CR Operator Dose	1.71	5.0
	EAB Dose	4.31	6.3
	LPZ Dose	0.95	6.3
	Case 2: Steam Generator Release Path		
	CR Operator Dose	2.38	5.0
	EAB Dose	2.52	6.3
	LPZ Dose	0.48	6.3

Attachment 2

AST Safety Assessment Report

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Acronyms and Abbreviations

μCi/gm	micro-curies per gram
χ/Q	atmospheric dispersion factor in sec/m^3
acfm	actual cubic feet per minute
AOP	Abnormal Operating Procedure
ASHRAE	American Society of Heating, Refrigeration and Air Conditioning Engineers
AST	Alternative Source Term
cc	cubic centimeter
cfm	cubic feet per minute
CFR	Code of Federal Regulations
Ci	curie
CLB	current licensing basis
CR	Control Room
CREA	control rod ejection accident
CRHE	Control Room Habitability Envelope
CsI	cesium iodine
d	day
DBA	Design Basis Accident
DE	Dose Equivalent
DF	decontamination factor
EAB	exclusion area boundary
EAHS	Emergency Air Handling System
ECCS	emergency core cooling system
EDE	effective dose equivalent
EOP	Emergency Operating Procedure
ESF	engineered safeguard features
°F	degrees Fahrenheit
FHA	fuel handling accident
FHB	Fuel Handling Building
FSAR	Final Safety Analysis Report
ft	feet
ft^3	cubic feet
gpm	gallons per minute
GTP	General Test Procedure
GWD/MTU	gigawatt days/metric tons uranium
HEPA	high efficiency particulate air
hr	hour
IB	Intermediate Building
in	inch
kw	kilowatt
lbm	pounds-mass
LAR	License Amendment Request
LCO	Limiting Condition For Operation
LOCA	loss-of-coolant accident
LPZ	low population zone
LRA	locked rotor accident
m	meters

Acronyms and Abbreviations

m^3	cubic meters
min	minute
MS	main steam
MSLB	main steam line break
MTU	metric tons uranium
MWD/MTU	megawatt days/metric tons uranium
MWt	megawatt thermal
NaOH	Sodium Hydroxide
NRC	Nuclear Regulatory Commission
PF	partition factor
PORV	power operated relief valves
psig	pounds per square inch gauge
PWR	pressurized water reactor
RB	Reactor Building
RCCA	rod cluster control assembly
RCS	reactor coolant system
RCP	reactor coolant pump
Rem	roentgen equivalent man
RG	Regulatory Guide
RTP	rated thermal power
RWST	Refueling Water Storage Tank
SCE&G	South Carolina Gas & Electric
scfm	standard cubic feet per minute
sec	second
SER	Safety Evaluation Report
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SRP	Standard Review Plan
SSV	secondary safety valves
TEDE	total effective dose equivalent
TS	Technical Specification
TSC	Technical Support Center
USNRC	United States Nuclear Regulatory Commission
UUI	unanticipated unfiltered inleakage
VCSNS	Virgil C. Summer Nuclear Station

1.0 DESCRIPTION

In accordance with 10 CFR 50.67, "Accident Source Term," a licensee may voluntarily revise the accident source term used in design basis radiological consequence analyses. Paragraph 50.67(b) requires that applications under this section contain an evaluation of the consequences of applicable design basis accidents (DBAs) previously analyzed in the plant Final Safety Analysis Report (FSAR). Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 1), provides guidance to licensees on performing evaluations, and reanalyzes as required to adopt an alternative source term (AST).

VCSNS has performed radiological consequence analyses of the six applicable pressurized water reactor (PWR) DBAs identified in RG 1.183. These DBAs are a loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, fuel handling accident (FHA), steam generator tube rupture (SGTR), reactor coolant pump (RCP) locked rotor accident (LRA) and the control rod ejection accident (CREA). These analyses were performed using the guidance of RG 1.183 and Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 2). The analyses were prepared, reviewed, and approved in accordance with the VCSNS 10 CFR 50, Appendix B Quality Assurance Program. Comparison with the guidance contained in RG 1.183 and RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," (Reference 3) is summarized in Attachments 3 and 4 respectively of this license amendment request (LAR).

The supporting analyses consisted of the following steps:

- Determination of the AST based on plant-specific analysis of the fission product inventory.
- Application of the release fractions for the six PWR DBAs.
- Application of the deposition and removal mechanisms.
- Evaluation of activity transport pathways to the environment.
- Analysis of the atmospheric dispersion for the radiological propagation pathways.
- Calculation of the offsite and Control Room personnel Total Effective Dose Equivalent (TEDE).
- Evaluation of other related design and licensing bases pertaining to NUREG-0737 (Reference 4) requirements.

The radiological dose analyses have been performed assuming reactor operation at 2958 MWt (102% of the rated power level of 2900 MWt). This results in a conservative estimate of fission product releases.

2.0 PROPOSED CHANGES

The licensing and design basis changes included in this LAR are described below. The proposed Technical Specification (TS) and Bases changes are described in Attachment 5 and a mark-up of the affected TS and Bases pages is provided in Attachments 6 and 7 respectively.

3.0 BACKGROUND

On December 23, 1999, the NRC published 10 CFR 50.67, "Accident Source Term," in the Federal Register. This regulation provides a mechanism for licensed power reactors to replace the current accident source term used in design basis accident (DBA) analyses with an alternative source term. The direction provided in 10 CFR 50.67 is that licensees who seek to revise their current accident source term in design basis radiological consequence analyses must apply for a license amendment under 10 CFR 50.90.

Regulatory Guide (RG) 1.183 and Standard Review Plan Section 15.0.1 were used by SCE&G in preparing the AST analyses. These documents were prepared by the NRC staff to address the use of ASTs at current operating power reactors. The RG establishes the parameters of an acceptable AST and identifies the significant attributes of an AST acceptable to the NRC staff. In this regard, the RG provides guidance to licensees for operating power reactors on acceptable applications for an AST; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on risk; and acceptable radiological analysis assumptions. The SRP provides guidance to the staff on the review of AST submittals.

Acceptance criteria consistent with that required by 10 CFR 50.67 were used to replace VCSNS current design basis source term acceptance criteria. The AST analyses were performed for the six PWR DBAs identified in RG 1.183 that could potentially result in Control Room and offsite doses. These include the loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, fuel handling accident (FHA), steam generator tube rupture (SGTR), reactor coolant pump (RCP) locked rotor accident (LRA) and the control rod ejection accident (CREA).

4.0 TECHNICAL ANALYSIS

4.1 Atmospheric Dispersion Factors – Offsite and CRHE

The χ/Q values used in the AST analyses at the EAB and LPZ are the current licensing values described in the VCSNS Updated FSAR section 2.3.4. Use of the χ/Q values previously approved by the staff during the initial facility licensing is acceptable for use in the AST analyses as discussed in RG 1.183, Section 5.3.

The χ/Q values at the CR intake are calculated using the NRC-sponsored computer codes ARCON96 consistent with the procedures in RG 1.194.

4.1.1 Atmospheric Dispersion Factors – Offsite

Table 4.1-1 provides the offsite atmospheric diffusion coefficients (χ/Qs) used for the AST.

Table 4.1-1 Offsite χ/Qs (sec/m^3)		
Time Period	EAB	LPZ
0 – 2 hours	4.08E-04	-
0 – 8 hours		2.37E-05
8 - 24 hours		2.44E-06
1 - 4 days		1.11E-06
4 - 30 days		6.28E-07

4.1.2 Atmospheric Dispersion Factors - CRHE

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the new CRHE χ/Qs is provided in Attachment 10 as Calculation DC00040-079, "Atmospheric Dispersion Coefficients for Control Room."

The atmospheric dispersion factors (χ/Q) were calculated using plant specific meteorological data and the ARCON96 (Reference 5) computer code per RG 1.194. The meteorological data files used for the CRHE includes a record of each hourly data containing a location identifier, Julian day (1-366), hour (0 to 23), low-level direction, low-level speed, stability class (1=A to 7=G), upper level direction, and upper level speed. Wind speeds are entered in tenths of a reporting unit with no decimal. Wind directions are from 1 to 360 in degrees. The five yearly files were combined into one file for ease of code execution. This file contains all the data for the VCSNS site from 2002-2006, which satisfies the ARCON96 requirements for having 3 to 5 years of hourly data. The meteorological tower used for collecting the data is located on the plant site. Instrumentation is provided at the 10 and 60 meter level. Attachment 10 of this LAR contains a CD with the updated meteorological data used to calculate the new CRHE atmospheric diffusion coefficients.

4.1.2.1 Infiltration

Section 3.3.3, "Infiltration Pathways", of RG 1.194 states that a χ/Q should be determined for any infiltration pathway that could result in a significant intake of radioactive contaminated air into the Control Room Habitability Envelope (CRHE). The infiltration pathways actually applicable to a particular facility will be identified via inleakage testing or CRHE inspections and surveillances.

The current licensing basis for the infiltration rate is 10 scfm, attributed to leakage through improperly sealed doors or by personnel ingress and egress, per NUREG-0800, USNRC Standard Review Plan, Section 6.4, "Control Room Habitability Systems," Reference 6.

A base line ASTM E741 integrated test was performed in March 2005 to measure leakage into the CRHE. As reported in Reference 10 and acknowledged by Reference 7, filtered outside air was found to be within current limits of 1000 scfm per train and the maximum unanticipated unfiltered inleakage (UUI) recorded was 41 scfm. Although the maximum UUI recorded was 41 scfm, a UUI of 200 scfm is utilized in the AST analyses to provide margin for future testing.

This base infiltration rate is augmented by adding to it the estimated contribution from opening and closing of doors associated with such activities in accordance with by the plant emergency plans and procedures. Normally, 10 scfm is used for this additional contribution. A value of 10 scfm is utilized in this evaluation.

4.1.2.2 Calculations

The ARCON96 computer code (Reference 5) was used to calculate the Control Room χ/Qs. The receptors considered in the calculation are the two Control Room intakes. The 'A' train intake is at elevation 484 ft and the 'B' train is at elevation 507 ft-9 in above grade. Ground level for the VC Summer site is at elevation 436 ft. Therefore, the intake height for Intake 'A' is 48 ft or 14.6 m and Intake 'B' 71.75 ft or 21.9 m. All elevations are measured from the same reference point and the elevation difference is zero.

The release points to the environment evaluated for possible modeling with the ARCON96 computer code are listed in Table 4.1-2.

Table 4.1-2: Release Points to the Environment

Release Point	Release Height (height – 436 ft)m	Direction from Intake to Release		Straight-line Horizontal Distance (m)	
		Intake A	Intake B	Intake A	Intake B
Main Plant Vent	524 ft-3 in (26.9m)	38°	42°	89.0	86.9
Purge Exhaust	524 ft-0 in (26.8m)	38°	42°	87.5	86.3
MS PORV A	504 ft-0 in (20.7m)	64°	59°	64.0	63.1
MS SSV A	495 ft-0 in (B,C,D,E) (18.0m)	57°	64°	60.0	60.0
MS SSV A	503 ft-0 in (A) (20.4m)	57°	64°	60.0	60.0
MS PORV B	470 ft-0 in (10.4m)	72°	78°	80.8	82.0
MS SSV B	495 ft-0 in (18.0m)	71°	76°	75.6	78.0
MS PORV C	470 ft-0 in (10.4m)	69°	73°	105.8	105.8
MS SSV C	495 ft-0 in (18.0m)	73°	76°	104.5	103.6
IB Blowout Panel Releasing Directly to Environment	463 ft-0 in (8.2m)	73°	78°	67.4	69.5
RB Nearest Point	0 ft	63°	57°	61.0	61.0

The resulting Control Room χ/Qs are given in Table 4.1-3.

Table 4.1-3: CRHE χ/Qs (sec/m^3)							
	Time	Intake A	Intake B		Time	Intake A	Intake B
RB Nearest Point				MS PORV B			
	0-2 h	1.39E-03	1.30E-03		0-2 h	8.69E-04	8.31E-04
	2-8 h	1.17E-03	1.09E-03		2-8 h	7.16E-04	6.89E-04
	8-24 h	5.70E-04	5.27E-04		8-24 h	3.44E-04	3.23E-04
	1-4 d	4.17E-04	3.90E-04		1-4 d	2.49E-04	2.33E-04
	4-30 d	3.00E-04	2.82E-04		4-30 d	1.84E-04	1.69E-04
Main Plant Vent				MS SSV B			
	0-2 h	7.11E-04	7.43E-04		0-2 h	9.61E-04	9.29E-04
	2-8 h	5.05E-04	5.41E-04		2-8 h	7.64E-04	7.08E-04
	8-24 h	2.51E-04	2.75E-04		8-24 h	3.67E-04	3.39E-04
	1-4 d	2.04E-04	2.16E-04		1-4 d	2.65E-04	2.44E-04
	4-30 d	1.39E-04	1.49E-04		4-30 d	1.98E-04	1.79E-04
Purge Exhaust				MS PORV C			
	0-2 h	7.23E-04	7.57E-04		0-2 h	5.18E-04	5.10E-04
	2-8 h	5.24E-04	5.47E-04		2-8 h	4.37E-04	4.34E-04
	8-24 h	2.59E-04	2.78E-04		8-24 h	2.09E-04	2.05E-04
	1-4 d	2.13E-05	2.19E-04		1-4 d	1.51E-04	1.49E-05
	4-30 d	1.44E-04	1.51E-04		4-30 d	1.11E-04	1.08E-04
IB Blowout Panel				MS SSV C			
	0-2 h	1.22E-03	1.12E-03		0-2 h	5.36E-04	5.44E-04
	2-8 h	1.01E-03	9.16E-04		2-8 h	4.10E-04	4.18E-04
	8-24 h	4.79E-04	4.32E-04		8-24 h	1.99E-04	2.00E-04
	1-4 d	3.48E-04	3.12E-04		1-4 d	1.44E-04	1.44E-04
	4-30 d	2.53E-04	2.24E-04		4-30 d	1.06E-04	1.06E-04
MS SSV A (Relief's B,C,D,E)				MS PORV A			
	0-2 h	1.50E-03	1.51E-03		0-2 h	1.34E-03	1.37E-03
	2-8 h	1.15E-03	1.17E-03		2-8 h	1.01E-03	1.03E-03
	8-24 h	5.64E-04	5.75E-04		8-24 h	4.97E-04	5.07E-04
	1-4 d	4.23E-04	4.18E-04		1-4 d	3.64E-04	3.77E-04
	4-30 d	3.03E-04	3.10E-04		4-30 d	2.69E-04	2.72E-04
MS SSV A (A-relief only)							
	0-2 h	1.50E-03	1.51E-03				
	2-8 h	1.12E-03	1.15E-03				
	8-24 h	5.51E-04	5.67E-04				
	1-4 d	4.15E-04	4.13E-04				
	4-30 d	2.97E-04	3.02E-04				

Based on the calculated Control Room χ/Qs, the conservative values used in each of the AST dose analyses are summarized as follows:

LOCA	Reactor Building nearest point (Intake A)
MSLB	MS SSV A (Relief's B, C, D, E), Intake B
FHA (Inside Containment)	RB Nearest Point Intake A
FHA (Outside Containment)	Main Plant Vent Intake B
SGTR	MS SSV A (Relief's B, C, D, E), Intake B
RCP LRA	MS SSV A (Relief's B, C, D, E), Intake B
CREA (Containment Release)	RB Nearest Point Intake A
CREA (Steam Generator Release)	MS SSV A (Relief's B, C, D, E), Intake B

4.2 Accident Source Terms

The inventory of fission products in the reactor core available for release to the containment is based on a core thermal power of 2958 MWt (102% of RTP). A list of the 60 isotopes used in the AST analysis is given in Table 4.2-1.

The core inventories are taken from the Westinghouse Radiation Analysis Manual, Revision 1 for VCSNS Uprating updated 12/98 (Reference 8). Reference 8 does not provide core inventory values for all of these 60 isotopes. Therefore, the default core inventories from Table 1.4.3.2-2 of NUREG/CR-6604 (Reference 13), corrected to a core thermal power of 2958MWt, were included in the analysis. In addition, the noble gas and iodine core inventories from the Westinghouse Radiation Analysis Manual and the corrected core inventories from NUREG/CR-6604 were compared and the larger of the two inventories was conservatively used in the subsequent analyses.

Table 4.2-1: Core Activity Available for Release – Design Basis

Isotope	Curies	Isotope	Curies	Isotope	Curies
Co-58	7.55E+05	Ru-103	1.06E+08	Cs-136	3.08E+06
Co-60	5.78E+05	Ru-105	6.92E+07	Cs-137	5.66E+06
Kr-85	8.30E+05	Ru-106	2.42E+07	Ba-139	1.47E+08
Kr-85m	2.72E+07	Rh-105	4.79E+07	Ba-140	1.46E+08
Kr-87	4.96E+07	Sb-127	6.53E+06	La-140	1.49E+08
Kr-88	6.71E+07	Sb-129	2.31E+07	La-141	1.37E+08
Rb-86	4.43E+04	Te-127	6.31E+06	La-142	1.32E+08
Sr-89	8.41E+07	Te-127m	8.35E+05	Ce-141	1.32E+08
Sr-90	4.54E+06	Te-129	2.17E+07	Ce-143	1.29E+08
Sr-91	1.08E+08	Te-129m	5.72E+06	Ce-144	7.98E+07
Sr-92	1.13E+08	Te-131m	1.10E+07	Pr-143	1.26E+08
Y-90	4.87E+06	Te-132	1.09E+08	Nd-147	5.65E+07
Y-91	1.02E+08	I-131	8.20E+07	Np-239	1.51E+09
Y-92	1.13E+08	I-132	1.20E+08	Pu-238	8.58E+04
Y-93	1.28E+08	I-133	1.68E+08	Pu-239	1.94E+04
Zr-95	1.29E+08	I-134	1.80E+08	Pu-240	2.44E+04
Zr-97	1.35E+08	I-135	1.54E+08	Pu-241	4.11E+06
Nb-95	1.22E+08	Xe-133	1.70E+08	Am-241	2.72E+03
Mo-99	1.43E+08	Xe-135	3.70E+07	Cm-242	1.04E+06
Tc-99m	1.23E+08	Cs-134	1.01E+07	Cm-244	6.08E+04

4.3 Loss of Coolant Accident

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the LOCA is provided in Attachment 10 as Calculation DC00040-097, "Loss of Coolant Accident - AST". Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the LOCA CRHE and offsite doses.

Table 4.3-1 provides a comparison of key parameters utilized to determine the existing licensing basis LOCA and the AST LOCA doses.

Table 4.3-1: Design Input Comparison– Current Licensing Basis vs. AST Design – LOCA

Parameter	CLB Parameter	AST Parameter
Core Thermal Power Level	2958 MWt	2958 MWt
Activity Inventory in Core	18 isotopes (I, Kr and Xe)	60 dose significant isotopes used in RADTRAD
Radioisotope Decay Properties	Table of Isotopes	RADTRAD Table 1.4.3.2-3
Activity Release to Containment	Per R.G. 1.4	Per R.G. 1.183 Table 2 (Gap & Early In-Vessel Phases Only)
Release Timing	Instantaneous	Per R.G. 1.183 Table 4
Radioiodine Chemical Species	91% Elemental 5% Particulate 4% Organic	95% Aerosol (CsI) 4.85% Elemental 0.15% Organic
Primary Containment Volume	Total free volume = 1,840,000 ft^3	Total free volume = 1,840,000 ft^3
Primary Containment Cleanup (Natural Deposition)	50% plateout of released iodine	Aerosol removal via Natural Deposition only credited when sprays are not operating (10th percentile Powers Model)
Containment Sprays	$\lambda e = 20\ hr^{-1}$ (maximum DF for the elemental iodine spray removal coefficient is 100) $\lambda p = 5.68\ hr^{-1}$(0 – 2176 sec) $\lambda p = 0.568\ hr^{-1}$ (>2176 sec) No credit is taken for the removal of organic iodine.	$\lambda e = 20\ hr^{-1}$ (maximum DF for the elemental iodine spray removal coefficient is 200) $\lambda p = 5.68\ hr^{-1}$(0 – 2176 sec) $\lambda p = 0.568\ hr^{-1}$ (>2176 sec) No credit is taken for the removal of organic iodine.
Containment Recirculation Particulate Filter	90 percent for the removal of iodine particulates only	90 percent for the removal of iodine particulates only
Primary Containment Design Leak Rate	0.2%/day for first 24 hours; 0.1%/day thereafter.	0.2%/day for first 24 hours; 0.1%/day thereafter.
Minimum Sump Volume Post LOCA	55,730 ft^3	55,730 ft^3
ESF System Leakage Source Term to Environment	Iodine only	Iodine only
ESF Leakage Outside of the RB	12,000 cc/hr	12,000 cc/hr
ESF Leakage post-LOCA Time	Begins at 1382 sec – ends at 30 days	Begins at 1460 sec – ends at 30 days
Sump Maximum Temperature	< 245°F	< 245 °F
ESF Flash Fraction	10%	10%
Iodine Re-evolution	None assumed	None assumed since pH >7

Table 4.3-1: Design Input Comparison– Current Licensing Basis vs. AST Design – LOCA		
Parameter	CLB Parameter	AST Parameter
RB Sump Iodine Species	91% elemental, 5% particulate, & 4% organic	97% elemental, 3% organic
Dose Conversion Factors	ICRP 30	RADTRAD Table 1.4.3.3-2
Control Room Habitability Envelope (CRHE) Volume	226,040 ft^3	226,040 ft^3
CRHE Isolation Time	0	0
CRHE Emergency Filtered Intake Air Flow	1,000 cfm	1,265 cfm
CRHE Unfiltered Air Inleakage	10 cfm	243 cfm
CRHE Filtered Recirculation Flow Rate	19,125 cfm	19,125 cfm
CRHE Emergency Filter Bed Depth	2 in. charcoal	2 in. charcoal
CRHE Emergency Filter Bed Removal Efficiency	95%	95%
CRHE Operator Breathing Rates	3.5E-04 m^3/sec (0 – 30 days)	3.5E-04 m^3/sec (0 – 30 days)
CRHE Operator Occupancy Factors	1.0 0-24 hrs 0.6 1-4 days 0.4 4-30 days	1.0 0-24 hrs 0.6 1-4 days 0.4 4-30 days

4.3.1 Introduction and Assumptions - LOCA

The following primary assumptions from previous LOCA analyses continue to apply:

1. Two release pathways to the environment are considered: containment leakage (0.2%/day for first 24 hours; 0.1%/day thereafter.) and ECCS recirculation leakage (12,000 cc/hr). Back leakage through the RWST and NaOH Tank is not considered in the LOCA analysis as a potential source of post LOCA leakage based on plant procedures (EOPs) that require closure of the 20" RWST outlet valve (6700) and closure of the 3" NaOH outlet valve (3012) following the transition to cold leg recirculation. This results in 3 valve isolation and a minimum of 2 valve isolation in the long term with a single failure.
2. A single electrical train is assumed to remove the following equipment from service: one containment spray train, one ECCS train and one of the two reactor building cooling units.
3. Containment spray removal coefficients continue to be based on the methodology in NUREG-0800, "Standard Review Plan For The Review Of Safety Analysis Reports For Nuclear Power Plants", Section 6.5.2, "Containment Spray as a Fission Product Cleanup System".

The following DBA LOCA dose contributors to the CRHE are included in this analysis:

1. Contamination of the Control Room atmosphere by the intake or the infiltration of the radioactive material contained in the radioactive plume.
2. Radiation shine from the external radioactive plume released from the facility.
3. Radioactive shine from radioactive materials in containment.

4.3.2 Source Term

The core inventory is provided in Table 4.2-1. The release fractions and timing are taken from RG 1.183, Tables 2 and 4 as shown below.

4.3.3 Mitigation

The radiological consequences of the LOCA are actively mitigated by containment spray, natural deposition, reactor building cooling unit particulate filter, radiological decay, limitations on leakage, and control room filtration (i.e., with an assumed charcoal filtration efficiency of 95%).

As described in Section 9.4.1 of the VCSNS updated FSAR, the control room ventilation system is automatically placed in the emergency mode, with filtration of incoming and recirculated air, following receipt of a SI or high radiation signal from the gaseous activity channel of RM-A1. Since a large break LOCA results in a near instantaneous Reactor Trip and SI actuation [i.e., within the first few seconds on either low RCS pressure or high containment pressure (Hi-1)], the control room is assumed to enter the emergency recirculation mode on event initiation since delay times of this order (i.e., seconds) are inconsequential to the results of the analysis.

4.3.4 Radiological Transport Modeling

A simplified radiological release model and individual pathway models developed to calculate LOCA doses utilizing RADTRAD is shown in Figures 1 and 2. Specific model details and the supporting RADTRAD runs are provided in Attachment 10 as Calculation DC00040-097, "Loss of Coolant Accident – AST." Based on the calculated Control Room χ/Qs provided in Table 4.1-3, the conservative values used in the LOCA AST dose analyses are taken at the following release point.

- LOCA — Reactor Building nearest point (intake A)

4.3.5 Results – Control Room Operator Dose

The following DBA LOCA dose contributors to the CRHE are included in this analysis:

- Contamination of the Control Room atmosphere by the intake or the infiltration of the radioactive material contained in the radioactive plume.
- Radiation shine from the external radioactive plume released from the facility.
- Radioactive shine from radioactive materials in containment.

The contribution from each pathway and the total CR operator dose is shown as follows.

LOCA CR Operator Dose	Rem TEDE
• Containment leakage =	7.04E-01
• Containment shine =	2.38E-03
• External cloud from Containment leakage =	2.02E-02
• ESF Recirculation leakage =	2.69E-01
• External cloud from ESF Recirculation leakage =	1.42E-02
• Total	1.01E+00

4.3.6 Results – Offsite Doses

The RADTRAD computer code was used to determine the offsite dose. The calculated doses are shown as follows where the EAB dose represents the maximum 2-hour TEDE over the accident period.

LOCA EAB Dose	Rem TEDE
• Containment leakage =	4.52
• ESF Recirculation leakage =	0.35
• Total =	4.87

LOCA LPZ Dose	Rem TEDE
• Containment leakage =	0.43
• ESF Recirculation leakage =	0.11
• Total =	0.54

4.3.7 Conclusion

The LOCA Control Room operator dose is below the 5 rem TEDE regulatory limit and the offsite doses are well below the 25 rem TEDE regulatory limit.

4.3.8 Summary of Calculation Conservatisms

- Primary containment cleanup by natural deposition assumes the 10th percentile Power's Aerosol Decontamination Model.

- The analysis is based on the 60 isotopes of NUREG/CR-6604. For those isotopes not included in the VCSNS plant data, the default core inventories from Table 1.4.3.2-2 of NUREG/CR-6604, corrected to a core thermal power of 2958 MWt, are included in this analysis. In addition, the noble gas and iodine core inventories from VCSNS plant data and the corrected core inventories from NUREG/CR-6604 were compared and the larger of the two concentrations is used in this analysis.

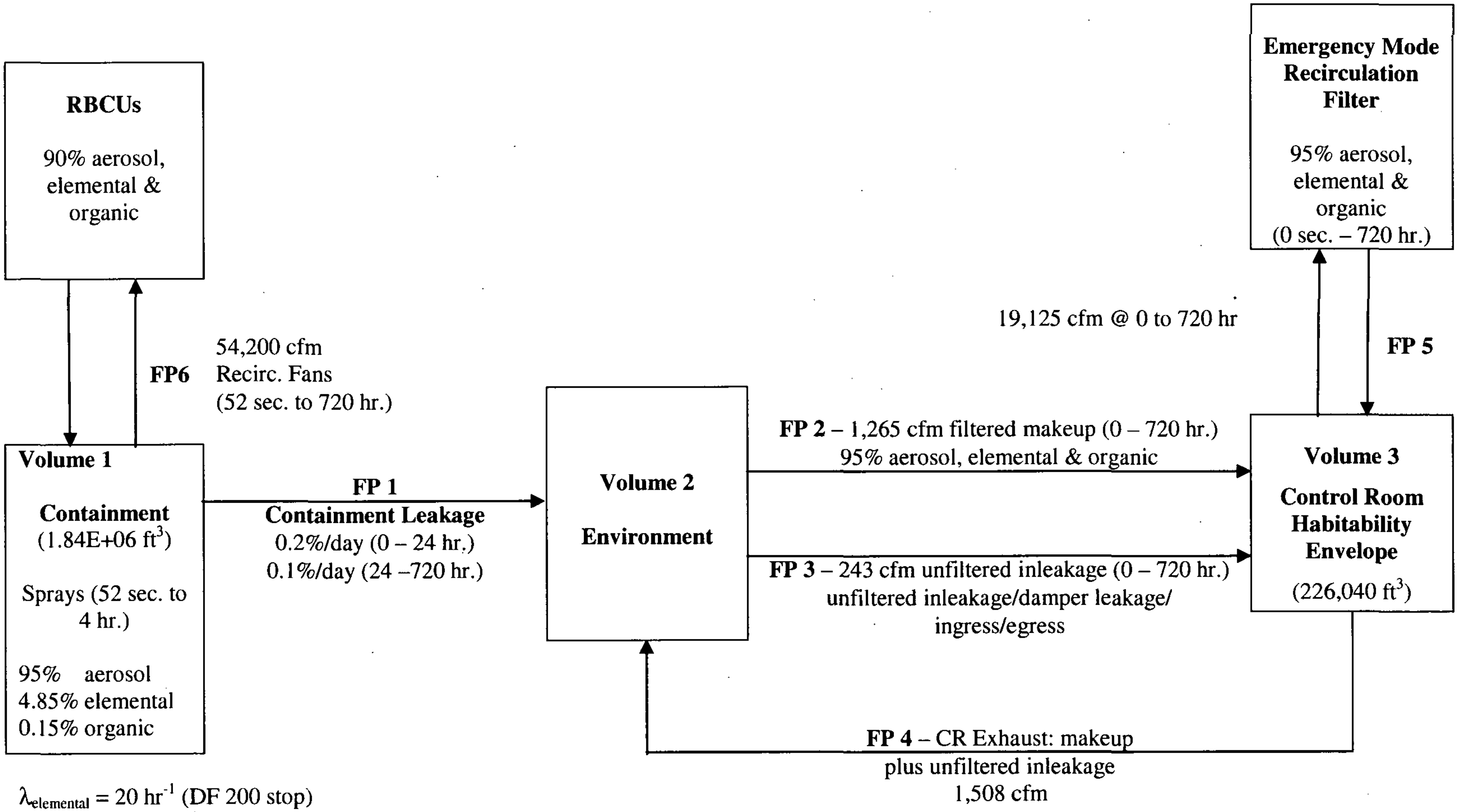
- The Control Room unanticipated unfiltered inleakage (UUI) value of 41 scfm, based on tracer tests, was increased to 200 scfm. An uncertainty value of 25 percent is conservatively applied to the Control Room filtered makeup flow rate and the unfiltered makeup flow rate which bypasses the damper.

- The VCSNS technical specifications do not provide a specific limit for operational leakage that is allowed within the recirculation loop. Administrative limits, however, ensure that operational leakage is adequately controlled. A post LOCA recirculation leakage of 12,000 cc/hr (7.063E-03 cfm) is used as input to the dose calculations. This is twice the operational limit that is used in plant procedures for system leakage assessments (GTP-006, Reference 11). In the event total recirculation loop leakage exceeds 6,000 cc/hr, a Condition Evaluation Report is generated to facilitate a licensing basis impact assessment and an operability determination. No credit is taken for holdup or filtration of this leakage in the Auxiliary Building, i.e., the iodine released by the recirculation loop leakage is assumed to be immediately available for release to the atmosphere.

- The containment recirculation HEPA efficiency is assumed to be 90 percent for the removal of iodine particulates only. This is conservative since Regulatory Guide 1.52 permits a 99 percent removal efficiency, based on the filter's characteristics.

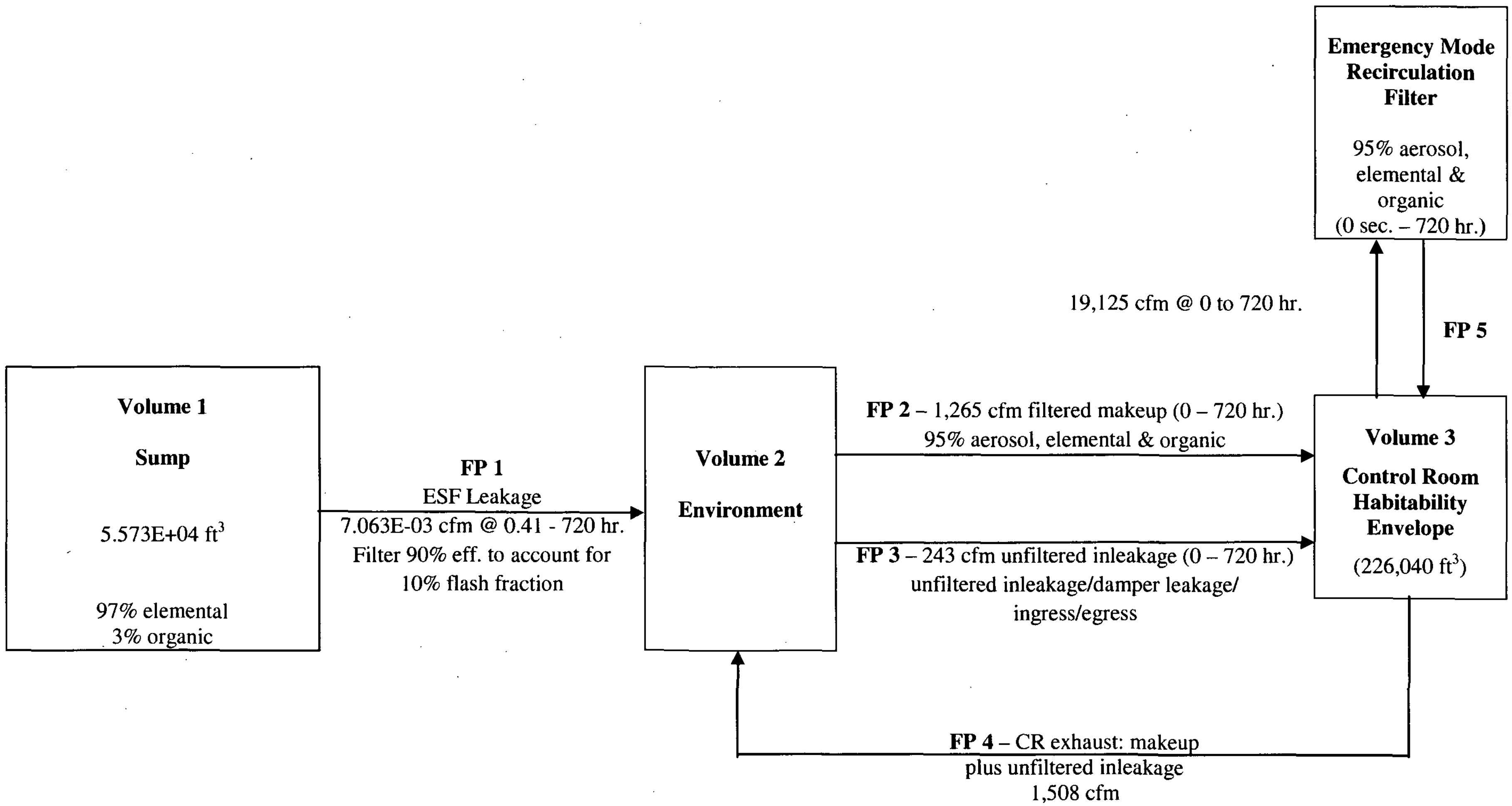
- It is assumed that the sprays will operate for the first 4 hours of the accident.

Figure 1: Activity Flow Path Model Developed to Calculate LOCA Doses from Containment Leakage



$\lambda_{elemental} = 20\ hr^{-1}$ (DF 200 stop)
$\lambda_{particulate} = 5.68\ hr^{-1}$ (DF 50 reduce by 10%)
$\lambda_{natural\ dep.} = 0.1\ hr^{-1}$ (stop when sprays operating)

Figure 2: Activity Flow Path Model Developed to Calculate LOCA Doses from ESF Leakage



4.4 Fuel Handling Accident

A fuel handling accident (FHA) during refueling could release a fraction of the fission product inventory in the plant to the environment. Two accident scenarios are considered: (1) a refueling accident occurring inside containment and (2) a refueling accident occurring outside containment. A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the FHA are provided in Attachment 10 as Calculation DC00040-102, "Fuel Handling Accidents – AST. Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the FHA onsite and offsite TEDE.

Table 4.4-1 provides a comparison of the design inputs utilized to determine the existing licensing basis FHA and the AST FHA doses.

Table 4.4-1: Design Input Comparison – Current Licensing Basis vs. AST Design – FHA

Parameter	CLB Parameter	AST Parameter
Core Thermal Power Level	2958 MWt	2958 MWt
Earliest Fuel Handling Time	72 hr	72 hr
Number of Damaged Fuel Pins	314	314
Core Radial Peaking Factor	1.7	1.7
Fraction of Fission Products in the Gap Available for Release from Damaged Rods	Iodines except I-131 0.10 Iodine-131 0.12 Kr-85 0.30 Other nobles gases 0.10	I-131 0.16 Other Halogens 0.10 Kr-85 0.20 Other Noble Gases 0.10 Alkali Metals 0.24
Core Burnup (MWD/MTU)	Up to 70,000	Up to 70,000
Maximum Fuel Rod Pressurization	<1200 psig	<1200 psig
Release Timing (Gap)	Instantaneously released & mixed into pool water	Instantaneously released & mixed into pool water
Minimum Pool Water Depth	23 feet	23 feet
Iodine Species Released to Pool	99.75%Elemental 0.25% Organic	99.85%Elemental 0.15% Organic
Pool DF	Noble gases - 1.0 Iodine - 100	Noble gases - 1.0 Aerosols - infinite Iodine - 200
Release Duration	2 hours	2 hours
Filter Efficiency FHA Outside Containment	95% for all iodine species	No credit for filtration
Filter Efficiency FHA Inside Containment	No credit for filtration	No credit for filtration
Containment locks and equipment hatch	Closed	Open
Dose Conversion Factors	ICRP 30	RADTRAD Table 1.4.3.3-2

Table 4.4-1: Design Input Comparison – Current Licensing Basis vs. AST Design – FHA		
Parameter	CLB Parameter	AST Parameter
Control Room Habitability Envelope (CRHE) Volume	Note 1	226,040 ft^3
CRHE Isolation Time	Note 1	0.5 hours
CRHE Emergency Filtered Intake Air Flow	Note 1	1,265 cfm
CRHE Unfiltered Air Inleakage	Note 1	243 cfm
CRHE Filtered Recirculation Flow Rate	Note 1	19,125 cfm
CRHE Emergency Filter Bed Depth	Note 1	2 in. charcoal
CRHE Emergency Filter Bed Removal Efficiency	Note 1	95%
CRHE Operator Breathing Rates	Note 1	3.5E-04 m^3/sec (0 – 30 days)
CRHE Operator Occupancy Factors	1.0 0-24 hrs 0.6 1-4 days 0.4 4-30 days	1.0 0-24 hrs 0.6 1-4 days 0.4 4-30 days

Note 1: The FHA CLB does not include a CR operator dose evaluation.

4.4.1 Introduction and Background

Two accident scenarios are considered: (1) a refueling accident occurring inside containment and (2) a refueling accident occurring outside containment. This analysis was performed in accordance with the requirements of USNRC Regulatory Guide 1.183.

The postulated FHA inside containment is the dropping of a spent fuel assembly onto the core during refueling which results in damage to the fuel assemblies. There are numerous administrative controls and physical limitations which are imposed to prevent a fuel handling accident from occurring during refueling operations. Nevertheless, an accident sequence has been postulated with the objective of assessing the potential risk to the public health and safety.

It is postulated that a spent fuel assembly is dropped onto the core during refueling resulting in breaching of the fuel rod cladding. As a result of the damage, a portion of the volatile fission gases are released to the water pool covering the core. Subsequently, a fraction of the water soluble gases are absorbed in the pool with the remainder being transported through the water and into the Reactor Building atmosphere through open personnel and equipment hatches. The escaped gases are assumed to be immediately available for release to the environment and dispersed into the atmosphere.

The fuel handling accident outside containment is postulated as the dropping of a spent fuel assembly into the Spent Fuel Pool which results in damage to the fuel assemblies and the release of the volatile gaseous fission products. The conditions and parameters assumed in analyzing the effects and consequences of this accident are identical to those utilized in the FHA inside containment except that the activity released to the environment is released through the Fuel Handling Building Exhaust System. For this evaluation no treatment of the radioactive release by the HEPA and charcoal filters of the Fuel Handling Building Exhaust System is credited. Accordingly, the activity released into the Fuel Handling Building is identical to that presented for the FHA inside containment case.

The NRC approved computer code RADTRAD, endorsed by RG 1.183, is used to calculate the dose to the Control Room operator as well as the doses at the EAB and LPZ.

4.4.2 Source Term

Core inventory based on a power level of 2958 MWt and at time equal to 72 hours after shutdown was determined by running the computer code ORIGIN-S/ARP. Results are provided in as follows. The maximum isotopic activity for all fuel exposures is conservatively utilized in the analysis.

Table 4.4-2: Design Basis Core Inventory @ 72 Hours in Ci/MTU

	Fuel Exposure - MWD/MTU					
Nuclide	35,000	40,000	45,000	50,000	70,000	maximum
I-131	9.44E+05	9.49E+05	9.53E+05	9.54E+05	9.61E+05	9.61E+05
I-132	9.32E+05	9.38E+05	9.39E+05	9.38E+05-	9.33E+05	9.39E+05
I-133	2.31E+05	2.31E+05	2.30E+05	2.30E+05	2.24E+05	2.31E+05
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	1.18E+03	1.19E+03	1.18E+03	1.18E+03	1.16E+03	1.19E+03
Kr-85	1.06E+04	1.17E+04	1.28E+04	1.38E+04	1.70E+04	1.70E+04
Kr-85m	4.65E+00	4.47E+00	4.25E+00	4.07E+00	3.37E+00	4.65E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	2.05E-02	1.96E-02	1.86E-02	1.77E-02	1.43E-02	2.05E-02
Xe-131m	1.51E+04	1.56E+04	1.60E+04	1.64E+04	1.78E+04	1.78E+04
Xe-133	1.97E+06	1.90E+06	1.91E+06	1.89E+06	1.92E+06	1.97E+06
Xe-133m	4.41E+04	4.29E+04	4.43E+04	4.28E+04	4.40E+04	4.43E+04
Xe-135	2.57E+04	2.57E+04	2.55E+04	2.54E+04	2.46E+04	2.57E+04
Xe-135m	1.93E+02	1.94E+02	1.93E+02	1.93E+02	1.90E+02	1.94E+02

Spent fuel source terms are based on reactor core source terms as discussed above, with a conservative factor of 2.0 multiplier to account for the gap fractions of fuel exceeding 54 GWD/MTU burnup with a maximum linear heat generation rate exceeding the 6.3 kW/ft peak rod average power limit to address RG 1.183 footnote 11. This is taken into account in the RADTRAD analysis by multiplying the core inventory gap fractions listed in Table 3 of RG 1.183 by the factor of 2 for input to the RADTRAD computer code.

Group	**Gap Fraction per RG 1.183**	**Gap Fraction input to RADTRAD**
I-131	0.08	0.16
Kr-85	0.10	0.20
Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.24

4.4.3 Mitigation

The radiological consequences of the FHA are actively mitigated by pool scrubbing of the released iodine as shown in Table 4.4-1 and control room filtration (i.e., with an assumed charcoal filtration efficiency of 95%).

As described in Section 9.4.1 of the VCSNS updated FSAR, the control room ventilation system is automatically placed in the emergency mode, with filtration of incoming and recirculated air, following receipt of a SI or high radiation signal from the gaseous activity channel of RM-A1. Although very sensitive and fast acting, RM-A1 is not credited due to lack of redundancy. Since a safety injection will not occur for the FHA, manual initiation is credited in the AST analysis at 30 minutes. With direct communications maintained between the control room and personnel at the fuel handling stations and local radiations monitors (e.g., RM-G6 on the RB refueling bridge, RM-G8 on the FHB refueling bridge, RM-G17A/B on the RB Manipulator Crane, etc) with indication in the control room, this timing assumption (30 minutes) allows adequate time for the refueling personnel to contact the control room and for the control room operators to assess the significance of the event and the need for protective action to ensure Control Room Habitability.

4.4.4 Radiological Transport Modeling

A simplified radiological release model and individual pathway models developed to calculate FHA doses utilizing RADTRAD is shown in Figure 3. Specific model details and the supporting RADTRAD runs are provided in Attachment 10 as Calculation DC00040-102, "Fuel Handling Accidents – AST." Based on the calculated Control Room χ/Qs provided in Table 4.1-3, the conservative values used in the FHA AST dose analyses are taken at the following release points.

- FHA (Inside Containment) RB Nearest Point Intake A
- FHA (Outside Containment) Main Plant Vent Intake B

4.4.5 Results – Control Room Operator Dose

The RADTRAD computer code was used to determine the Control Room operator dose for the FHA. The resultant doses are shown as follows.

FHA CR Operator Dose	Rem TEDE
• FHA Inside Containment =	0.76
• FHA Outside Containment =	0.41

4.4.6 Results – Offsite Doses

The RADTRAD computer code was used to determine the offsite dose. The calculated doses are shown as follows where the EAB dose represents the maximum 2-hour TEDE over the accident period.

FHA EAB Dose	Rem TEDE
• FHA Inside Containment =	4.29
• FHA Outside Containment =	4.29

FHA LPZ Dose	Rem TEDE
• FHA Inside Containment =	1.06
• FHA Outside Containment =	1.06

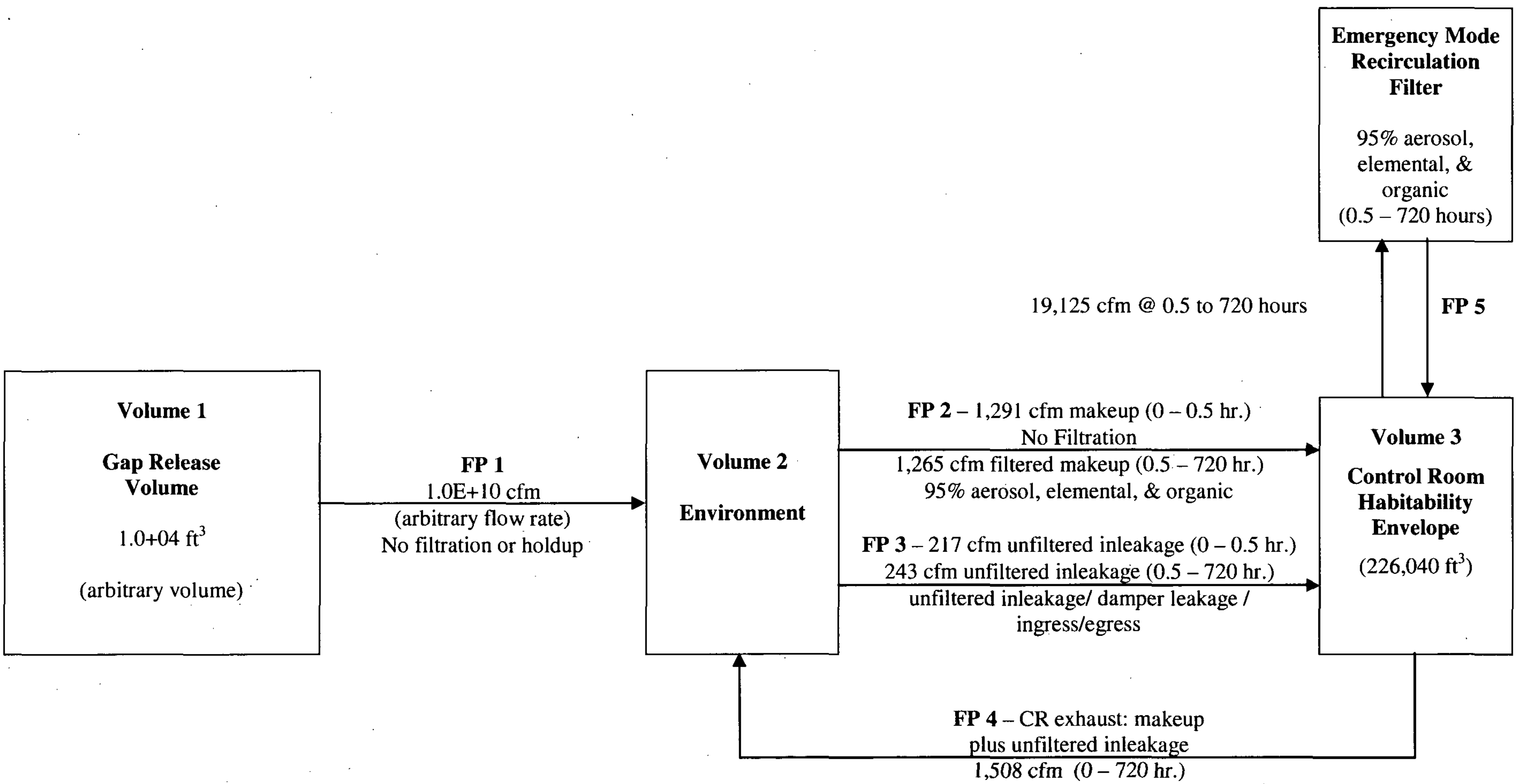
4.4.7 Conclusions

The FHA Control Room operator dose is below the 5 rem TEDE regulatory limit and each offsite dose is below the 6.3 rem TEDE regulatory limit for the inside Containment and inside the Fuel Handling Building (FHB) release pathways.

4.4.8 Summary of Calculation Conservatisms

- Mixing and/or dilution of the radioactivity released from the pool in the FHB or Containment is not considered in this calculation.
- Iodine removal by filtration/charcoal for the FHA in the Fuel Handling Building is not credited.
- For the FHA inside and outside Containment, no credit is taken for ESF filter operation, dilution, or mixing in the Reactor Building or the Fuel Handling Building.
- The FHA release inside Containment is assumed to occur as a ground level release from the nearest point to the CR (Reactor Building nearest point).
- Regulatory Guide 1.183, Table 3, non-LOCA fraction of fission products inventory in the gap is conservatively doubled.
- Isotopes considered are restricted to the 60 isotopes addressed in NUREG/CR-6604. For conservative reasons, additional noble gas isotopes are included in this analysis (Xe-131m, Xe-133m, and Xe-135m).
- The Control Room unanticipated unfiltered inleakage (UUI) value of 41 scfm, based on tracer tests, was increased to 200 scfm. An uncertainty value of 25 percent is conservatively applied to the Control Room filtered makeup flow rate and the unfiltered makeup flow rate which bypasses the damper.

Figure 3: Activity Flow Path Model Developed to Calculate FHA Doses



4.5 Main Steam Line Break

The MSLB accident is postulated as a break of one of the large steam lines outside the containment leading from a SG. A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the MSLB accident are provided in Attachment 10 as Calculation DC00040-099, "Main Steam Line Break - AST". Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the MSLB accident CRHE and offsite doses.

Table 4.5-1 provides a comparison of the design inputs utilized to determine the existing licensing basis MSLB and the proposed AST MSLB accident doses.

Table 4.5-1: Design Input Comparison – Current Licensing Basis vs. AST Design - MSLB		
Parameter	CLB Parameter	AST Parameter
Core Thermal Power Level	2958 MWt	2958 MWt
Reactor Coolant Concentrations	1% Failed Fuel (Defects)	1% Failed Fuel (Defects)
Secondary Coolant Concentrations	Secondary coolant specific activity based on 0.1% µCi/gm DE I-131 secondary activity.	Secondary coolant specific activity based on 0.1% µCi/gm DE I-131 secondary activity.
Radioiodine Chemical Species	NA	97% Elemental 3% Organic
Pre-existing Iodine Spike	60 µCi/gm DE I-131	60 µCi/gm DE I-131
Concurrent Iodine Spike	500 Times the Iodine Equilibrium Release Rate with the Reactor Coolant Activity at the 1.0 µCi/gm DE I-131 assumed for 8 hours.	500 Times the Iodine Equilibrium Release Rate with the Reactor Coolant Activity at the 1.0 µCi/gm DE I-131 assumed for 8 hours.
Iodine Partition Factor for Initial Steam Release from Faulted SG	1.0	1.0
Iodine Partition Factor in Intact SG	100	100
Noble Gas Partition Factor in Intact SG	1.0	1.0
Primary to Secondary Leak Rate to Faulted SG	1 gpm for 8 hour duration	0.35 gpm for 24 hour duration
Primary to Secondary Leak Rate to Intact SGs	None	0.65 gpm for 24 hour duration
Total Release Duration	8 hours	24 hours
Steam Release from Faulted SG	406,000 lbm (0 – 30 min)	406,000 lbm (0 – 30 min)
Steam Release from Intact SG	343,700 lbm (0 – 2 hr) 733,900 lbm (2 – 8 hr)	343,700 lbm (0 – 2 hr) 733,900 lbm (2 – 8 hr) 1,200,000 lbm (8 – 24 hr)
Dose Conversion Factors	ICRP 30	RADTRAD Table 1.4.3.3-2
Control Room Habitability Envelope (CRHE) Volume	Note 1	226,040 ft^3
CRHE Isolation Time	Note 1	2 hours
CRHE Emergency Filtered Intake Air Flow	Note 1	1,265 cfm

Table 4.5-1: Design Input Comparison – Current Licensing Basis vs. AST Design - MSLB		
Parameter	CLB Parameter	AST Parameter
CRHE Unfiltered Air Inleakage	Note 1	243 cfm
CRHE Filtered Recirculation Flow Rate	Note 1	19,125 cfm
CRHE Emergency Filter Bed Depth	Note 1	2 in. charcoal
CRHE Emergency Filter Bed Removal Efficiency	Note 1	95%
CRHE Operator Breathing Rates	Note 1	3.5E-04 m^3/sec
CRHE Operator Occupancy Factors	Note 1	1.0, 0-24 hrs 0.6, 1-4 days 0.4, 4-30 days

Note 1: CLB for the MSLB did not include a dose evaluation for CRHE.

4.5.1 Introduction and Background

The MSLB accident is postulated as a break of one of the large steam lines outside the containment leading from a SG. For the two intact SGs loops, primary-to-secondary coolant leakage transfers activity into the secondary coolant. This makes it available for release into the environment via steaming through the SG PORV or SSVs. For the coolant loop with the broken steam line (i.e., faulted SG), primary-to-secondary coolant leakage is assumed to be released from the RCS directly into the environment without passing through any secondary coolant. Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the MSLB accident CRHE and offsite doses.

4.5.2 Source Term

No fuel melt or fuel clad breach is postulated for the MSLB event at VCSNS. Consistent with RG 1.183 Appendix E, Section 2.2, if no or minimal fuel damage is postulated for the limiting event, the activity release should be the maximum allowed by technical specification for two cases of iodine spiking (1) maximum pre-existing (pre-accident) iodine spike and (2) maximum concurrent iodine spike.

The reactor coolant DE I-131 values are based on VCSNS Technical Specification limit of 1.0 μCi/gm DE I-131 for the concurrent spike scenario and 60 μCi/gm DE I-131 for the pre-existing spike scenario. The concurrent spike activity is based on an activity release rate from the fuel to the reactor coolant of 500 times the iodine equilibrium release rate consistent with the limiting condition for operation with the reactor coolant activity at the Technical Specification limit of 1.0 μCi/gm DE I-131.

The resulting activity release to the environment is based on the following:

- Activity in secondary coolant released from faulted SG.
- Activity in reactor coolant released due to 0.35 gpm tube leak in faulted SG (0 to 24 hours).
- Activity in reactor coolant released due to 0.65 gpm tube leak in intact SGs (0 to 24 hours).
- Activity in secondary coolant released from intact SGs.

4.5.3 Mitigation

Activity Removal Mechanisms in Containment
The design basis MSLB at VCSNS releases activity directly into the RCS, therefore no plateout or other activity deposition is credited.

Decay Credited - MSLB
Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the RADTRAD NIF file. The RADTRAD decay option is used.

Control Room Filtration
Credit is taken for control room filtration (i.e., with an assumed charcoal filtration efficiency of 95%). As described in Section 9.4.1 of the VCSNS updated FSAR, the control room ventilation system is automatically placed in the emergency mode, with filtration of incoming and recirculated air, following receipt of a SI or high radiation signal from the gaseous activity channel of RM-A1. Although very sensitive and fast acting, RM-A1 is not credited due to lack of redundancy. For a large MSLB, SI actuation would occur on low RCS pressure, low steam line pressure, or high RB pressure (i.e., for breaks inside containment) within the first minute. However, as break size decreases, the time to SI actuation will increase. Spectrum analyses for breaks outside containment ranging in break size from 0.2 to 4.6 ft^2 predict SI actuation within 10 minutes; and, FSAR analyses (Section 15.2.13) for accidental depressurization of the main steam system (i.e., due to a stuck open steam dump or safety valve) also show SI actuation on low RCS pressure within 10 minutes. Based on these analyzed times, the control room is conservatively assumed to enter the emergency recirculation mode, via automatic or manual initiation, within 30 minutes for the AST MSLB analysis. This timing assumption bounds expected variations due to break size effects and is judged to be adequate for any MSLB accident that might challenge CR habitability.

Release of Activity to the Environment
Per Technical Specification Bases 3/4.4.5, a total primary to secondary leakage for all three steam generators of 1 gpm is used in the evaluation of design basis accidents. Prior to the event this leakage is assumed to be distributed throughout the three steam generators. Recognizing that an extended plant cooldown may be required under natural circulation conditions, the MSLB is conservatively analyzed using 24 hours for the cooldown time. The activity associated with the 1 gpm leak is assumed to be released to the environment via the faulted steam generator at a rate of 0.35 gpm for the 24 hour duration of the event with no credit taken for any reduction or mitigation, i.e. a partition factor of 1.0. This is conservative in that the actual maximum value allowed by Technical Specification 3.4.6.2.c for any one SG is 150 gpd (~ 0.104 gpm). The remaining 0.65 gpm is released to the environment via the two intact steam generators for the 24 hour duration of the event crediting a partition factor of 100.

4.5.4 Radiological Transport Modeling

A simplified radiological release model and individual pathway models developed to calculate MSLB doses utilizing RADTRAD is shown in Figure 4. Specific model details and the supporting RADTRAD runs are provided in Attachment 10 as Calculation DC00040-099, "Main Steam Line Break – AST." Based on the calculated Control Room χ/Qs provided in Table 4.1-3, the conservative values used in the MSLB AST dose analyses are taken at the following release point.

- MSLB — MS SSV A (Relief's B, C, D, E), Intake B

4.5.5 Results – Control Room Operator Dose
The RADTRAD computer code was used to determine the Control Room operator dose for the MSLB. The resultant doses are shown as follows.

MSLB CR Operator Dose	Rem TEDE
• Pre-existing Iodine Spike =	1.15
• Concurrent Iodine Spike =	0.37

4.5.6 Results – Offsite Doses

The RADTRAD computer code was used to determine the offsite dose. The calculated doses are shown as follows where the EAB dose represents the maximum 2-hour TEDE over the accident period.

MSLB EAB Dose	Rem TEDE
• Pre-existing Iodine Spike =	1.96
• Concurrent Iodine Spike =	0.78

MSLB LPZ Dose	Rem TEDE
• Pre-existing Iodine Spike =	0.13
• Concurrent Iodine Spike =	0.16

4.5.7 Conclusions

The calculated MSLB Control Room operator doses for the pre-existing and concurrent iodine spike cases are below the 5 rem TEDE regulatory limit.

The calculated MSLB offsite (EAB and LPZ) doses for the pre-existing iodine spike case are below the 25 rem TEDE regulatory limit. The calculated MSLB offsite doses for the concurrent iodine spike case are below the 2.5 rem TEDE regulatory limit.

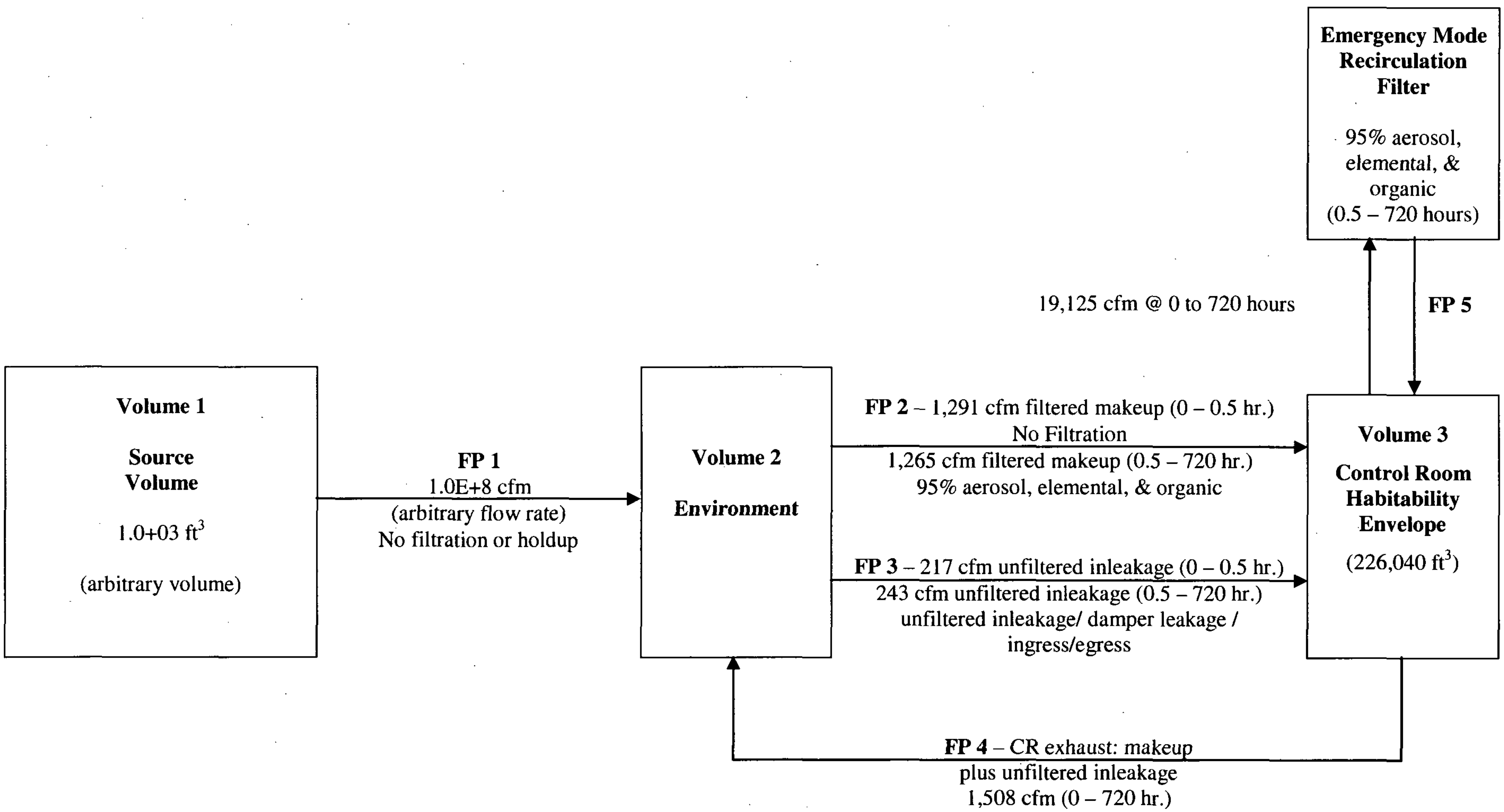
4.5.8 Summary of Calculation Conservatisms

- The reactor coolant system (RCS) activity conservatively remains constant throughout the pre-existing Iodine Spike MSLB event. For the Concurrent Iodine Spike MSLB event, a similar assumption is made with the exception that the iodine activity increases during the 8 hours of the transient as a result of release from the defective fuel at a rate of 500 times the iodine equilibrium appearance rate.

- Isotopes considered are restricted to the 60 isotopes addressed in NUREG/CR-6604. For conservative reasons, additional isotopes are included in this analysis (Xe-131m, Xe-133m, Xe-135m, Rb-88, Br-83, Br-84, and Cs-138).

- The Control Room unanticipated unfiltered inleakage (UUI) value of 41 scfm, based on tracer tests, was increased to 200 scfm. An uncertainty value of 25 percent is conservatively applied to the Control Room filtered makeup flow rate and the unfiltered makeup flow rate which bypasses the damper.

Figure 4 Activity Flow Path Model Developed to Calculate MSLB Doses



4.6 Steam Generator Tube Rupture Accident

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the SGTR accident are provided in Attachment 10 as Calculation DC00040-098, "Steam Generator Tube Rupture - AST". Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the SGTR accident CRHE and offsite doses.

Table 4.6-1 provides a comparison of the design inputs utilized to determine the existing licensing basis SGTR and the proposed AST SGTR accident doses.

Table 4.6-1: Design Input Comparison – Current Licensing Basis vs. AST Design - SGTR

Parameter	CLB Parameter	AST Parameter
Core Thermal Power Level	2958 MWt	2958 MWt
Reactor Coolant Concentrations	1% Failed Fuel (Defects)	1% Failed Fuel (Defects)
Secondary Coolant Concentrations	Secondary coolant specific activity based on 0.1% μCi/gm DE I-131 secondary activity.	Secondary coolant specific activity based on 0.1% μCi/gm DE I-131 secondary activity.
Radioiodine Chemical Species	NA	97% Elemental 3% Organic
Pre-existing Iodine Spike	60 μCi/gm DE I-131	60 μCi/gm DE I-131
Concurrent Iodine Spike	500 Times the Iodine Equilibrium Release Rate with the Reactor Coolant Activity at the 1.0 μCi/gm DE I-131 assumed for 8 hours	335 Times the Iodine Equilibrium Release Rate with the Reactor Coolant Activity at the 1.0 μCi/gm DE I-131 assumed for 8 hours
Iodine Partition Factor for Initial Steam Release from Faulted SG	10	1.0
Iodine Partition Factor in Intact SG	10	100
Noble Gas Partition Factor in Intact SG	1.0	1.0
SG tube leak prior to and during the accident assumed all into the two intact SGs	1.0 gpm for 8 hour duration	1.0 gpm for 24 hour duration
Steam Release from Faulted SG	56,800 lbm (0 – 30 min.)	56,800 lbm (0 – 30 min)
Steam Release from Intact SG	381,400 lbm (0 – 2 hr) 924,900 lbm (2 – 8 hr)	381,400 lbm (0 – 2 hr) 924,900 lbm (2 – 8 hr) 1,200,000 lbm (8 – 24 hr)
Reactor Coolant Release to Faulted SG from Broken Tube	92,900 lbm (0 – 30 min)	19,400 lbm (0 – 385 sec.) 73,500 lbm (385 sec. – 30 min) or 92,900 lbm (0 – 30 min)
Time at Which Primary to Secondary Leakage from Broken Tube is Assumed to be Terminated	30 minutes	30 minutes
Total Release Duration	8 hours	24 hours
Total Release Duration	8 hours	24 hours

Table 4.6-1: Design Input Comparison – Current Licensing Basis vs. AST Design - SGTR		
Parameter	CLB Parameter	AST Parameter
Dose Conversion Factors	ICRP 30	RADTRAD Table 1.4.3.3-2
Control Room Habitability Envelope (CRHE) Volume	Note 1	226,040 ft^3
CRHE Isolation Time	Note 1	2 hr
CRHE Emergency Filtered Intake Air Flow	Note 1	1,265 cfm
CRHE Unfiltered Air Inleakage	Note 1	243 cfm
CRHE Filtered Recirculation Flow Rate	Note 1	19,125 cfm
CRHE Emergency Filter Bed Depth	Note 1	2 in. charcoal
CRHE Emergency Filter Bed Removal Efficiency	Note 1	95%
CRHE Operator Breathing Rates	Note 1	3.5E-04 m^3/sec
CRHE Operator Occupancy Factors	Note 1	1.0, 0-24 hrs 0.6, 1-4 days 0.4, 4-30 days

Note 1: CLB for the SGTR did not include a dose evaluation for CRHE.

4.6.1 Introduction and Background

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power operated relief valves.

4.6.2 Source Term

No fuel melt or fuel clad breach is postulated for the SGTR event at VCSNS. Consistent with RG 1.183 Appendix E, Section 2.2, if no or minimal fuel damage is postulated for the limiting event, the activity release should be the maximum allowed by technical specification for two cases of iodine spiking (1) maximum pre-existing (pre-accident) iodine spike and (2) maximum concurrent iodine spike.

The reactor coolant DE I-131 values are based on VCSNS Technical Specification limit of 1.0 μCi/gm DE I-131 for the concurrent spike scenario and 60 μCi/gm DE I-131 for the pre-existing spike scenario. The concurrent spike activity is based on an activity release rate from the fuel to the reactor coolant of 335 times the iodine equilibrium release rate consistent with the limiting condition for operation with the reactor coolant activity at the Technical Specification limit of 1.0 μCi/gm DE I-131.

4.6.3 Mitigation

Activity Removal Mechanisms in Containment
The design basis SGTR at VCSNS releases activity directly from the RCS to the steam generators, therefore no plateout or other activity deposition in containment is credited.

Decay Credited - SGTR
Decay of radioactivity is credited in all compartments, prior to release. This is implemented in RADTRAD using the half-lives in the RADTRAD NIF file. The RADTRAD decay option is used.

Control Room Filtration
Credit is taken for control room filtration (i.e., with an assumed charcoal filtration efficiency of 95%). As described in Section 9.4.1 of the VCSNS updated FSAR, the control room ventilation system is automatically placed in the emergency mode, with filtration of incoming and recirculated air, following receipt of a SI or high radiation signal from the gaseous activity channel of RM-A1. Although very sensitive and fast acting, RM-A1 is not credited due to lack of redundancy. The CLB analysis (FSAR Section 15.4.3) assumes the double-ended rupture of a single tube and predicts a reactor trip and SI actuation on low RCS pressure in approximately 6.5 minutes. The FSAR analysis does not examine break size affects as primary to secondary size leakage and consequently doses are maximized for a large double-ended rupture. A timely automatic SI actuation is, however, expected to occur for large to medium size SGTRs; and, the operator is also instructed to initiate an SI (per Reference 12) for any tube leak that is outside of the capacity of the normal charging system. For the AST SGTR analysis, 30 minutes is conservatively assumed for the time after event initiation to initiate the emergency mode of operation. This timing assumption bounds the calculated times from the FSAR analyses and allows for substantial variation to cover break size effects, including the need to manually initiate SI.

Release of Activity to the Environment
Table 4.6-1 provides the mass releases. The SGTR activity release to the environment is based on the following:

Activity in reactor coolant released due to tube rupture in the faulted SG: A portion of the primary to secondary leakage through the SGTR is assumed to flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. The leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG and enter the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water is credited. All leakage that does not immediately flash is assumed to mix with the bulk water. The radioactivity within the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 is assumed for the leakage from the bulk water.

Activity in reactor coolant released due to SG tube leak: Per Technical Specification Bases 3/4.4.5, a total primary to secondary leakage for all three steam generators of 1 gpm is used in the evaluation of design basis accidents. Prior to the event this leakage is assumed to be distributed throughout the three steam generators. Recognizing that an extended plant cooldown may be required under natural circulation conditions, the SGTR is conservatively analyzed using 24 hours for the cooldown time. The activity associated with the 1 gpm leak is conservatively assumed to be released to the environment via the two intact steam generators for the 24 hour duration of the event. A partition coefficient for iodine of 100 is assumed for this leakage. This is conservative in that the actual maximum value allowed by Technical Specification 3.4.6.2.c for any one SG is 150 gpd (~ 0.104 gpm).

Initial activity in secondary coolant released from the faulted and intact SGs: The initial secondary coolant activity is released to the environment with a partition factor of 100.

Noble gases have a partition factor of 1 for all release paths.

4.6.4 Radiological Transport Modeling

A simplified radiological release model and individual pathway models developed to calculate SGTR doses utilizing RADTRAD is shown in Figure 5. Specific model details and the supporting RADTRAD runs are provided in Attachment 10 as Calculation DC00040-098, "Steam Generator Tube Rupture – AST." Based on the calculated Control Room χ/Qs provided in Table 4.1-3, the conservative values used in the SGTR AST dose analyses are taken at the following release point.

- SGTR MS SSV A (Relief's B, C, D, E), Intake B

4.6.5 Results – Control Room Operator Dose

The RADTRAD computer code was used to determine the Control Room operator dose for the SGTR. The resultant doses are shown as follows:

SGTR CR Operator Dose	Rem TEDE
• Pre-existing Iodine Spike =	1.18
• Concurrent Iodine Spike =	0.37

4.6.6 Results – Offsite Doses

The RADTRAD computer code was used to determine the offsite dose. The calculated doses are shown as follows where the EAB dose represents the maximum 2-hour TEDE over the accident period.

SGTR EAB Dose	Rem TEDE
• Pre-existing Iodine Spike =	2.08
• Concurrent Iodine Spike =	0.71

SGTR LPZ Dose	Rem TEDE
• Pre-existing Iodine Spike =	0.12
• Concurrent Iodine Spike	0.047

4.6.7 Conclusions

The calculated SGTR Control Room operator doses for the pre-existing and concurrent iodine spike cases are below the 5 rem TEDE regulatory limit.

The calculated SGTR offsite (EAB and LPZ) doses for the pre-existing iodine spike case are below the 25 rem TEDE regulatory limit. The calculated SGTR offsite doses for the concurrent iodine spike case are below the 2.5 rem TEDE regulatory limit.

4.6.8 Summary of Calculation Conservatisms

- Regulatory Guide 1.183, Table 3, non-LOCA fraction of fission products inventory in the gap are conservatively doubled.

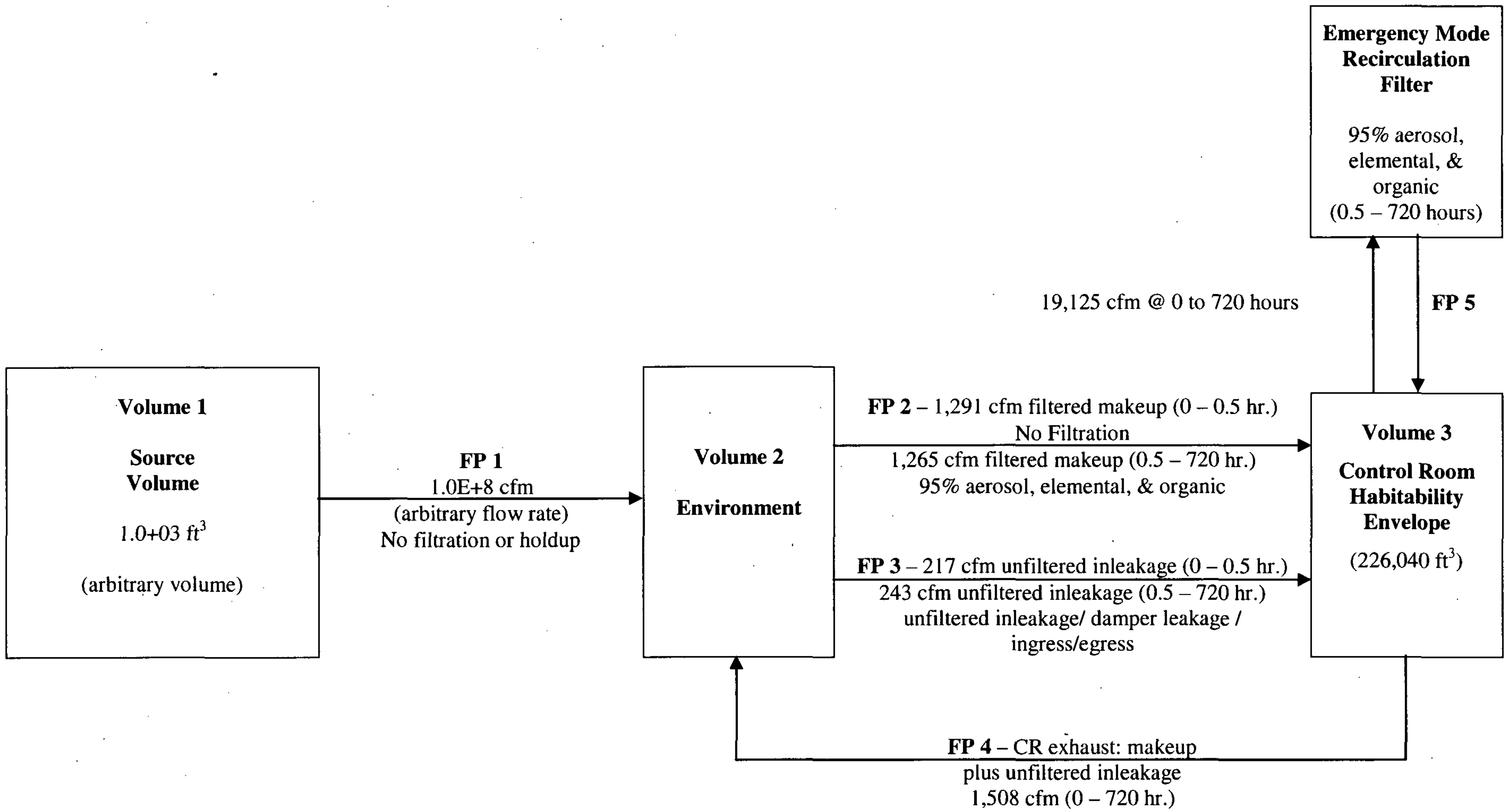
- Rb-86, Cs-134, Cs-136, and Cs-137 default core inventories from Table 1.4.3.2-2 of NUREG/CR-6604, corrected to a core thermal power of 2958 MWt, are included in this analysis. In addition, VCSNS noble gas and iodine core inventories and the corrected core inventories Table 1.4.3.2-2 of NUREG/CR-6604 were compared and the larger of the two concentrations are used in this analysis.

- The Control Room unanticipated unfiltered inleakage (UUI) value of 41 scfm, based on tracer tests, was increased to 200 scfm. An uncertainty value of 25 percent is conservatively applied to the Control Room filtered makeup flow rate and the unfiltered makeup flow rate which bypasses the damper.

- The flashed break flow from the ruptured SG tube prior to reactor trip (< 170 seconds) is released to the condenser with an iodine partition factor of 100. The SGTR analysis conservatively does not credit this reduction and assumes the entire flashed break flow is released to the environment with no reduction.

- A total SG tube leak of 1 gpm is assumed for the duration of the accident into the two intact SGs even though the maximum value in any one steam generator is limited to 150 gpd (~ 0.104 gpm) in accordance with Technical Specification 3.4.6.2.c.

Figure 5: RADTRAD Model Developed to Calculate SGTR Accident Doses



4.7 Reactor Coolant Pump Locked Rotor Accident

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the Locked Rotor accident are provided in Attachment 10 as Calculation DC00040-100, "Reactor Coolant Pump Locked Rotor - AST". Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the Locked Rotor accident CRHE and offsite doses.

Table 4.7-1 provides a comparison of the design inputs utilized to determine the existing licensing basis Locked Rotor and the proposed AST Locked Rotor accident doses.

Table 4.7-1: Design Input Comparison – Current Licensing Basis vs. AST Design – LRA		
Parameter	CLB Parameter	AST Parameter
Core Thermal Power Level	2958 MWt	2958 MWt
Percent of Fuel Rods in Core Failed	15%	15%
Fraction of Fission Products in the Gap Available for Release	Iodine 0.10 Kr-85 0.30 Other Noble Gases 0.10	I-131 0.16 Other Halogens 0.10 Kr-85 0.20 Other Noble Gases 0.10 Alkali Metals 0.24
Core Radial Peaking Factor	1.0	1.7
Defective Fuel Prior to Event	1%	1%
Steam Released from the 3 SGs	447,900 lbm (0 – 2 hr) 868,300 lbm 02 – 8 hr)	447,900 lbm (0 – 2 hr) 868,300 lbm (2 – 8 hr) 1,2000,000 lbm (8 – 24 hr)
Mass of Reactor Coolant System	400,000 lbm	400,000 lbm
Total Water Mass of the 3 SGs	340,000 lbm	340,000 lbm
SG Blowdown Flow Rate	12,756 lb/hr	12,756 lb/hr
Chemical Form of Radioiodine Released from the Damaged Fuel	Not used	Particulate 95% Elemental 4.85% Organic 0.15%
Chemical Form of Radioiodine Released from SGs to Environs	Not used	Elemental 97% Organic 3%
Dose Conversion Factors	ICRP 30	RADTRAD Table 1.4.3.3-2
Control Room Habitability Envelope (CRHE) Total Volume	Note 1	226,040 ft^3
CRHE Isolation Time	Note 1	2 hours
CRHE Emergency Filtered Intake Air Flow	Note 1	1,265 cfm
CRHE Unfiltered Air Inleakage	Note 1	243 cfm
CRHE Filtered Recirculation Flow Rate	Note 1	19,125 cfm
CRHE Emergency Filter Bed Depth	Note 1	2 in. charcoal
CRHE Emergency Filter Bed Removal Efficiency	Note 1	95%
CRHE Operator Breathing Rates	3.5E-04 m^3/sec	3.5E-04 m^3/sec
CRHE Operator Occupancy Factors	1.0, 0-24 hrs 0.6, 1-4 days 0.4, 4-30 days	1.0, 0-24 hrs 0.6, 1-4 days 0.4, 4-30 days

Note 1: The CRHE dose was not calculated in the CLB analysis.

4.7.1 Introduction and Background

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal. Following initiation of the reactor trip heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power operated relief valves, and opens the pressurizer safety valves, in that sequence. The three power operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis.

4.7.2 Source Term

The core inventory of the radionuclide groups required for non-LOCA events, based on RG 1.183, at 102% of the core thermal power are listed in Table 4.7-2.

Table 4.7-2: Core Activity @ T = 0 Hours

Isotope	Activity (Ci)	Isotope	Activity (Ci)
Kr-85	8.30E+05	Xe-133	1.70E+08
Kr-85m	2.72E+07	Xe-135	3.70E+07
Kr-87	4.96E+07	Cs-134	1.01E+07
Kr-88	6.71E+07	Cs-136	3.08E+06
Rb-86	4.43E+04	Cs-137	5.66E+06
I-131	8.20E+07		
I-132	1.20E+08		
I-133	1.68E+08		
I-134	1.80E+08		
I-135	1.54E+08		

Reactor coolant equilibrium fission and corrosion product specific activity, based on 1% fuel defects, are summarized in Table 4.7-3 for the applicable isotopes.

Table 4.7-3: Reactor Coolant Fission and Corrosion Product Specific Activity			
Isotope	Activity (µCi/gm)	Isotope	Activity (µCi/gm)
Kr-85	7.6E+00	Xe-133	2.9E+02
Kr-85m	1.8E+00	Xe-135	8.6E+00
Kr-87	1.1E+00	Cs-134	4.4E+00
Kr-88	3.2E+00	Cs-136	4.5E+00
Rb-86	3.6E-02	Cs-137	2.1E+00
I-131	3.0E+00		
I-132	3.1E+00		
I-133	4.6E+00		
I-134	6.0E-01		
I-135	2.4E+00		

Fuel source terms are based on reactor core source terms as discussed above, with a conservative factor of 2.0 multiplier to account for the gap fractions of fuel exceeding 54 GWD/MTU burnup with a maximum linear heat generation rate exceeding the 6.3 kW/ft peak rod average power limit to address RG 1.183 footnote 11. This is taken into account in the RADTRAD analysis by multiplying the core inventory gap fractions listed in Table 3 of RG 1.183 by the factor of 2 for input to the RADTRAD computer code.

Group	**Gap Fraction per RG 1.183**	**Gap Fraction Input to RADTRAD**
I-131	0.08	0.16
Kr-85	0.10	0.20
Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.24

4.7.3 Mitigation

The primary-to-secondary leakage is assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100° C (212° F). The release of radioactivity is assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated. In less than twenty four hours after the accident, the Residual Heat Removal System starts operation to cool down the plant. At this point, no steam and activity are released to the environment. All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation. The radioactivity in the secondary water is assumed to become a vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine and the alkali metals of 100 is assumed.

No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip. Credit is taken for isolation of the CR via initiation of its emergency mode of operation filtered makeup (i.e., with an assumed charcoal filtration efficiency of 95%) at 0 hours after the accident.

Credit is taken for control room filtration (i.e., with an assumed charcoal filtration efficiency of 95%). As described in Section 9.4.1 of the VCSNS updated FSAR, the control room ventilation system is automatically placed in the emergency mode, with filtration of incoming and recirculated air, following receipt of a SI or high radiation signal from the gaseous activity channel of RM-A1. Although very sensitive and fast acting, RM-A1 is not credited due to lack of redundancy; and, without an additional failure, a SI actuation is also not anticipated for the LRA. Fuel failure would, however, be indicated via elevation radiation levels and abnormal radiation readings on a number of radiation monitors. Consequently, for the AST LRA analysis, 2 hours from event initiation is conservatively assumed for manual initiation of the emergency mode of operation. This timing assumption provides adequate time for recognition of an ongoing atmospheric relief that requires the use of protective action to ensure Control Room Habitability.

4.7.4 Radiological Transport Modeling

A simplified radiological release model and individual pathway models developed to calculate RCP LRA doses utilizing RADTRAD is shown in Figure 6. Specific model details and the supporting RADTRAD runs are provided in Attachment 10 as Calculation DC00040-100, "Reactor Coolant Pump Locked Rotor – AST." Based on the calculated Control Room χ/Qs provided in Table 4.1-3, the conservative values used in the MSLB AST dose analyses are taken at the following release point.

- LRA — MS SSV A (Relief's B, C, D, E), Intake B

4.7.5 Results – Control Room Operator Dose

The RADTRAD computer code was used to determine the Control Room operator dose for the LRA. The resultant dose is shown as follows.

LRA CR Operator Dose	2.43 Rem TEDE

4.7.6 Results – Offsite Doses

The RADTRAD computer code was used to determine the offsite dose. The calculated doses are shown as follows where the EAB dose represents the maximum 2-hour TEDE over the accident period.

LRA EAB Dose	2.20 Rem TEDE
LRA LPZ Dose	0.45 Rem TEDE

4.7.7 Conclusions

The LRA Control Room operator dose for the 1 gpm primary to secondary release case is below the 5 rem TEDE regulatory limit.

The LRA offsite (EAB and LPZ) doses are below the 2.5 rem TEDE regulatory limit at the EAB and the LPZ.

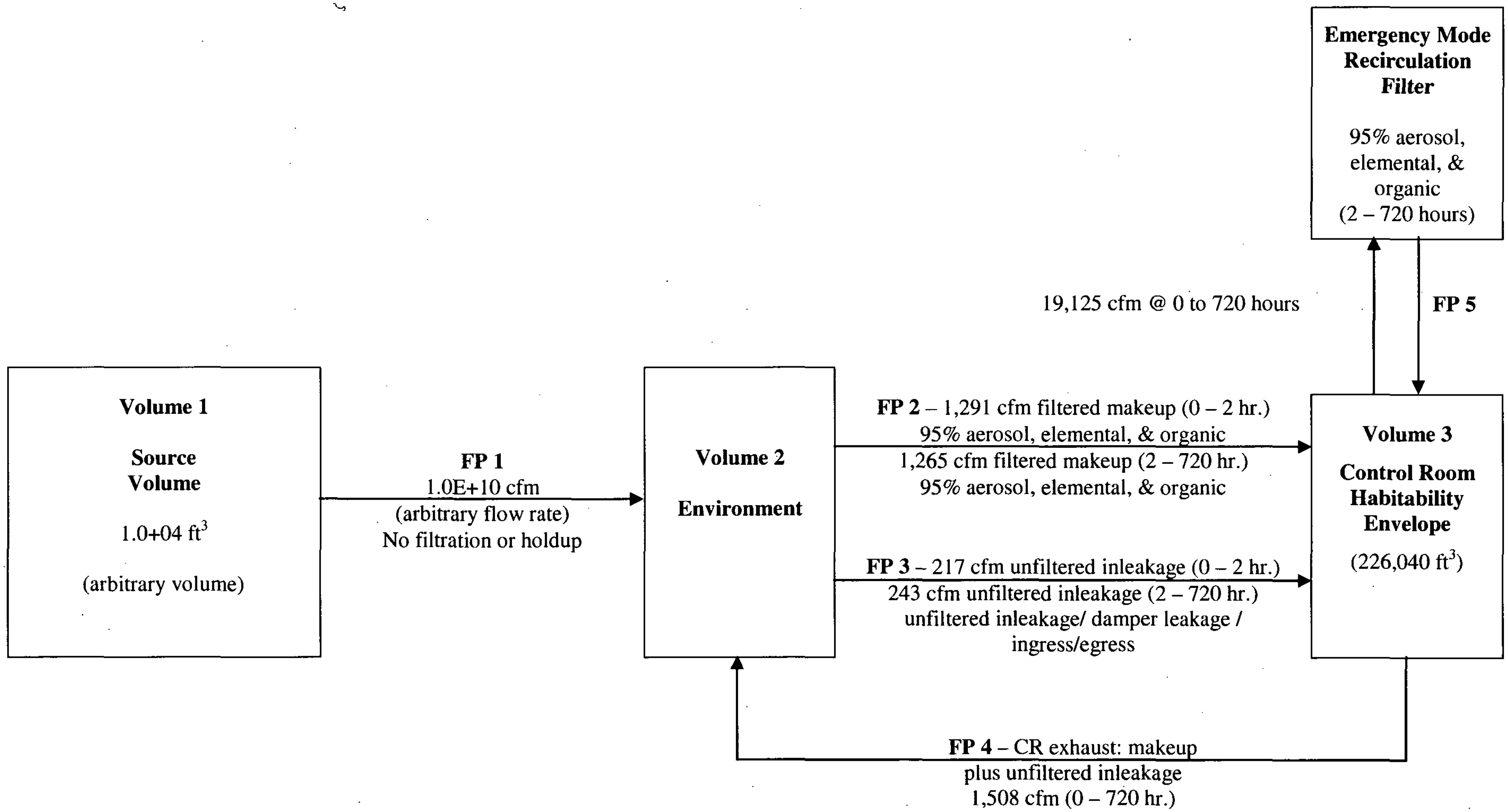
4.7.8 Summary of Calculation Conservatisms

- Regulatory Guide 1.183, Table 3, non-LOCA fraction of fission products inventory in the gap is conservatively doubled.

- Rb-86, Cs-134, Cs-136, and Cs-137 default core inventories from Table 1.4.3.2-2 of NUREG/CR-6604, corrected to a core thermal power of 2958 MWt, are included in this analysis. In addition, VCSNS noble gas and iodine core inventories and the corrected core inventories Table 1.4.3.2-2 of NUREG/CR-6604 were compared and the larger of the two concentrations are used in this analysis.

- The Control Room unanticipated unfiltered inleakage (UUI) value of 41 scfm, based on tracer tests, was increased to 200 scfm. An uncertainty value of 25 percent is conservatively applied to the Control Room filtered makeup flow rate and the unfiltered makeup flow rate which bypasses the damper.

Figure 6: RADTRAD Model Developed to Calculate RCP LRA Doses



4.8 Control Rod Ejection Accident

A detailed discussion of the design input parameters, assumptions, methodology, analysis, and results supporting the Control Rod Ejection accident are provided in Attachment 10 as Calculation DC00040-101, "Rod Ejection - AST". Attachment 3 provides a matrix which compares the RG 1.183 regulatory position with the parameters and methodologies utilized to calculate the Control Rod Ejection accident CRHE and offsite doses.

Table 4.8-1 provides a comparison of the design inputs utilized to determine the existing licensing basis Control Rod Ejection and the proposed AST Control Rod Ejection accident doses.

Table 4.8-1: Design Input Comparison – Current Licensing Basis vs. AST Design – CREA

Parameter	CLB Parameter	AST Parameter
Core Thermal Power Level	2958 MWt	2958 MWt
Percent of Fuel Rods in Core Failed	10%	10%
Fraction of Fission Products in the Gap Available for Release from Failed Rods	Iodine 0.10 Noble Gases 0.10	Halogens 0.20 Noble Gases 0.20 Alkali Metals 0.24
Core Radial Peaking Factor	1.0	1.7
% of Fuel Melted	0.25%	0.25%
Fraction of Fission Products in the Melted Fuel for Containment Release Pathway	Iodine 0.5 Noble Gases 1.0	Halogens 0.25 Noble Gases 1.0 Alkali Metals 0.30
Fraction of Fission Products in the Melted Fuel for Steam Generator Release Pathway	Iodine 0.5 Noble Gases 1.0	Halogens 0.50 Noble Gases 1.0 Alkali Metals 0.50
Defective Fuel Prior to Event	1%	1%
Steam Released from the 3 SGs	33,000 lbm (0 – 150 sec)	447,900 lbm (0 – 2 hr) 868,300 lbm (2 – 8 hr) 1,200,000 lbm (8 – 24 hr)
Mass of Reactor Coolant System	400,000 lbm	400,000 lbm
Total water mass of the 3 SGs	340,000 lbm	340,000 lbm
SG Blowdown Flow Rate	12,756 lb/hr	12,756 lb/hr
Containment Natural Deposition of Aerosols	None	10% Power's Model
Secondary Coolant Partition Factor in SG	Iodine 100	Iodine 100 Alkali Metals 100
Containment Free Volume	1,840,000 ft^3	1,840,000 ft^3
Containment Leak Rate	0.2%/day (0 – 24 hr) 0.1%/day (24 – 720 hr)	0.2%/day (0 – 24 hr) 0.1%/day (24 – 720 hr)
Chemical Form of Radioiodine Released from the Damaged Fuel	Not Used	Particulate 95% Elemental 4.85% Organic 0.15%
Chemical Form of Radioiodine Released from SGs to Environs	Not Used	Elemental 97% Organic 3%
Dose Conversion Factors	ICRP 30	RADTRAD Table 1.4.3.3-2
Control Room Habitability Envelope (CRHE) Total Volume	Note 1	226,040 ft^3

Table 4.8-1: Design Input Comparison – Current Licensing Basis vs. AST Design – CREA		
Parameter	CLB Parameter	AST Parameter
CRHE Isolation Time	Note 1	0.5 hours Containment Release 2 hours SG Release
CRHE Emergency Filtered Intake Air Flow	Note 1	1,265 cfm
CRHE Unfiltered Air Inleakage	Note 1	243 cfm
CRHE Filtered Control Room Recirculation Flow Rate	Note 1	19,125 cfm
CRHE Emergency Filter Bed Depth	Note 1	2 in. charcoal
CRHE Emergency Filter Bed Removal Efficiency	Note 1	95%
CRHE Operator Breathing Rates	3.5E-04 m^3/sec	3.5E-04 m^3/sec
CRHE Operator Occupancy Factors	1.0, 0-24 hrs 0.6, 1-4 days 0.4, 4-30 days	1.0, 0-24 hrs 0.6, 1-4 days 0.4, 4-30 days

Note 1: The CRHE dose was not calculated in the CLB analysis.

4.8.1 Introduction and Background

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Following the postulated rod ejection accident, two separate scenarios for activity release to the environment are considered: (1) a breach of the RPV and containment leakage for 30 days or (2) primary-to secondary leakage and secondary steaming via the relief valves until cold shutdown.

4.8.2 Source Term

The core inventory of the radionuclide groups required for non-LOCA events, based on RG 1.183, at 102% of the core thermal power are listed in Table 4.8-2.

Table 4.8-2: Core Activity @ T = 0 Hours			
Isotope	Activity (Ci)	Isotope	Activity (Ci)
Kr-85	8.30E+05	Xe-133	1.70E+08
Kr-85m	2.72E+07	Xe-135	3.70E+07
Kr-87	4.96E+07	Cs-134	1.01E+07
Kr-88	6.71E+07	Cs-136	3.08E+06
Rb-86	4.43E+04	Cs-137	5.66E+06
I-131	8.20E+07		
I-132	1.20E+08		
I-133	1.68E+08		
I-134	1.80E+08		
I-135	1.54E+08		

Reactor coolant equilibrium fission and corrosion product specific activity, based on 1% fuel defects, are summarized in Table 4.8-3 for the applicable isotopes.

Table 4.8-3: Reactor Coolant Fission and Corrosion Product Specific Activity at Equilibrium			
Isotope	Activity (μCi/gm)	Isotope	Activity (μCi/gm)
Kr-85	7.6E+00	Xe-133	2.9E+02
Kr-85m	1.8E+00	Xe-135	8.6E+00
Kr-87	1.1E+00	Cs-134	4.4E+00
Kr-88	3.2E+00	Cs-136	4.5E+00
Rb-86	3.6E-02	Cs-137	2.1E+00
I-131	3.0E+00		
I-132	3.1E+00		
I-133	4.6E+00		
I-134	6.0E-01		
I-135	2.4E+00		

Fuel source terms are based on reactor core source terms as discussed above, with a conservative factor of 2.0 multiplier to account for the gap fractions of fuel exceeding 54 GWD/MTU burnup with a maximum linear heat generation rate exceeding the 6.3 kW/ft peak rod average power limit to address RG 1.183 footnote 11. This is taken into account in the RADTRAD analysis by multiplying the core inventory gap fractions listed in Table 3 of RG 1.183 by the factor of 2 for input to the RADTRAD computer code.

Group	Gap Fraction per RG 1.183	Gap Fraction Input to RADTRAD
Noble Gases	0.10	0.20
Halogens	0.10	0.20
Alkali Metals	0.12	0.24

4.8.3 Mitigation

For the Containment release pathway, leakage is assumed at the Technical Specifications leak rate for peak accident pressure for the first 24 hours and 50% of this leak rate for the remaining duration of the accident. Credit is taken for natural deposition removal of aerosols utilizing the RADTRAD 10% Power's model. No credit is assumed in the containment for removal of released activity by the containment sprays or the internal containment recirculation HEPA filters.

For the SG release pathway, per Technical Specification Bases 3/4.4.5, a total primary to secondary leakage for all three steam generators of 1 gpm is used in the evaluation of design basis accidents. Prior to the event this leakage is assumed to be distributed throughout the three steam generators. Recognizing that an extended plant cooldown may be required under natural circulation conditions, the CREA is conservatively analyzed using 24 hours for the cooldown time. The activity associated with the 1 gpm leak is assumed to be released to the environment via the 3 intact SGs until the primary system pressure is less than the secondary system pressure (24 hours) at which time the release of radioactivity from the secondary system as a result of steam dump through the relief valve to the atmosphere terminates. At this point, no steam and activity are released to the environment. All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation. The radioactivity in the secondary water is assumed to become a vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine and the alkali metals of 100 is assumed.

Credit is taken for control room filtration (i.e., with an assumed charcoal filtration efficiency of 95%). As described in Section 9.4.1 of the VCSNS updated FSAR, the control room ventilation system is automatically placed in the emergency mode, with filtration of incoming and recirculated air, following receipt of a SI or high radiation signal from the gaseous activity channel of RM-A1. Although very sensitive and fast acting, RM-A1 is not credited due to lack of redundancy. For the CREA with releases to the RB, the emergency mode of operation is assumed to occur as a result of automatic SI actuation. For the maximum possible break size (2.75 in diameter), automatic SI actuation on low RCS pressure would be expected within the first few minutes of the event (per FSAR Section 15.4.6.2.10). However, because the timing of the SI signal is break size dependent, a 30-minute time delay is conservatively assumed to account for break size effects. For the SG release analysis, 2 hours from event initiation is conservatively assumed for manual initiation of the emergency mode of operation. This timing assumption provides adequate time for recognition of an ongoing atmospheric relief that requires the use of protective action to ensure Control Room Habitability.

4.8.4 Radiological Transport Modeling

A simplified radiological release model and individual pathway models developed to calculate CREA doses utilizing RADTRAD is shown in Figure 7. Specific model details and the supporting RADTRAD runs are provided in Attachment 10 as Calculation DC00040-101, "Rod Ejection – AST." Based on the calculated Control Room χ/Qs provided in Table 4.1-3, the conservative values used in the MSLB AST dose analyses are taken at the following release point.

- CREA Containment Release — RB Nearest Point
- CREA SG Release — MS SSV A (Relief's B, C, D, E), Intake B

4.8.5 Results – Control Room Operator Dose

The RADTRAD computer code was used to determine the Control Room operator dose for the CREA. The resultant doses are shown as follows.

CREA CR Operator Dose	Rem TEDE
• Case 1 CREA Containment Release=	1.71
• Case 2 CREA SG Release =	2.38

4.8.6 Results – Offsite Doses

The RADTRAD computer code was used to determine the offsite dose. The calculated doses are shown as follows where the EAB dose represents the maximum 2-hour TEDE over the accident period.

CREA EAB Dose	Rem TEDE
• Case 1 CREA Containment Release=	4.31
• Case 2 CREA SG Release =	2.52

CREA LPZ Dose	Rem TEDE
• Case 1 CREA Containment Release=	0.95
• Case 2 CREA SG Release =	0.48

4.8.7 Conclusions

The CREA Control Room operator dose is below the 5 rem TEDE regulatory limit.

The CREA offsite (EAB and LPZ) doses are below the 6.3 rem TEDE regulatory limit at the EAB and the LPZ.

4.8.8 Summary of Calculation Conservatisms

- Regulatory Guide 1.183, Table 3, non-LOCA fraction of fission products inventory in the gap are conservatively doubled.

- Rb-86, Cs-134, Cs-136, and Cs-137 default core inventories from Table 1.4.3.2-2 of NUREG/CR-6604, corrected to a core thermal power of 2958 MWt, are included in this analysis. In addition, VCSNS noble gas and iodine core inventories and the corrected core inventories Table 1.4.3.2-2 of NUREG/CR-6604 were compared and the larger of the two concentrations are used in this analysis.

- The Control Room unanticipated unfiltered inleakage (UUI) value of 41 scfm, based on tracer tests, was increased to 200 scfm. An uncertainty value of 25 percent is conservatively applied to the Control Room filtered makeup flow rate and the unfiltered makeup flow rate which bypasses the damper.

Figure 7A: RADTRAD Model Developed to Calculate CREA Doses (Containment Release)

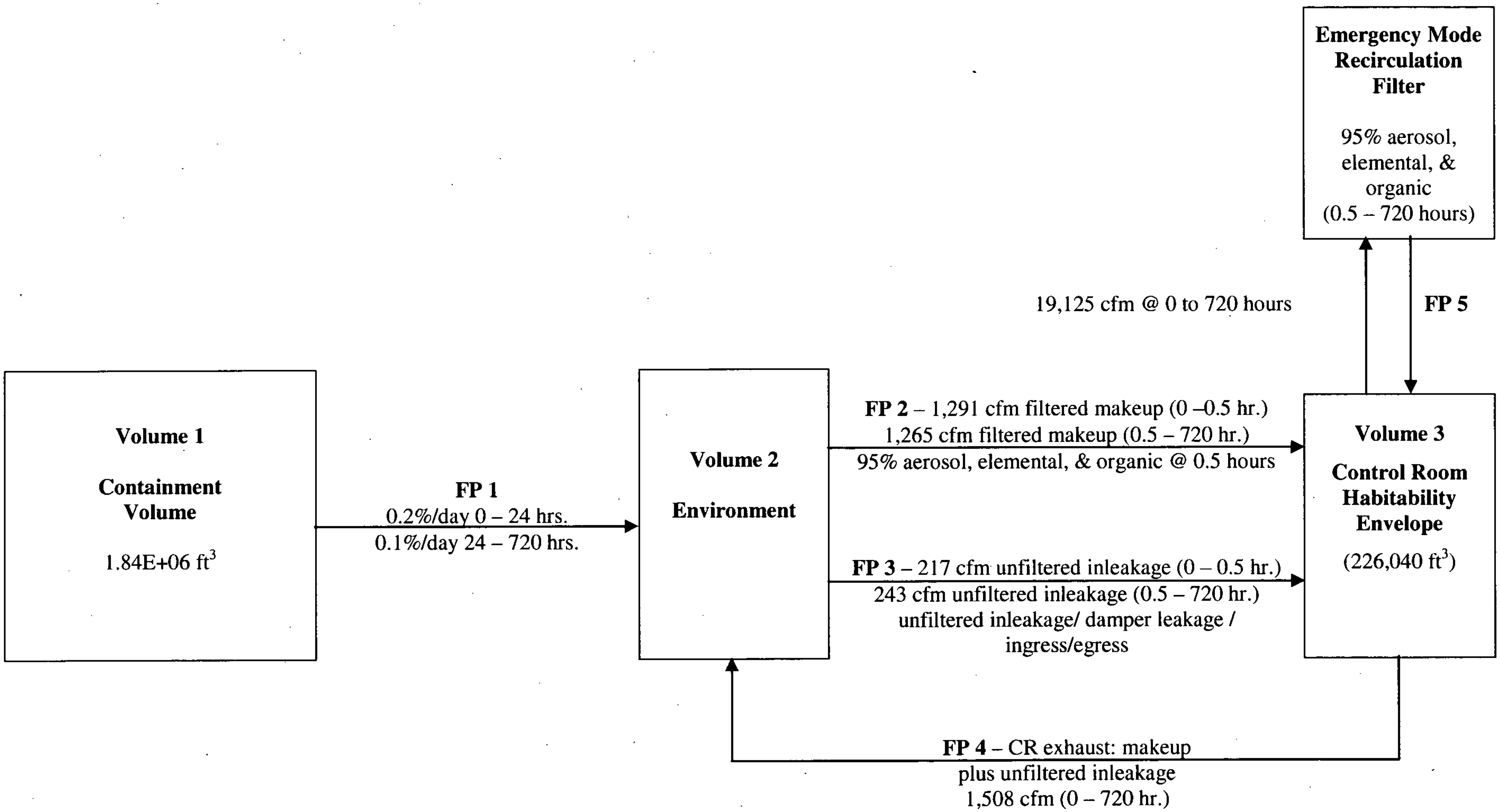
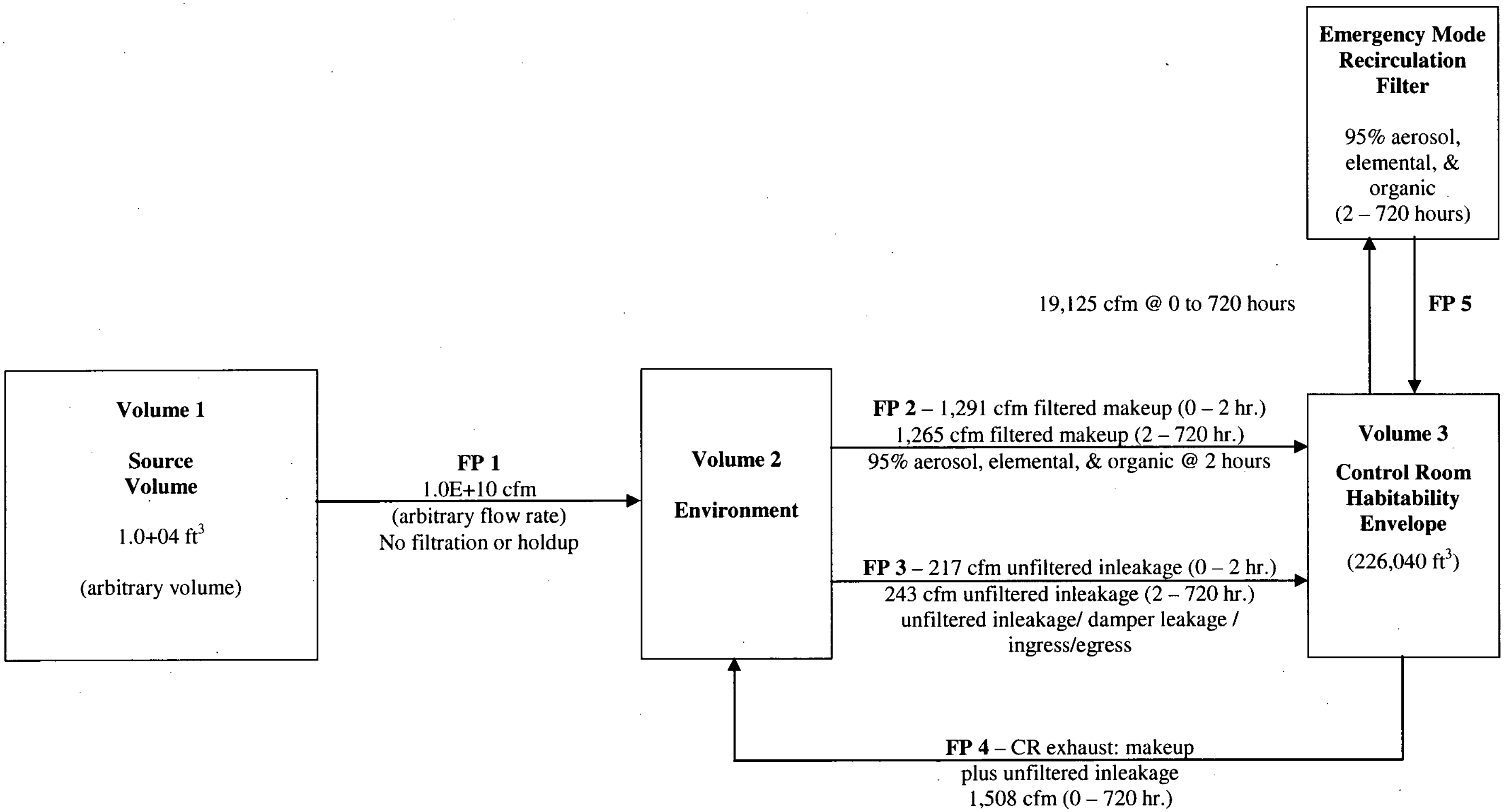


Figure 7B: RADTRAD Model Developed to Calculate CREA Doses (SG PORV Release)



4.9 Equipment Qualification and NUREG-0737

As previously noted, this request for an amendment to the licensing basis for the Virgil C. Summer Nuclear Station that a full implementation application of an Alternative Source Term (AST) methodology with the following exceptions. The exceptions are that the current TID-14844 (Reference 9) accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than Control Room Habitability Envelope (CRHE) doses, and FSAR accidents not included in Regulatory Guide 1.183.

5.0 REGULATORY SAFETY ANALYSIS (10 CFR 50.92 Evaluation)

The 10 CFR 50.92 Evaluation is included as Attachment 9 of this submittal.

6.0 ENVIRONMENTAL CONSIDERATIONS (10 CFR 50.21 Evaluation)

The 10 CFR 50.21 Evaluation is included as Attachment 9 of this submittal.

7.0 REFERENCES

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

2. NUREG-0800, Section SRP 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms", Revision 0, July 2000.

3. USNRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations For Control Room Radiological Habitability Assessments At Nuclear Power Plants", June 2003.

4. NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

5. J.V. Ramsdell and C.A. Simonen, "Atmospheric Relative Concentrations in Building Wakes", NUREG-6331, Revision 1, USNRC, May 1997 (ARCON96 computer code).

6. USNRC, "Standard Review Plan For the Review of Safety Analysis Reports for Nuclear Power Plants", Chapter 6.4, "Control Room Habitability System," NUREG-0800, USNRC, 1987.

7. NRC Letter Virgil C. Summer Nuclear Station – NRC Receipt of response to Generic Letter 2003-01, "Control Room Habitability" (TAC NO. MB9860), from Robert E. Martin to Jeffery Archie, 10/24/2006.

8. Westinghouse Electric Company, "Radiation Analysis Manual", for Virgil C. Summer (CGE/3-1), Revision 1, attached to letter CGE-98-036, dated October 14, 1998.

9. U.S. Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", dated March 23, 1962.

10. Jeffery B. Archie (SCE&G) letter, RC-05-0193 to Document Control Desk (NRC), "Response to NRC Generic Letter 2003-01 Control Room Habitability", November 18, 2005.

11. General Test Procedure, GTP-006, "General Procedure for System Leakage Assessment."

12. Abnormal Operating Procedure, AOP-112.2, "Steam Generator Tube Leak Not Requiring SI."

13. NUREG/CR-6604, RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation, December 1997, Supplement 1, June 8, 1999 & Supplement 2, October 2002.

Attachment 3

Regulatory Guide 1.183 Compliance Table

Regulatory Guide 1.183 Compliance		
Regulatory Guide 1.183 Sections 3 through 7 and Appendices A, B, E, F, G and H provide methodologies and assumptions that are acceptable to the NRC staff related to design basis radiological analyses for Alternate Source Term. Compliance with Regulatory Guide 1.183 positions are discussed below: Please note: the information provided in this table is based on the calculations provided in Attachment 10.		
RG 1.183 Section	**Regulatory Guide 1.183 Position**	**Basis of Compliance**
3.	ACCIDENT SOURCE TERM This section provides an AST that is acceptable to the NRC staff. The data in Regulatory Positions 3.2 through 3.5 are fundamental to the definition of an AST. Once approved, the AST assumptions or parameters specified in these positions become part of the facility's design basis. Deviations from this guidance must be evaluated against Regulatory Position 2. After the NRC staff has approved an implementation of an AST, subsequent changes to the AST will require NRC staff review under 10 CFR 50.67.	Conforms Accident source terms were developed based on the guidance of Section 3 of RG 1.183.

3.1	Fission Product Inventory The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. Note: the uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. Note that some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel offload. Thus, the maximum inventory at the end of life should be used. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels. For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, 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. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or Technical Specifications should be applied in determining the inventory of the damaged rods. No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms The inventory of fission products in the reactor core and available for release to the containment was based on the maximum full power operation with a core thermal power of 2958 MWt [102% (ECCS evaluation uncertainty) of 2900 MWt], The maximum inventory at the end of life was used.

3.2	Release Fractions Note: the release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel. The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.	Conforms For the LOCA event, the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases in Table 2 were utilized. For non-LOCA events, the fraction of the core inventory assumed to be in the gap by radionuclide group in Table 3 were utilized in conjunction with the maximum core radial peaking factor. The CREA was evaluated per Note 11 of RG 1.183 (the gap fractions are assumed to be 10% for iodines and noble gases). To account for possible variations in burnup and rod power, the gap fractions in Table 3 were increased by a factor of 2. This factor was applied in the FHA, RCP LRA and CREA analyses.

Table 2
PWR Core Inventory Fraction
Released Into Containment

Group	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	0.05	0.95	1.00
Halogens	0.05	0.35	0.40
Alkali Metals	0.05	0.25	0.30
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.

Table 3
Non-LOCA Fraction of Fission Product Inventory in Gap

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

Note: Table 3 release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.

<table>
<tr>
<td>3.3</td>
<td>Timing of Release Phases

Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.

Table 4
LOCA Release Phases
<table><tr><th>Phase</th><th>Onset</th><th>Duration</th></tr><tr><td>Gap Release</td><td>30 sec.</td><td>0.5 hr</td></tr><tr><td>Early In-Vessel</td><td>0.5 hr</td><td>1.3 hr</td></tr></table>
In lieu of treating the release in a linear ramp manner, the activity for each phase can be modeled as being released instantaneously at the start of that release phase, i.e., in step increases.

For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.</td>
<td>Conforms

Table 4 onset and duration of each sequential release phase for the DBA LOCA were utilized in the analysis in a linear manner.</td>
</tr>
</table>

3.4	Radionuclide Composition Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** **Group** — **Elements** Noble Gases — Xe, Kr Halogens — I, Br Alkali Metals — Cs, Rb Tellurium Group — Te, Sb, Se, Ba, Sr Noble Metals — Ru, Rh, Pd, Mo, Tc, Co Lanthanides — La, Zr, Nd, Eu, Nb, Pm, Pr Sm, Y, Cm, Am Cerium — Ce, Pu, Np	Conforms Table 5 elements in each radionuclide group were utilized in design basis analyses.
3.5	Chemical Form Of the radioiodines released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Conforms Of the radioiodines released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released was assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. With the exception of elemental and organic iodine and noble gases, fission products were assumed to be in particulate form. The same chemical form was assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs.

3.6	Fuel Damage in Non-LOCA DBAs The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases. The amount of fuel damage caused by a FHA is addressed in Appendix B of this guide.	Conforms The amount of fuel damage caused by non-LOCA design basis events was analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached.
4.0	DOSE CALCULATION METHODOLOGY The NRC staff has determined that there is an implied synergy between the ASTs and total effective dose equivalent (TEDE) criteria, and between the TID-14844 source terms and the whole body and thyroid dose criteria, and therefore, they do not expect to allow the TEDE criteria to be used with TID-14844 calculated results. The guidance of this section applies to all dose calculations performed with an AST pursuant to 10 CFR 50.67. Certain selective implementations may not require dose calculations as described in Regulatory Position 1.3 of this guide.	Conforms The DBA analyses, based on ASTs, utilized the dose calculation methodology of Section 4 of RG 1.183.
4.1	Offsite Dose Consequences The following assumptions should be used in determining the TEDE for persons located at or beyond the boundary of the exclusion area (EAB):	Conforms The dose calculation methodology of Section 4.1 of RG 1.183 was utilized to calculate the offsite dose consequences.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity. Note: The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of TEDE and the characteristics of the revised source term.	Conforms The dose calculations determine the TEDE and consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences.

4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms Conversion factors for isotopes other than the standard 60 isotopes of the RADTRAD computer program (default FGR 11 files) were taken from Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms Breathing rates provided in Section 4.1.3 of RG 1.183 were utilized to calculate the offsite dose consequences.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms Conversion factors for isotopes other than the standard 60 isotopes of the RADTRAD computer program (default FGR 12 files) were taken from Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil."
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6). Note: With regard to the EAB TEDE, the maximum two-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.	Conforms The TEDE was determined for the most limiting person at the EAB. The maximum two-hour TEDE was determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods.

4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms The TEDE was determined for the most limiting receptor at the outer boundary of the low population zone (LPZ).
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms No correction was made for depletion of the effluent plume by deposition on the ground.
4.2	Control Room Dose Consequences The following guidance should be used in determining the TEDE for persons located in the Control Room:	Conforms The DBA analyses utilized the guidance of Section 4.2 of RG 1.183 to determining the TEDE for persons located in the Control Room.
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to Control Room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the Control Room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility. • Contamination of the Control Room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the Control Room envelope. • Radiation shine from the external radioactive plume released from the facility. • Radiation shine from radioactive material in the reactor containment. • Radiation shine from radioactive material in systems and components inside or external to the Control Room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms The TEDE analysis considered all significant sources of radiation that will cause exposure to Control Room personnel.
4.2.2	The radioactive material releases and radiation levels used in the Control Room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the Control Room.	Conforms The radioactive material releases and radiation levels used in the Control Room dose analysis were determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values as appropriate.

4.2.3	The models used to transport radioactive material into and through the Control Room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to Control Room personnel. The iodine protection factor (IPF) methodology of Reference 22 may not be adequately conservative for all DBAs and Control Room arrangements since it models a steady-state Control Room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 23) and RADTRAD (Ref. 24) incorporate suitable methodologies.	Conforms The models used to transport radioactive material into and through the Control Room, and the shielding models used to determine radiation dose rates from external sources, were developed to provide suitably conservative estimates of the exposure to Control Room personnel.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the Control Room may be assumed. Such features may include Control Room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. The Control Room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, Control Room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the Control Room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	Conforms Credit for engineered safety features that mitigate airborne radioactive material within the Control Room were assumed as appropriate. Credit for engineered safety features varied for each of the analyzed DBAs.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms Credit was not taken for the use of personal protective equipment or prophylactic drugs.

4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the Control Room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5 x 10^{-4} cubic meters per second. Note: This occupancy is modeled in the χ/Q values determined in Reference 22 and should not be credited twice. The ARCON96 Code (Ref. 26) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.	Conforms Occupancy factors and breathing rate of RG 1.183, Section 4.2.6 were utilized to determine the doses to the hypothetical maximum exposed individual who is present in the Control Room. Control Room χ/Q values were determined utilizing the ARCON96 computer code. Occupancy factors were included in the RADTRAD computer code for dose evaluations.
4.2.7	Control Room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the Control Room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the Control Room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the Control Room (Ref. 22). $DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$ Equation 1	Conforms The DDE from photons was corrected for the difference between finite cloud geometry in the Control Room and the semi-infinite cloud assumption used in calculating the dose conversion factors by Equation 1 as necessary.
4.3	Other Dose Consequences The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms Exception – The current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than Control Room Habitability Envelope (CRHE) doses, and FSAR accidents not included in Regulatory Guide 1.183.

4.4	Acceptance Criteria The radiological criteria for the EAB, the outer boundary of the LPZ, and for the Control Room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The Control Room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).	Conforms The DBAs were updated for consistency with *the TEDE criterion in 10 CFR 50.67(b)(2)(iii).*

Table 6
Accident Dose Criteria

Accident or Case	**AB and LPZ Dose Criteria**	**Analysis Release Duration**
LOCA	25 rem TEDE	30 days for containment and ECCS leakage
PWR Steam Generator Tube Rupture		
Fuel Damage or Pre-incident Spike	25 rem TEDE	Affected SG: time to isolate; Unaffected SG(s): until cold shutdown is established
Coincident Iodine Spike	2.5 rem TEDE	
PWR Main Steam Line Break		
Fuel Damage or Pre-incident Spike	25 rem TEDE	Until cold shutdown is established
Coincident Iodine Spike	2.5 rem TEDE	
PWR Locked Rotor Accident	2.5 rem TEDE	Until cold shutdown is established
PWR Rod Ejection Accident	6.3 rem TEDE	30 days for containment pathway; until cold shutdown is established for secondary pathway
Fuel Handling Accident	6.3 rem TEDE	2 hours

The column labeled "Analysis Release Duration" is a summary of the assumed radioactivity release durations identified in the individual appendices to this guide. Refer to these appendices for complete descriptions of the release pathways and durations.

5.0	ANALYSIS ASSUMPTIONS AND METHODOLOGY	
5.1	General Considerations	
5.1.1	Analysis Quality The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than being modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence – the proposed deviation may not be conservative for other accident sequences.	Conforms Analyses performed per 10 CFR 50, Appendix B and the guidance consistent with RG 1.183.
5.1.2	Credit for Engineered Safeguard Features Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by Technical Specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms Credit was taken for Engineered Safeguard Features with failure assumptions to maximize the calculated doses. Assumptions regarding the occurrence and timing of a loss of offsite power were also selected with the objective of maximizing the postulated radiological consequences

5.1.3	Assignment of Numeric Input Values The numeric values that are chosen as inputs to the analyses required by *10 CFR 50.67* should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be non-conservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be non-conservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by Technical Specifications, the value used in the analysis should be that specified in the technical specifications. If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing, e.g., steam generator nondestructive testing (NDT), consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value. Note that for some parameters, the Technical Specification value may be adjusted for analysis purposes by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 25) and in Generic Letter 99-02 (Ref. 27) rather than the surveillance test criteria in the Technical Specifications. Generally, these adjustments address potential changes in the parameter between scheduled surveillance tests.	Conforms The numeric values that were chosen as inputs to the analyses required by *10 CFR 50.67* were selected with the objective of determining a conservative postulated dose. For a range of values, the value that resulted in a conservative postulated dose was used.
5.1.4	Applicability of Prior Licensing Basis The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Conforms Licensee has ensured that analysis assumptions and methods are compatible with the AST and the TEDE criteria.

5.2	Accident-Specific Assumptions The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. The DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations, updated technical analyses, or, in some cases, a previously approved licensing basis consideration. The assumptions in the appendices are deemed consistent with the AST identified in Regulatory Position 3 and internally consistent with each other. Although licensees are free to propose alternatives to these assumptions for consideration by the NRC staff, licensees should avoid use of previously approved staff positions that would adversely affect this consistency. The NRC is committed to using probabilistic risk analysis (PRA) insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and addresses accident considerations not adequately addressed by the core damage frequency (CDF) and large early release frequency (LERF) surrogate indicators of overall risk.	Conforms Licensee analyzed the DBAs that are affected by the specific proposed applications of an AST, utilizing the guidance provided in the appendices of RG 1.183.

5.3	Meteorology Assumptions Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the Control Room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28). References 22 and 28 should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 (Ref. 26) is generally acceptable to the NRC staff for use in determining Control Room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in χ/Q analysis methodology should be reviewed by the NRC staff.	Conforms Atmospheric dispersion values (χ/Q) for the EAB and the LPZ, that were approved by the staff during initial facility licensing or in subsequent licensing proceedings are used in performing the radiological analyses identified by this guide. New χ/Qs for the Control Room were developed based on more recent meteorological data, utilizing the guidance of RG 1.194.
6.0	<u>ASSUMPTIONS FOR EVALUATING THE RADIATION DOSES FOR EQUIPMENT QUALIFICATION</u> The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 11), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective. The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs. TID14844) on EQ doses pending the outcome of the evaluation of the generic issue.	N/A An AST assessment was not performed for equipment qualification. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification and radiation zone maps/shielding calculations.

D.	**IMPLEMENTATION** The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the methods described in this guide will be used in the evaluation of submittals related to the use of ASTs in radiological consequence analyses at operating power reactors.	Conforms The AST analysis utilized the guidance of RG 1.183 in the AST evaluations and did not use alternative method for complying with the specified portions of the NRC's regulations.

Appendix A	**ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LWR LOSS-OF-COOLANT ACCIDENT** The assumptions in this appendix are acceptable to the NRC staff for evaluating the radiological consequences of loss-of-coolant accidents (LOCAs) at light water reactors (LWRs). These assumptions supplement the guidance provided in the main body of this guide. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system are included. The LOCA, as with all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.	Conforms An analysis was performed utilizing the guidance of Appendix A and appropriate sections in the main body of RG 1.183 to evaluate a LOCA.
	SOURCE TERM ASSUMPTIONS	
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms Assumptions regarding core inventory and the release of radionuclides from the fuel were per Regulatory Position 3 of RG 1.183.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms The sump pH is controlled at values of 7 or greater.
	ASSUMPTIONS ON TRANSPORT IN PRIMARY CONTAINMENT	
3	Acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the primary containment in PWRs are as follows:	Conforms The analysis utilized the assumptions related to the transport, reduction, and release of radioactive material in and from the primary containment in PWRs per RG 1.183.

3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The release into the primary containment should be assumed to terminate at the end of the early in-vessel phase.	Conforms The radioactivity released from the fuel was assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it was released. This distribution was not adjusted because there is adequate ventilation exchange.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Conforms The 10^{th} percentile Power's Aerosol Decontamination Model was conservatively used in the analysis for natural deposition.

3.3	*Reduction in airborne radioactivity in the containment by containment spray systems that have been* designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"1 (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3). The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology). This document describes statistical formulations with differing levels of uncertainty. The removal rate constants selected for use in design basis calculations should be those that will maximize the dose consequences.	*Conforms* The VCSNS containment building atmosphere is considered a single, well-mixed volume since the spray covers at least 90% of the volume and adequate mixing of unsprayed compartments is provided in the design. The elemental iodine spray removal coefficient is 20 hr^{-1} and the particulate iodine spray removal coefficient is 5.68 hr^{-1}. The removal rate for the particulate iodines is reduced by a factor of 10 (0.568 hr^{-1}) when a DF of 50 is reached. There is no specified maximum DF for aerosols removed by sprays. No credit is taken for the removal of organic iodine by the spray system.

3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Conforms. The Reactor Building Cooling System recirculation flow rate is rated at 60, 270 ± 10% acfm. The minimum allowable system flow is 54,243 acfm. 54, 200 cfm is used in the RADTRAD analyses. The recirculation HEPA efficiency is assumed to be 90 percent for the removal of iodine particulates only. This is conservative since Regulatory Guide 1.52 permits a 99 percent removal efficiency for particulates in accident dose evaluations.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	NA
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	N/A No credit is taken in this analysis for reduction of airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above.
3.7	The primary containment should be assumed to leak at the peak pressure Technical Specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the Technical Specification leak rate. Leakage from sub-atmospheric containments is assumed to terminate when the containment is brought to and maintained at a sub-atmospheric condition as defined by Technical Specifications.	Conforms The primary containment peak pressure leak rate is defined as 0.2% by weight of containment air. This leak rate is reduced to 0.1% after the first 24 hours of the accident.

3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the Technical Specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	N/A The primary containment is not routinely purged during power operations.
	ASSUMPTIONS ON DUAL CONTAINMENTS	
4	For facilities with dual containment systems, the acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the secondary containment or enclosure buildings are as follows:	NA
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in Technical Specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	NA
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in Technical Specifications.	NA
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	NA
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	NA

4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the Technical Specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	NA
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	NA
	ASSUMPTIONS ON ESF SYSTEM LEAKAGE	
5	ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (Ref. A-7). The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment:	Conforms This analysis utilized the ESF leakage assumptions in Section 5 of RG 1.183.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Table 1 of this guide) should be assumed to instantaneously and homogeneously mix in the suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are non-conservative with regard to the buildup of sump activity.	Conforms With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Table 1 of RG 1.183) were assumed to instantaneously and homogeneously mix in the primary containment sump water for this analysis.

5.2	*The leakage should be taken as two times the sum of the simultaneous leakage from all components* in the ESF recirculation systems above which the Technical Specifications, or licensee commitments to Item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	*Conforms* The VCSNS technical specifications do not provide a specific limit for operational leakage that is allowed within the recirculation loop. Administrative limits, however, ensure that operational leakage is adequately controlled. A post LOCA recirculation leakage of 12,000 cc/hr (4.238E-01 cfh or 7.063E-03 cfm) is used as input to the dose calculations. This is twice the operational limit that is used in plant procedures for system leakage assessments (GTP-006). In the event total recirculation loop leakage exceeds 6,000 cc/hr, a Condition Evaluation Report is generated to facilitate a licensing basis impact assessment and an operability determination. Leakage through the RWST and NaOH Tank is neglected in this calculation based on plant procedures (EOPs) that require closure of the 20" RWST outlet valve (6700) and closure of the 3" NAOH outlet valve (3012) following the transition to CL recirculation. This results in 3 valve isolation and a minimum of 2 valve isolation in the long term with a single failure. As outlined in Attachment 8, EOP changes will be made prior to AST implementation to support this analysis assumption.
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms With the exception of iodine, all radioactive materials in the recirculating liquid were assumed to be retained in the liquid phase.

5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$ *Where:* h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.	N/A The temperature of the leakage does not exceed 212°F.
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms The temperature of the leakage does not exceed 212°F and a flash fraction of 10% was assumed for the iodine.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms The radioiodine that is postulated to be available for release to the environment was assumed to be 97% elemental and 3% organic. No reductions due to dilution, holdup, or by ESF ventilation filtration systems were assumed.
	ASSUMPTIONS ON MAIN STEAM ISOLATION VALVE LEAKAGE IN BWRS	
6	For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of MSIV leakage.	NA
6.1	For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.	NA

6.2	*All the MSIVs should be assumed to leak at the maximum leak rate above which the Technical* Specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.	NA
6.3	Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.	NA
6.4	In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the Turbine Building should not be assumed.	NA
6.5	A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.	NA
	ASSUMPTION ON CONTAINMENT PURGING	
7	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	N/A

Appendix B	**ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT** This appendix provides assumptions acceptable to the staff for evaluating the radiological consequences of a fuel handling accident at light water reactors. These assumptions supplement the guidance provided in the main body of this guide.	Conforms An analysis was performed utilizing the guidance of Appendix B and appropriate sections in the main body of RG 1.183 to evaluate a fuel and an equipment accident.
1	SOURCE TERM Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The following assumptions also apply.	Conforms Assumptions regarding core inventory and the release of radionuclides from the fuel are taken from Regulatory Position 3 of RG 1.183.
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms For the postulated FHA inside Containment and inside the FHB, a total of 314 pins are assumed to be damaged as a result of this event. All 264 pins in the dropped spent fuel assembly and 50 pins in the impacted assembly are assumed to be rupture. These assumptions are consistent with the CLB analysis.
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms The fission product release from the breached fuel is based on Regulatory Position 3.2 of RG 1.183 (see response to RG 1.183, Section 3.2 of Attachment 3) and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that were considered include xenons, kryptons, halogens, cesiums, and rubidiums. Please note, RG 1.183, Appendix B, Section 3, the pool DF for particulates which includes cesiums, and rubidiums is infinite. Therefore, they are neglected from further consideration in the analysis.

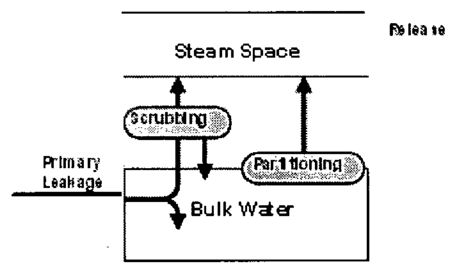
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms All iodine released to the spent fuel pool dissociates and re-evolves as elemental iodine *instantaneously*.
2	WATER DEPTH If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).	Conforms The minimum water depth over the reactor core when handling fuel and over the spent fuel in the FHB is 23 feet
3	NOBLE GASES The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conform Noble gas DF = 1 Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).
4	FUEL HANDLING ACCIDENTS WITHIN THE FUEL BUILDING For fuel handling accidents postulated to occur within the fuel building, the following assumptions are acceptable to the NRC staff.	Conforms The fuel handling accident was evaluated within the Fuel Handling Building outside containment.
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms The radioactive material that escapes from the fuel pool to the Fuel Handling Building is assumed to be released to the environment over a 2-hour time period.

4.2	*A reduction in the amount of radioactive material released from the fuel pool by engineered safety* feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses Note: These analyses should consider the time for the radioactivity concentration to reach levels corresponding to the monitor setpoint, instrument line sampling time, detector response time, diversion damper alignment time, and filter system actuation, as applicable.	NA Consistent with the proposed change to VCSNS Technical Specification 3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM, no credit is taken for filtration in the FHA analysis
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Conforms No credit is taken for mixing or dilution in the fuel building.
5	FUEL HANDLING ACCIDENTS WITHIN CONTAINMENT For fuel handling accidents postulated to occur within the containment, the following assumptions are acceptable to the NRC staff.	Conforms The fuel handling accident was evaluated within the Containment Building.
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed. Note: Containment isolation does not imply containment integrity as defined by Technical Specifications for non-shutdown modes. The term isolation is used here collectively to encompass both containment integrity and containment closure, typically in place during shutdown periods. To be credited in the analysis, the appropriate form of isolation should be addressed in Technical Specifications.	NA Consistent with the proposed change to VCSNS Technical Specification 3/4.9.4 REACTOR BUILDING PENETRATIONS, no credit is taken for containment isolation in the FHA analysis.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	NA No credit is taken for containment isolation in the FHA analysis.

5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period. Note: The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.	Conforms The containment is assumed open during the postulated FHA and the activity is released over a 2-hour period.
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	NA No credit is taken for reduction in the amount of radioactive material released from the containment by ESF filter systems.
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	NA No credit is taken for dilution or mixing in containment prior to release of activity to the environment.

Appendix E	**ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR MAIN STEAM LINE BREAK ACCIDENT** This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a main steam line break accident at PWR light water reactors. These assumptions supplement the guidance provided in the main body of this guide.	
1	SOURCE TERMS Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.	Conforms No fuel damage is postulated to occur during the MSLB.
2	If no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.	Conforms There is no fuel damage for the VCSNS MSLB. Two cases of iodine spiking are used in the analysis.
2.1	A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Conforms The VCSNS analysis is based on the maximum value of 60 μCi/gm DE I-131 permitted by the technical specifications.
2.2	The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8- hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms The VCSNS analysis uses a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications. The assumed iodine spike duration is 8 hours.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms Iodine chemical form is in accordance with this guidance, 97% elemental and 3% organic.
5	TRANSPORT Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows.	Conforms The noble gases are released to the environment without reduction or mitigation.
5.1	For facilities that have not implemented alternative repair criteria (see Ref. E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms Per Technical Specification Bases 3/4.4.5, a total primary to secondary leakage for all three steam generators of 1 gpm is used in the evaluation of design basis accidents. Prior to the event this leakage is assumed to be distributed throughout the three steam generators. Recognizing that an extended plant cooldown may be required under natural circulation conditions, the MSLB is conservatively analyzed using 24 hours for the cooldown time. The activity associated with the 1 gpm leak is assumed to be released to the environment via the faulted steam generator at a rate of 0.35 gpm for the 24 hour duration of the event with no credit taken for any reduction or mitigation, i.e. a partition factor of 1.0. This is conservative in that the actual maximum value allowed by Technical Specification 3.4.6.2.c for any one SG is 150 gpd (~ 0.104 gpm). The remaining 0.65 gpm is released to the environment via the two intact steam generators for the 24 hour duration of the event crediting a partition factor of 100.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms The density used is 1.0 gm/cc (62.4 lbm/ft^3).

5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms See item 5.1.
5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms The noble gases are released to the environment without reduction or mitigation.
5.5	The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below: **Figure E-1** **Transport Model**  	Conforms See item 5.1.
5.5.1	A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. • During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. • With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.	Conforms See item 5.1.
5.5.2	The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Ref. E-2), during periods of total submergence of the tubes.	Conforms See item 5.1.
5.5.3	The leakage that does not immediately flash is assumed to mix with the bulk water.	Conforms See item 5.1.

5.5.4	The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for elemental iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Conforms See item 5.1.
5.6	Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	NA See item 5.1.

Appendix F	**ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR STEAM GENERATOR TUBE RUPTURE ACCIDENT** This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a steam generator tube rupture accident at PWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.	
1	SOURCE TERMS Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms. No fuel damage is postulated to occur during the SGTR.
2	If no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Conforms There is no fuel damage for the VCSNS SGTR. Two cases of iodine spiking are used in the analysis.
2.1	A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Conforms The VCSNS analysis is based on the maximum value of 60 μCi/gm DE I-131 permitted by the technical specifications.
2.2	The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8- hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms The VCSNS analysis uses a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications. The assumed iodine spike duration is 8 hours.

3	The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
4	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms Iodine chemical form is in accordance with this guidance, 97% elemental and 3% organic.
5	TRANSPORT Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:	Conforms The VCSNS analysis is in accordance with this guidance.
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms The SGTR analysis accounts for a bounding primary-to-secondary leakage rate equal to 1 gpm and the leakage rate associated with a double-ended rupture of a single tube. Leakage through the ruptured tube is the dominate contributor to dose releases. Since contaminated fluid in the ruptured steam generator is only briefly released to the atmosphere as steam via the main steam safety valves, the entire 1 gpm primary-to-secondary leakage is conservatively assumed to occur in the intact steam generators where it can be released during the subsequent cooldown of the plant. Per Technical Specification 3.4.6.2.c, the maximum value in any one steam generator is limited to 150 gpd (~ 0.104 gpm). This release is therefore conservative in that the actual maximum value allowed by TS for the two intact SGs would be ~0.208 gpm.

5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density *should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).*	Conforms The density used is 1.0 gm/cc (62.4 lbm/ft^3).
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (2 12°F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms The activity associated with the 1 gpm primary-to-secondary leak is conservatively assumed to be released to the environment via the intact steam generator for the 24 hour duration of the event.
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms A coincident loss of offsite power is assumed.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms The noble gases are released to the environment without reduction or mitigation.
5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates is considered as appropriate in the SGTR.

Appendix G	**ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR LOCKED ROTOR ACCIDENT** This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a locked rotor accident at PWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.	
1	SOURCE TERMS Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms See the responses to RG 1.183, Sections 3.1 and 3.2 of Attachment 3.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.	Conforms Analysis uses 15% fuel damage in the core.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms Iodine chemical form is in accordance with this guidance, 97% elemental and 3% organic.
5	RELEASE TRANSPORT Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows.	
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.	Conforms Per the VCSNS Technical Specification Bases, for use in the accident analysis the total leakage for all three steam generators is 1 gpm.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms The density used is 1.0 gm/cc (62.4 lbm/ft^3).

5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (2 12°F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms The activity is released for a duration of 24 hours.
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms A coincident loss of offsite power is assumed.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms The noble gases are released to the environment without reduction or mitigation.
5.6	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates is considered as appropriate in the RCP LRA analysis.

Appendix H	**ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR ROD EJECTION ACCIDENT** This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a rod ejection accident at PWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.	
1	SOURCE TERM Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.	Conforms See the responses to RG 1.183 Sections 3.1 and 3.2 of Attachment 3.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.	Conforms Analysis uses 10% fuel damage and 0.25% fuel melt in the core.
3	Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.	Conforms Two release cases are considered. 100% of the activity released from the fuel is released instantaneously and homogeneously through the containment atmosphere. 100% of the activity released from the fuel is assumed to be completely dissolved in the primary coolant and available for release to the secondary system.
4	The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms The chemical form of radioiodine released to the containment atmosphere is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide.

5	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.
6	TRANSPORT FROM CONTAINMENT Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows.	
6.1	A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.	Conforms Credit is taken for natural deposition removal of aerosols using the RADTRAD 10% Power's model in this analysis. No credit is taken for containment sprays or the containment reactor building cooling unit recirculating filter system.
6.2	The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.	Conforms The primary containment peak pressure leak rate is defined as 0.2% by weight of containment air. This leak rate is reduced to 0.1% after the first 24 hours of the accident.
7	TRANSPORT FROM SECONDARY SYSTEM Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the secondary system are as follows.	Conforms
7.1	A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms Per the VCSNS Technical Specification Bases, for use in accident analyses the total leakage for all three steam generators is 1 gpm. The activity is released for a duration of 24 hours.
7.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms The density used is 1.0 gm/cc (62.4 lbm/ft^3).

7.3	All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Conforms The noble gases are released to the environment without reduction or mitigation.
7.4	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates is considered as appropriate in the CREA analysis.

Attachment 4

Regulatory Guide 1.194 Compliance Table

Regulatory Guide 1.194 Compliance

Regulatory Guide 1.194 Sections 3 through 7 and Table A-2 provide methodologies and assumptions that are acceptable to the NRC staff related to atmospheric relative concentrations for Control Room radiological habitability assessments at nuclear power plants. Compliance with Regulatory Guide 1.194 positions are discussed below:

Please Note: The information provided in this table is based on the calculations provided in Attachment 10.

RG 1.194 Section	Regulatory Guide 1.194 Position	Basis of Compliance
3.	CALCULATION OF χ/Q USING ARCON96 This section addresses the use of the ARCON96 code for calculating χ/Q values for design basis Control Room radiological habitability assessments. The ARCON96 code should be obtained and maintained under an appropriate software quality assurance program that complies with the applicable criteria of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 and applicable industry consensus standards to which the licensee has committed.	Conforms The ARCON96 code is maintained under a software quality assurance program that complies with Appendix B to 10 CFR Part 50 and applicable industry consensus standards.
3.1	Meteorological Data Input The meteorological data needed for χ/Q calculations include wind speed, wind direction, and a measure of atmospheric stability. These data should be obtained from an onsite meteorological measurement program based on the guidance of Safety Guide 23, "Onsite Meteorological Programs" (Ref. 12), that includes quality assurance provisions consistent with Appendix B to 10 CFR Part 50. The meteorological data set used in these assessments should represent hourly averages as defined in Safety Guide 23. Data should be representative of the overall site conditions and be free from local effects such as building and cooling tower wakes, brush and vegetation, or terrain. Collected data should be reviewed to identify instrumentation problems and missing or anomalous observations (see Ref. 13). The size of the data set used in the χ/Q assessments should be sufficiently	Conforms The meteorological data includes wind speed, wind direction, and a measure of atmospheric stability. These data were obtained from an onsite meteorological measurement program based on the guidance of Safety Guide 23, that includes quality assurance provisions consistent with Appendix B to 10 CFR Part 50.

	large such that it is representative of long-term meteorological trends at the site. The NRC *staff considers 5 years of hourly observations to be representative of long-term trends at* most sites. With sufficient justification of its representativeness, however, the minimum meteorological data set is one complete year (including all four seasons) of hourly observations. Wind direction should be expressed as the direction from which the wind is blowing (i.e., the upwind direction from the center of the site) referenced from true north. Atmospheric stability should be determined by the vertical temperature difference (ΔT) measured over the difference in height appropriate for the projected release height (including plume rise as applicable). A table of ΔT values in units of degrees Centigrade per 100 meters (°C/100m) versus stability class is given in Safety Guide 23 (Ref. 12). If other well-documented methodologies are used to estimate atmospheric stability (with appropriate justification), the models described in this guide may require modification. A well-documented methodology is one that is substantiated by diffusion data for conditions similar to those at the nuclear power plant site involved. Appendix A provides information on the structure and content of the meteorological data set and input parameters used by the ARCON96 code.	The meteorological data consists of five years of hourly data, covering the years from 2002 to 2006. Each record of the hourly data contains a location identifier, Julian day (1-366), hour (0 to 23), low-level direction, low-level speed, stability class (1=A to 7=G), upper level direction, and upper level speed. Wind speeds are entered in tenths of a reporting unit with no decimal. Wind speed is entered in units of meters/second. Wind directions are from 1 to 360 in degrees. Atmospheric stability was determined by the vertical temperature difference (ΔT) measured over the difference in height appropriate for the projected release height per Safety Guide 23.
3.2	Determination of Release Point (Source) Characteristics A 95th-percentile χ/Q value should be determined for each identified source-receptor combination. However, it may be possible to identify bounding combinations in order to reduce the needed calculational effort. In determining the bounding combinations, it will be necessary to consider the distance, direction, release mode, and height of the various release points to the environment in relation to the various Control Room intakes. Additional parameters, such as those used in establishing plume rise, may need to be considered in determining the bounding combination. For cases involving two or more release pathways associated with a single release source, a calculated composite value of χ/Q may be considered on a case-by-case basis if the licensee can demonstrate an acceptable modeling approach and justify the conservatism of any assumed weighting factors.	Conforms A total of 10 potential release points were evaluated, based on distance, direction, release mode, and height of the various release points to the environment in relation to the Control Room intake. There were no cases crediting two or more release pathways associated with a single release source. The most limiting release path was always used.

	Changes in associated parameters that could occur as a result of differences between normal operation and accident conditions, differences between accidents, differences that occur over the duration of the accident, single failure considerations, and considerations of loss of offsite power, consistent with accident sequences and descriptions, must all be considered in the characterization of the release points. The ARCON96 code provides options that allow an analyst to model ground-level, elevated stack, and vent-point source releases. In addition, the analyst can model diffuse area sources as a sub mode of the ground-level release type. These modes and limitations on their use are discussed in the positions that follow.	
3.2.1	Ground-Level Releases The ground-level release mode is appropriate for the majority of Control Room χ/Q assessments. If the release type is ground level, ARCON96 ignores all user inputs related to release velocity and radius. Release height is used to establish the plume slant path.	Conforms All release points are assumed to be ground level releases.
3.2.2	Elevated (Stack) Releases The stack release mode is appropriate for releases from a freestanding, vertical, uncapped stack that is outside the directionally dependent zone of influence of adjacent structures. Such a stack should be more than 2-1/2 times the height of the adjacent structures or be located: • more than 5L downwind of the trailing edge of upwind buildings, and • more than 2L upwind of the leading edge of downwind buildings, and • more than 0.5L crosswind of the closest edge of crosswind buildings Where L is the lesser of the height or width of the building creating the downwind, upwind, or crosswind wake. Since L will be dependent on wind direction for most building clusters, it will generally be necessary to assess the zone of influence for all directions within the 90° wind direction sector centered on the line of sight between the stack and the Control Room intake. If multiple intakes are involved such that upwind, downwind, and crosswind	N/A See Section 3.2.1.

	orientations are confounded, 5L could be used for each orientation. Plume rise from buoyancy or mechanical jet effects are not calculated by ARCON96. The analyst may determine plume rise and add the amount of rise to the physical height of the stack to obtain an effective plume height as described in Regulatory Position 6 of this guide (Note: The plume rise may not be added to the physical height of the stack for the purpose of meeting the 2-1/2 times height criterion). Although ARCON96 does not determine plume rise, the input values of stack flow, radius, and vertical velocity are used by ARCON96 to assess downwash and to estimate a limiting χ/Q value. If the Control Room intake is located close to the base of a tall stack, the elevated release model in ARCON96 generates negligibly low χ/Q values. Although perhaps numerically correct, these model results may not be sufficiently conservative for a design basis assessment since the model does not adequately address meteorological conditions that could result in higher χ/Q values. Although the staff has previously suggested that licensees model fumigation as a mechanism to address this situation, the fumigation model did not appear to adequately estimate the effluent concentrations at the bases of industrial stacks. Concentrations greater than those predicted by ARCON96 could result from diurnal wind direction changes, meander, or stagnation. Therefore, the following procedure should be used to assess whether a particular stack-intake configuration is subject to this concern and to determine the appropriate χ/Q values. In addition to running ARCON96 to determine the elevated stack χ/Q values for the Control Room assessment, the analyst should calculate the maximum elevated stack χ/Q value (non-fumigation) using the methodology of Regulatory Guide 1.145 (Ref. 9) to determine the maximum χ/Q value at ground level for the 0-2 hour interval and for the 24-96 and 96-720 hour intervals. The NRC-sponsored code, PAVAN (Ref. 14), is acceptable to the staff for this assessment. For this assessment, the input parameters should be adjusted such that the effective release height is measured from the elevation of the Control Room outside air intake rather than plant grade. The same release point characterization and meteorological data sets used in ARCON96 should be used to determine the χ/Q values for several distances in each wind direction sector with the objective of identifying the maximum χ/Q value. Figure A.4 of Reference 15 may be useful in this regard. The maximum χ/Q value obtained for the 0-2 hour interval should be compared to the corresponding χ/Q value generated by ARCON96 and the higher value used in habitability assessments. The χ/Q values generated by ARCON96 for the 2-8 and the 8-24 hour intervals may be used without adjustment.	

	For the 24-96 hour and 96-720 hour intervals, the following expressions may be used to determine the effective χ/Q. This deterministic approach assumes that the stack plume reverses direction for 1 hour of each day for the duration of the event. The plume is assumed to fold over itself such that the ground level concentration is at its maximum value at the Control Room intake. $$\left(\frac{X}{Q}\right)_{24\rightarrow 96\text{ HRS}} = \frac{1*\left(\frac{X}{Q}\right)^{\text{PAVAN}}_{24\rightarrow 96\text{ HRS}} + 23*\left(\frac{X}{Q}\right)^{\text{ARCON96}}_{24\rightarrow 96\text{ HRS}}}{24} \quad (1)$$ $$\left(\frac{X}{Q}\right)_{96\rightarrow 720\ hrs} = \frac{1*\left(\frac{X}{Q}\right)^{\text{PAVAN}}_{96\rightarrow 720\ hrs} + 23*\left(\frac{X}{Q}\right)^{\text{ARCON96}}_{96\rightarrow 720\ hrs}}{24} \quad (2)$$	
3.2.3	Vent Releases The ARCON96 calculation of vent releases includes an algorithm to model mixed-mode releases as described in Regulatory Guide 1.111 (Ref. 10), which addresses χ/Q values used in the assessment of routine effluent releases. The development of this algorithm was based in part on limited field experiments. Given the limited experiment set, the results obtained with this algorithm may not be sufficiently conservative for accident evaluations. For this reason, the vent release mode should not be used in design basis assessments. This position is consistent with the guidance of Regulatory Guide 1.145 (Ref. 9) for offsite χ/Q values. These releases should be treated as a ground level release (Section 3.2.1) or as an elevated release (Section 3.2.2).	N/A See Section 3.2.1.

3.2.4	Diffuse Area Sources The diffusion models in ARCON96 are based on point-source formulations. However, some release sources may be better characterized as area sources. Examples of possible area sources are postulated releases from the surface of a reactor or a secondary containment building. Typical assessments for loss-of-coolant accidents (LOCAs) have conservatively assumed that the containment structure could leak anywhere on the exposed surface. As such, these assessments typically used the shortest distance between the building surface and the Control Room intake and have treated the building as a point source. This approach may be unnecessarily conservative. A more reasonable approach, while still maintaining adequate conservatism, would be to model the building surface as a vertical planar area source. This approach is not intended to address dispersion resulting from building-induced turbulence. Treatment of a release as a diffuse source will be acceptable for design basis calculations if the guidance herein is followed. The staff may consider deviations from this guidance on a case-by-case basis.	N/A The diffusion models are based on point-source formulations.
3.2.4.1	Diffuse source modeling should be used only for those situations in which the activity being released is homogeneously distributed throughout the building and when the assumed release rate from the building surface would be reasonably constant over the surface of the building. For example, steam releases within a Turbine Building with roof ventilators or louvered walls would generally not be suitable for modeling as a diffuse source. (See Regulatory Positions 3.2.4.7 and 3.2.4.8.).	N/A The diffusion models are based on point-source formulations.
3.2.4.2	Since leakage is more likely to occur at a penetration, analysts must consider the potential impact of building penetrations exposed to the environment within this modeled area. If the penetration release would be more limiting, the diffuse area source model should not be used. Releases from personnel air locks and equipment hatches exposed to the environment, or containment purge releases prior to containment isolation, may need to be treated differently. It may be necessary to consider several cases to ensure that the χ/Q value for the most limiting location is identified.	N/A The diffusion models are based on point-source formulations.

	Note: Penetrations that are enclosed within safety-related structures need not be considered in this evaluation if the release would be captured and released via a plant ventilation system, as ventilation system releases should have already been addressed as a separate release point.	
3.2.4.3	The total release rate (e.g., $Ci{\cdot}s^{-1}$) from the building atmosphere is to be used in conjunction with the diffuse area source χ/Q in assessments. This release rate is assumed to be equally distributed over the entire diffuse source area from which the radioactivity release can enter the environment. For freestanding containments, this would be the entire periphery above grade or above a building that surrounds the lower elevations of the containment. When a licensee can justify assuming collection of a portion of the release from the containment within the surrounding building, the total release from the containment may be apportioned between the exposed and enclosed building surfaces. Similarly, if the building atmosphere release is modeled through more than one simultaneous pathway (e.g., drywell leakage and main steam safety valve leakage in a BWR), only that portion of the total release released through the building surface should be used with the diffuse area χ/Q. The release rate should not be averaged or otherwise apportioned over the surface area of the building. For example, reducing the release rate by 50 percent because only 50 percent of the surface faces the Control Room intake would be inappropriate.	N/A The diffusion models are based on point-source formulations.
3.2.4.4	ARCON96 uses two initial diffusion coefficients entered by the user to represent the area source. There are insufficient field measurements to mechanistically model these initial diffusion coefficients. The following deterministic equations should be used in the absence of site-specific empirical data. Note: See Regulatory Position 7 regarding the use of site-specific empirical measurements. $\sigma_{Y_0} = \frac{Width_{area\ source}}{6}$ (3) $\sigma_{Z_0} = \frac{Height_{area\ source}}{6}$ (4)	N/A The diffusion models are based on point-source formulations.

3.2.4.5	The height and width of the area source (e.g., the building surface) are taken as the maximum vertical and horizontal dimensions of the above-grade building cross-sectional area perpendicular to the line of sight from the building center to the Control Room intake. These dimensions are projected onto a vertical plane perpendicular to the line of sight and located at the closest point on the building surface to the Control Room intake. The release height is set at the vertical center of the projected plane. The source-to-receptor distance (slant path) is measured from this point to the Control Room intake.	N/A The diffusion models are based on point-source formulations.
3.2.4.6	Intentional releases from a secondary containment (e.g., standby gas treatment systems (SGTS) at BWR reactors) or annulus ventilation systems in dual containment structures should be treated as a ground-level release or an elevated stack release, as appropriate. The diffuse area source model may be appropriate for time intervals for which the secondary containment or annulus ventilation system is not capable of maintaining the requisite negative pressure differential specified in Technical Specifications or in the FSAR. Secondary containment bypass leakage (i.e., leakage from the primary containment that bypasses the secondary containment and is not collected by the SGTS) should be treated as a ground-level release or an elevated stack release, as appropriate.	N/A The diffusion models are based on point-source formulations.
3.2.4.7	A second possible application of the diffuse area source model is determining a χ/Q value for multiple (i.e., 3 or more) roof vents. This treatment would be appropriate for configurations in which (1) the vents are in a close arrangement, (2) no individual vent is significantly closer to the Control Room intake than the center of the area source, (3) the release rate from each vent is approximately the same, and (4) no credit is taken for plume rise. The distance to the receptor is measured from the closest point on the perimeter of the assumed area source. For assumed areas that are not circular, the area width is measured perpendicular to the line of sight from the center of the assumed source to the Control Room intake. The initial diffusion coefficient σ_{Y_o} is found by Equation 3; σ_{Z_o} is assumed to be 0.0. Note: The degree of significance will depend on the radius or width of the assumed area and the proximity of the vent cluster to the Control Room intake. As the radius decreases or the distance from the cluster to the Control Room intake increases, the less significance the position of any one vent has.	N/A The diffusion models are based on point-source formulations.

3.2.4.8	A third possible application of the diffuse area source model is determining a χ/Q value for large louvered panels or large openings (e.g., railway doors on BWR Mark I plants) on vertical walls. This treatment would be appropriate for a louvered panel or opening when (1) the release rate from the building interior is essentially equally dispersed over the entire surface of the panel or opening and (2) assumptions of mixing, dilution, and transport within the building necessary to meet condition 1 are supported by the interior building arrangement. The staff has traditionally not allowed credit for mixing and holdup in Turbine Buildings because of the buoyant nature of steam releases and the typical presence of high volume roof exhaust ventilators. The distance to the receptor and the release height is measured from the center of the louvered panel or opening. Initial diffusion coefficients are found using Equations 3 and 4 assuming the width and height is that of the panel or opening rather than that of the building. If the area source and the intake are on the same building surface such that wind flows along the building surface would transport the release to the intake, the initial dispersion coefficient will need to be adjusted. If the included angle between the source-receptor line of sight and the vertical axis of the assumed source is less than 45 degrees, σ_{Y_o} should be set to 0.0. If the included angle between the source-receptor line of sight and the horizontal axis of the assumed source is less than 45 degrees, σ_{Z_o} should be set to 0.0.	N/A The diffusion models are based on point-source formulations.
3.3	Determination of Control Room Intakes (Receptors) This section of the guide provides guidance to the meteorological analyst in applying models for determining χ/Q values that are appropriate for the as-built configuration of Control Room intakes. Radioactive materials released during an accident can enter the Control Room envelope via several potential pathways. These pathways may be intentional (e.g., ventilation system outside air intakes) and unintentional infiltration paths (e.g., doorways, envelope penetrations, leakage in ventilation system components). The applicable pathways will vary from site to site depending on the arrangement of the Control Room envelope in relation to other site buildings, the pressure differentials between these buildings and the Control Room, the configuration of Control Room ventilation systems, and the classification of the Control Room dose control (e.g., zone isolation with filtered pressurization, zone isolation with no pressurization). It may be necessary to determine χ/Q values for each potential pathway. However, the selection of one or more bounding intakes for the χ/Q evaluation may be sufficient to establish compliance with regulatory guidelines.	Conforms The receptors considered in the calculation are the two Control Room intakes. The 'A' train intake is at 484' and the 'B' train is at 507'-9" above grade. Ground level for the VC Summer site is 436'. Therefore, the intake height for Intake 'A' will be 48' or 14.6m and Intake 'B' 71.75' or 21.9m. All elevations are measured from the same reference point and the elevation difference is zero.

3.3.1	*Ventilation System Outside Air Intakes* All Control Room ventilation systems draw makeup air from the environment during normal operations and many draw air from the environment for the purpose of supplying filtered pressurization air. The configuration of these systems may change between normal and emergency modes. In some configurations, normal ventilation outside air intakes isolate and different intakes open to supply pressurization air. Some intake dampers may have failure modes related to loss of ac power or single failures. These considerations should be evaluated in identifying the Control Room outside air intakes for which χ/Q values should be calculated.	*Conforms* Control Room ventilation system configuration for normal and emergency modes was considered when identifying the Control Room outside air intakes for which χ/Q values should be calculated.
3.3.2	Dual Ventilation Outside Air Intakes This section applies to Control Room ventilation system configurations that have two outside air intakes, each of which meets applicable design criteria of an engineered safeguards feature (ESF), including single-failure criterion, missile protection, seismic criteria, and operability under loss-of-offsite AC power conditions. Operability requirements should be provided in Technical Specifications. The outside air intakes should be located with the intent of providing a low contamination intake regardless of wind direction. The assurance of a low contamination outside air intake depends on release point configuration, building wake effects, terrain, and the possibility of wind stagnation or wind direction reversals. The two intakes should not be within the same wind direction window, defined as a wedge centered on the line of sight between the source and the receptor with the vertex located on the release point. If ARCON96 is used, the wedge angle is 90° (i.e., 45 degrees on either side of the line of sight). If the methods of Regulatory Position 4 are used, the size of the wedge is as given in Table 2. Figure 3 illustrates four examples of the interplay between Control Room intakes, release points, and wind direction windows. In addition, the analyst should consider χ/Q values for infiltration pathways as discussed in Regulatory Position 3.3.3. The methods of this regulatory position involve identification of the limiting and favorable intakes with regard to their χ/Q value. Because of the interplay of building wake, plume rise, wind direction frequency, intake flow rate, and other parameters, it may not be possible to identify the limiting or favorable intake by observation. In these situations, χ/Q values should be calculated for each release point-intake combination and the limiting and favorable intakes identified on the basis of these values.	N/A No credit is taken in the radiological consequence analyses for use of alternate intakes during postulated accidents.

3.3.2.1	If both of the dual intakes are located within the same wind direction window, both intakes could be contaminated (See Figure 3(a)). In this case, the χ/Q values for each air intake should be calculated using ARCON96 as described in other sections of this guide and an effective χ/Q value calculated. Equation 5a should be used if the intake flow rates are equal. If the intake flow rates are not equal, but the imbalance does not shift between intakes, Equation 5b should be used. If the flow rate imbalance can shift between intakes, Equation 5c should be used. This calculation is repeated for each averaging time interval. $\overline{X/Q} = 0.5[(X/Q)_1 + (X/Q)_2]$ (5a) $\overline{X/Q} = \frac{F_1(X/Q)_1 + F_2(X/Q)_2}{F_1 + F_2}$ (5b) $\overline{X/Q} = \frac{\max(F_1, F_2) * \max[(X/Q)_1, (X/Q)_2] + \min(F_1, F_2) * \min[(X/Q)_1, (X/Q)_2]}{F_1 + F_2}$ (5c) Where: $\overline{X/Q}$ = Effective X/Q, s m^{-3} $(X/Q_1, (X/Q)_2$ = X/Q value for outside air intakes 1 and 2, s m^{-3} F_1, F_2 = Flow rate for outside air intakes 1 and 2, cfm	N/A No credit is taken in the radiological consequence analyses for use of alternate intakes during postulated accidents.
3.3.2.2	If the dual outside air intakes are not in the same wind direction window but cannot be isolated by design, the χ/Q values for the limiting outside air intake should be calculated for each time interval as described elsewhere in this guide. Equation 6a should be used if the intake flow rates are equal. If the intake flow rates are not equal, but the imbalance does not shift between intakes, Equation 6b should be used. If the flow rate imbalance can shift between intakes, Equation 6c should be used.	N/A No credit is taken in the radiological consequence analyses for use of alternate intakes during postulated accidents.

	$\overline{X/Q} = 0.5 \max [(X/Q)_1, (X/Q)_2]$ (6a) $\overline{X/Q} = \frac{\max[(F_1 (X/Q)_1, F_2 (X/Q)_2]}{F_1 + F_2}$ (6b) $\overline{X/Q} = \frac{\max (F_1, F_2) \max [(X/Q)_1, (X/Q)_2]}{F_1 + F_2}$ (6c)	
3.3.2.3	If the ventilation system design allows the operator to manually select the least contaminated outside air intake as a source of outside air makeup and close the other intake, the χ/Q values for each of the outside air intakes should be calculated for each time interval as described elsewhere in this guide. The χ/Q value for the limiting intake should be used for the time interval prior to intake isolation. This χ/Q value may be reduced by a factor of 2 to account for dilution by the flow from the other intake (see Equation 6a). The χ/Q values for the favorable intake are used for the subsequent time intervals. The χ/Q values for the favorable intake may be reduced by a factor of 4 to account for the dual inlet and the expectation that the operator will make the proper intake selection. This protocol should be used only if the dual intakes are in different wind direction windows and if there are redundant, ESF-grade radiation monitors within each intake, with Control Room indication and alarm, to monitor the intakes. The requisite steps to select the least contaminated outside air intake, and provisions for monitoring to ensure the least contaminated intake is in use throughout the event, should be addressed in procedures and in operator training. A conservative delay time should be assumed for the operator to complete the necessary actions. This delay period should consider: (1) the time for the operator to recognize the radiation monitor alarm and determine its validity (as provided for in the alarm response procedure), (2) delays associated with other accident response actions competing for the operator's attention, (3) the time needed to complete the actions, and (4) diesel generator sequencing time, if applicable. If actions are required outside the Control Room, delays associated with transit to the local control stations (including those delays caused by worker radiological protection controls associated with accident dose rates), and the availability of personnel should be considered.	N/A No credit is taken in the radiological consequence analyses for use of alternate intakes during postulated accidents.

	Note: The adjustment protocol and the numeric factors of this section are deterministic in nature and are expected to be conservative for most sites. Different factors may be considered on a case-by-case basis with sufficient justification.	
3.3.2.4	If the ventilation system design provides for automatic selection of the least contaminated outside air intake, the χ/Q values for the favorable intake should be calculated for each time interval as described elsewhere in this guide. The χ/Q values may be reduced by a factor of 10 to account for the ability to automatically select a "clean" intake. This protocol should be used only if the dual intakes are in different wind direction windows, there are redundant ESF-grade radiation monitors within each intake and an ESF-grade control logic and actuation circuitry is provided for the automatic selection of a clean intake throughout the event.	N/A No credit is taken in the radiological consequence analyses for use of alternate intakes during postulated accidents.
3.3.3	Infiltration Pathways Infiltration of contaminated air to a Control Room can be minimized by proper design and maintenance of the Control Room envelope (CRE). However, infiltration is always a possibility and the location and significance of these leakage pathways may warrant determination of χ/Q values. An unfiltered inleakage path of 100 cfm can admit the same quantity of radioactive material as a pressurization air intake having a flow of 2000 cfm through a 95 percent efficient filter. The situation can be further compounded if the χ/Q for the unfiltered pathway is more limiting than that for the Control Room outside air intake. The infiltration paths actually applicable to a particular facility will be identified via inleakage testing or CRE inspections and surveillances. Refer to Table H-1, "Determination of Vulnerability Susceptibility," of NEI 99-03, "Control Room Habitability Guidance" (Ref. 16), for further guidance on infiltration pathways. A 95th-percentile χ/Q value should be determined for each time interval for any infiltration path that could result in a significant intake of contaminated air into the CRE. Because of the interplay of source-to-receptor distance and direction, infiltration path flow rate, whether the path is filtered or unfiltered, and other considerations, it may not be possible to	Conforms

	identify the potential impact of an infiltration path by observation. In these situations, χ/Q values should be calculated for each pathway and the limiting χ/Q value(s) identified. If there is sufficient margin available, it may be possible to calculate χ/Q values assuming the shortest distance between the release point and any identified point of infiltration on the outside of the CRE.	
3.4	Determination of Source-Receptor Distances and Directions When the combinations of release points and intakes have been identified, the direction and distance between the release point and the intake should be determined. Wind direction data are recorded as the direction from which the wind blows (e.g., a north wind blows from the north; a wind blowing out of the west is recorded with a direction of 270 degrees). The direction input to ARCON96 is the wind direction that would carry the plume from the release point to the intake. For example, an analyst standing at the intake facing west to the release point, would enter 270 degrees; an analyst facing north, would enter 360 degrees, etc. The source-to-receptor distance is the shortest horizontal distance between the release point and the intake. ARCON96 will use this distance and the elevations of the source and receptor to calculate the slant path. For an area source such as building surface, the shortest horizontal distance from the building surface to the Control Room intake is used as the source-to-receptor distance. For releases within building complexes, the shortest horizontal distance between the release point and the intake could be through intervening buildings. In these cases, it is acceptable to take the length of the shortest path (e.g., "taut string length") around or over the intervening building as the source-to-receptor distance. If the distance to the receptor is less than about 10 meters, the ARCON96 code and the procedures in Regulatory Position 4 should not be used to assess χ/Q values. These situations will need to be addressed on a case-by-case basis. Note: The site meteorological tower wind direction sensors are generally calibrated with reference to true north (360 degrees). Analysts should use caution in measuring directions on site engineering drawings since these drawings typically incorporate a plant grid and a plant "north" that may not align with true north. The source-to-receptor directions input to ARCON96 must use the same north reference as the wind direction observations.	Conforms Appropriate wind directions and source-receptor distances were input into ARCON96 for determination of the χ/Qs for each of the accidents analyzed. No taut string distances were used in the χ/Q determination. All distances to the CRHE air intakes are > 10m.

4.0	ALTERNATIVE PROCEDURES FOR GROUND-LEVEL RELEASES This regulatory position addresses alternative methods for determining χ/Q values for Control Room radiological habitability assessments. The methods in Regulatory Positions 4.1 to 4.3 are based on Murphy-Campe (Ref. 2) and the Standard Review Plan Chapter 6.4 (Ref. 3).	N/A All ground level releases were determined per the preceding methodology.
4.1	Point Source-Point Receptor The 0-8 hour 95th-percentile (Note: The Murphy-Campe document identified this as the 5th-percentile /Q value.) χ/Q value for a single point source on the surface of the containment or other building and a single point receptor with a difference in elevation less than 30 percent of the building height may be estimated using Equation 7. $\frac{X}{Q} = \frac{1}{3\pi U \sigma_y \sigma_z}$ (7) Where: χ/Q = Relative concentration at plume centerline for time interval 0-8 hours, s m^{-3} 3 = Wake factor U = Wind speed at 10 meters, m s^{-1} σ_y, σ_z = Standard deviation, in meters, of the gas concentration in the horizontal and vertical cross wind directions evaluated at distance x and by stability class	N/A All ground level releases were determined per the preceding ARCON96 methodology.
4.2	Diffuse Source-Point Receptor Equation 8 may be used when the activity is assumed to leak from many points on the surface of a building such as the containment in conjunction with a single point receptor. This equation is also appropriate for point source-point receptors where the difference in elevation between the source and the receptor is greater than 30 percent of the height of the	N/A All ground level releases were determined per the preceding ARCON96 methodology and all sources were considered point sources.

	upwind building, typically the containment, which creates the most significant building wake impact. The equation is also applicable to a point source and volume receptor (e.g., an isolated Control Room with infiltration occurring at many locations). $$\frac{X}{Q} = \left[U\left(\pi\sigma_y\sigma_z + \frac{A}{K+2} \right) \right]^{-1} \quad (8)$$ Where: χ/Q =Relative concentration at plume centerline for time interval 0-8 hours, s m^{-3} U = Wind speed at 10 meters, m s^{-1} σ_y, σ_z = Standard deviation, in meters, of the gas concentration in the horizontal and vertical cross wind directions evaluated at distance x and by stability class $$K = \frac{3}{(s/d)^{1.4}}$$ s = Shortest distance between building surface and receptor location, m d = Diameter or width of building, m A = Cross-section area of building, m_2 The reference to "building" in the definitions of s, d, and A is to the diffuse source (e.g., containment). If the equation is used with a point source, the reference is to the building that has the greatest impact on the building wake. The values of the parameters σ_y, σ_z and U should be determined on the basis of the values of the site meteorological data. Some early analyses may have been based on generic meteorology conditions (e.g., F stability with wind speeds of 1.0 m·s^{-1}). If these early analyses are to be updated, the staff recommends that the ARCON96 code be used. If the ARCON96 code is not used, site-	

	specific hourly meteorological data should be used to determine the 95th-percentile χ/Q value. *Figures 4 and 5 provide sigma values by stability category for distances greater than* 10 meters. The data on these graphs should not be extrapolated for distances less than 10 meters.	
4.3	Point or Diffuse Source with Two Alternative Receptors Equations 7 and 8 of this guide may be used in conjunction with the procedures in Regulatory Position 3.3.2 to determine χ/Q values for Control Room designs having two or more Control Room outside air intakes, each of which meets the requirements of an engineered safety feature (ESF) including, as applicable, single-failure criteria for active components, seismic criteria, and missile criteria. If Equation 8 of this guide is used, the parameter K should be set to 0.0. In a change from previous practice, the staff no longer finds Equation 7 of Reference 2 to be acceptable for use in new applications.	N/A All ground level releases were determined per the preceding ARCON96 methodology.
4.4	Determination of χ/Q Values for Other Time Intervals Equations 7 and 8 are used to determine χ/Q values for the first time interval of 0-8 hours. The χ/Q values for other time intervals are obtained by adjusting for long-term meteorological averaging of wind speed and wind direction. This is accomplished by multiplying the 0-8 hour time interval χ/Q value by a correction factor for wind speed and a correction factor for wind direction. Note: Previous guidance also provided for including a factor to account for personnel occupancy factors. Since typical radiological analysis codes provide the capability to enter these factors separately, the staff recommends that the factors not be included in the χ/Q value to avoid inadvertent double crediting.	N/A All ground level releases were determined per the preceding ARCON96 methodology.
4.4.1	χ/Q Correction for Wind Speed Averaging This correction is defined as the ratio of the wind speed used to determine the 0-8 hour χ/Q value to the wind speed appropriate for each of the other time intervals. Column 2 of Table 1 tabulates the wind speed percentiles that correspond to each of these intervals. The hourly data should be arranged in order of increasing wind speed and the wind speed percentiles determined (i.e., the lowest wind speeds associated with the lowest percentiles).	N/A All ground level releases were determined per the preceding ARCON96 methodology utilizing standard time intervals; consequently, no χ/Q correction is required per wind speed averaging.

	Include only the wind speed data associated with wind directions from sectors that result in receptor contamination. Table 2 tabulates the size of the minimum wind direction window to be used. From this ranking, identify the wind speed value for each interval that is not exceeded more than the stated percentage of the time. Divide this wind speed value into the 5th-percentile wind speed used to determine the 0-8 hour χ/Q to obtain the χ/Q correction factor for wind speed. The values shown in Column 1 of Table 1 are representative correction factors that may be used if hourly observation meteorological data are not available.	

Table 1

χ/Q Correction for Wind Speed Averaging

Time Interval	Column 1 Representative χ/Q Factors	Column 2 Corresponding Wind Speed Percentile
0-8 hours	1.0	5
8-24 hours	0.67	10
1-4 days	0.50	20
4-30 days	0.33	40

Table 2

Wind Direction Sectors

s/d Ratio	Minimum Window (Note: Centered on the source-to-receptor direction.)
>2.5	68°
1.25 – 2.5	90°
0.8 – 1.25	113°
0.6 – 0.8	135°
0.5 – 0.6	158°
0.35 – 0.5	180°
<0.35	225°

	The s/d is defined as: $$\frac{s}{d} = \frac{\text{Shortest distance between building surface and receptor location, m}}{\text{Diameter or Width of building, m}}$$ The reference to "building" in Equation 9 is to the diffuse source (e.g., containment). If the equation is used with a point source, the reference is to the building that has the greatest impact on the building wake.	
4.4.2	χ/Q Correction for Wind Direction Averaging The average wind direction frequency F is obtained by summing the annual average wind direction frequencies within the minimum window. Table 2 tabulates the size of the minimum wind direction window to be used. Column 2 of Table 3 is used to determine the χ/Q correction factor for wind direction for each time interval. Column 1 is used when F has not been determined. Table 3 Wind Direction Averaging Correction Time Interval / Column 1 Representative χ/Q Factors / Column 2 Equations for χ/Q Factors 0-8 hours / 1.0 / 1.0 8-24 hours / 0.88 / 0.75 + F/4 1-4 days / 0.75 / 0.50 + F/2 4-30 days / 0.5 / F	N/A All ground level releases were determined per the preceding ARCON96 methodology utilizing standard time intervals; consequently, no χ/Q correction is required per wind direction averaging.

Table 3 (Wind Direction Averaging Correction):

Time Interval	Column 1 Representative χ/Q Factors	Column 2 Equations for χ/Q Factors
0-8 hours	1.0	1.0
8-24 hours	0.88	0.75 + F/4
1-4 days	0.75	0.50 + F/2
4-30 days	0.5	F

5.0	INSTANTANEOUS PUFF RELEASES The alternative method in this section may be used to model the release to the environment as an instantaneous puff release. One hundred percent of the radionuclides must be released directly to the environment over a period no longer than about 1 minute for a release to qualify as a puff release. Releases to enclosed buildings, intermittent releases that occur over a period longer than about 1 minute (e.g., releases from relief valves, atmospheric dumps), and releases that occur over a period longer than about 1 minute should be treated as continuous point source releases. The diffusion equation for an instantaneous puff ground level release, with no puff rise and no crosswind offset (i.e., center of puff is assumed to pass over Control Room intake), integrated over the duration of the puff passage is: Where: $$\frac{X}{Q}(x,u,k,h) = \frac{\int_0^T \frac{2}{(\sigma_z^2(x,k)+\sigma_i^2)^{1/2}\,(2\pi)^{3/2}\,(\sigma_{X,Y}^2(x,k)+\sigma_i^2)}}{\int_0^T f(t)dt} *$$ $$\exp\left[-\frac{1}{2}\left(\frac{(x-u*t)^2}{(\sigma_{x,y}^2\,(x,k)+\sigma_i^2)}+\frac{h^2}{(\sigma_z^2\,(x,k)+\sigma_i^2)}\right)\right] F(t)dt$$	N/A All ground level releases were determined per the preceding ARCON96 methodology. No puff releases are assumed.

	$\frac{X}{Q}(x, u, x, h)$ = Effective puff relative concentration, s m^{-3} X = Integrated concentration at control intake, Ci $m^{-3}s^{-1}$ Q_i = Relative quantity, for nuclide i, Ci x = Release point to receptor distance, m u = Wind speed, m/sec. Assume 1.0 m s^{-1} k = Stability Class. Assume F. h = Difference in elevation between the physical release point and the control room intake, m. If the control room intake is at a higher elevation than the release point and the puff is bouyant, assume h = 0. T = Time for trailing edge of puff to pass control room intake, sec. $= \frac{x + 3[\sigma_{x,y}(x, k) + \sigma_i]}{u}$ F = Control room total intake flow rate, cfm. (If the control room intake flow rate is constant over the period 0 to T seconds, the F(t) terms can be omitted from Equation 10. $\sigma_{x,y}(x, k)$ = Standard deviation, m, of the puff in the horizontal along the wind direction and cross – wind directions at the receptor locations. Use Figure 4 with the distance x and Stability Class K to determine $\sigma_{x,y}$ at the receptor, e.g., $\sigma_{x,y} = \sigma_y$. $\sigma_z(x, k)$ = Standard deviation, m, of the puff in the vertical cross – wind direction at the receptor location. Use Figure 5 with the distance x and Stability Class K to determine σ_z at the receptor.	

	σ_i = Initial standard deviation, m $= \left[\dfrac{2V}{(2\pi 2^{3/2}} \right]^{1/3}$ V = Initial puff volume (expanded to standard atmospheric conditions), m^3 (The puff dimensions that would exist when the puff is at the control room intake are assumed to exist during the entire puff transit. Equation 10 provides the *effective* relative concentration for the puff. This value can be input to dose assessment codes such as RADTRAD or HABIT as any value of X/Q would be if the intake flows, release duration, and release rates are modeled consistent with the inputs to Equation 10.	
6.0	PLUME RISE An applicant or licensee may propose adjustments to the release height for plume rise that are due to buoyancy or mechanical jet on a case-by-case basis. In order to credit these adjustments, the applicant or licensee must be able to demonstrate that the assumed buoyancy or vertical velocity of the effluent plumes will be maintained throughout the time intervals that plume rise is credited. Such justifications need to consider the availability of AC power, failure modes of dampers and ductwork, time-dependent release stream temperatures and pressures, and 95^{th}-percentile wind speeds and ambient temperatures. (Note: As used here, 95^{th}-percentile wind speed is that wind speed that is not exceeded more than 5 percent of the time. A 95^{th}-percentile ambient temperature is that temperature that is not exceeded more than 5 percent of the time). Plume rise may be considered for freestanding stacks and for vents located on plant buildings. However, plume rise may not be used in demonstrating that a particular stack meets the 2-1/2 times the adjacent structure height criterion in Regulatory Position 3.2.2. A mixed-mode release model, such as that in Regulatory Guide 1.111 (Ref. 10), should not be used for design basis assessments. The plume rise may be determined through the use of the following set of equations (Ref. 17). The plume rise for plant vents is determined using Equation 11. The distance x is entered as the horizontal distance between the vent and the Control Room outside air intake.	N/A Plume rise was not considered in the χ/Q determinations.

	The plume rise for isolated, free-standing stacks is calculated using Equations 11, 12, and 13. The distance x in Equation 11 should be based on the downwind location corresponding to the maximum X/Q value. See Regulatory Position 3.2.2. The plume rises calculated using Equations 12 and 13 should be compared and the larger plume rise identified. The result of this comparison is then compared to the plume rise determined using Equation 11 and the smaller plume rise selected for use. $$\Delta h = \left[\frac{3}{\beta_1^2} * \frac{F_m}{U^2} * x + \frac{3}{2\beta_1^2} * \frac{F_b}{U^3} * x^2 \right]^{1/3} \quad (11)$$ $$\Delta h = 2.6 \left(\frac{F_b}{U_s} \right)^{1/3} \quad (12)$$ $$\Delta h = 2.44 \left(\frac{F_m}{s} \right)^{1/4} \quad (13)$$	

	Where: Δh = Plume rise, m F_m = Momentum flux parameter, $m^4\ s^{-2}$ $= \frac{\rho_0 V_0 w_0}{\pi \rho_a}$ β_1 = Dimensionless entrainment constant for momentum = 0.6 U = Wind speed at release height, $m\ s^{-1}$ x = Distance from release point to receptor, m F_b = Bouyancy flux parameter, $m^4\ s^{-3}$ $= \frac{g(\rho_a - \rho_0)V_0}{\pi \rho_a}$ w_0 = Effluent exit velocity, ms^{-1} V_0 = Volumetric release rate, $m^3\ s^{-1}$ ρ_0 = Effluent density after expansion to atmospheric pressure, $kg\ m^{-3}$ ρ_a = Density of air, $kg\ m^{-3}$ s = 0.0001 s^{-2} for A, B, C, and D stability; 0.00049 s^{-2} for E stability; 0.0013 s^{-2} for F stability; 0.002 s^{-2} for G stability g = Gravitational acceleration, 9.8 $m\ s^{-2}$ Although ARCON96 processes ambient meteorological conditions on an hour-by-hour basis, the code cannot vary the other parameters that enter into a plume rise determination. For example, wind speed and stability class are varied hour by hour, but the density of air, the density of the effluent stream, and the vertical velocity are not varied hour-by-hour. As such, the analyst should ensure that these parameters are bounding for the entire period of the X/Q assessment or use individual time intervals to model the time-variant parameters. An alternative approach would be to calculate the plume rise for each hour independently of ARCON96 and to select a plume rise that is exceeded more than 95 percent of the time.	

	This rise is then added to the stack height as input to ARCON96. In lieu of mechanistically addressing the amount of buoyant plume rise associated with energetic releases from steam relief valves or atmospheric dump valves, the ground level X/Q value calculated with ARCON96 (on the basis of the physical height of the release point) may be reduced. (Note: This adjustment factor and the associated velocity ratio criterion are deterministic in nature and their selection was based on sensitivity analyses performed for typical steam release points at LWRs.) The adjustment factor should not be ratioed for different vertical velocity ratios) by a factor of 5. This reduction may be taken only if (1) the release point is uncapped and vertically oriented and (2) the time-dependent vertical velocity exceeds the 95th-percentile wind speed. [Note: As used here, 95th-percentile wind speed is that wind speed that is not exceeded more than 5 percent of the time. A 95th-percentile ambient temperature is that temperature that is not exceeded more than 5 percent of the time (at the release point height) by a factor of 5].	
7.0	USE OF SITE-SPECIFIC EXPERIMENTAL DATA The methods and parameters provided in this guide are acceptable for use for design basis Control Room habitability radiological assessments provided that all stated prerequisites and conditions are met. The staff believes that use of the guidance in this guide will result in X/Q values that are acceptably conservative. However, there may be circumstances in which these methods and parameters may not be advantageous for a particular plant configuration and site meteorological regimes and may lead to results that are deemed to be unnecessarily conservative. Licensees and applicants may opt to propose alternative methods and parameters such as those that are based in part on data obtained from site-specific experimental measurements. Data based on wind tunnel tests should be accompanied with an evaluation of the representativeness of the experiment results to the particular plant configuration and site meteorological regimes. These proposed alternatives, with supporting data, will be considered by the staff on a case-by-case basis. The staff recommends that licensees considering an experimental program request a meeting with the staff in advance of starting the program. The intent of this recommendation is to allow the staff and the licensee (or applicant) to discuss the proposed program, prior to resource expenditure, and for the staff to provide a preliminary assessment of the proposal. The staff's approval of the proposed alternative methods and	N/A No experimental data was utilized to calculate the χ/Qs.

	parameters will not be granted, however, until the licensee or applicant completes the experimental program and dockets the proposal with supporting analyses and data for formal staff review. An acceptable experimental program should incorporate the following standards:	
7.1	The experimental program should be appropriately structured so as to provide data of appropriate quantity and quality to support data analysis and conclusions drawn from that data. The program should be developed by personnel who have educational and work experience credentials in air dispersion meteorology and modeling.	N/A No experimental data was utilized to calculate the χ/Qs.
7.2	The experimental program should encompass a sufficient range of meteorological conditions applicable to the particular site so as to ensure that the data obtained address the site-specific meteorological regimes and the site-specific release point/receptor configurations that impact the Control Room X/Q values. Meteorological conditions observed at the particular site with a frequency of 5 percent or greater in a year should be addressed. Parameters derived from statistical analyses on the experimental data should represent the 95^{th}-percentile confidence level.	N/A No experimental data was utilized to calculate the χ/Qs.
7.3	The experimental program, including data reduction and analysis, should incorporate applicable quality control criteria of Appendix B to 10 CFR Part 50. The products of the experimental program should be verified and validated.	N/A No experimental data was utilized to calculate the χ/Qs.

Table A-2
ARCON96 INPUT PARAMETERS FOR DESIGN BASIS ASSESSMENTS

Parameter/Discussion/ Acceptable Input	Basis of Compliance
Lower Measurement Height, meters The value of this parameter is used by ARCON96 to adjust wind speeds for differences between the heights of the instrumentation and the release. Use the actual instrumentation height when known. Otherwise, assume 10 meters.	Conforms The actual instrumentation lower measurement height of 10 meters was utilized.
Upper Measurement Height, meters The value of this parameter is used by ARCON96 to adjust wind speeds for differences between the heights of the instrumentation and the release. Use the actual instrumentation height when known. Otherwise, use the height of the containment or the stack height, as appropriate. If wind speed measurements are available at more than two elevations, the instrumentation at the height closest to the release height should be used.	Conforms The actual instrumentation upper measurement height of 61 meters was utilized.
Wind Speed Units ARCON96 requires that wind speed be entered as miles per hour, m·s-1, or knots. Use the wind speed units that correspond to the units of the wind speeds in the meteorological data file.	Conforms Wind speed was entered as meters per second, which corresponds to the units of the wind speeds in the meteorological data files.

Release Height, meters The value of the release height is used for three purposes in ARCON96: (1) to adjust wind speeds for differences between the heights of the instrumentation and the release, (2) to determine slant path for ground level releases, (3) to correct off-centerline data for elevated releases. Use the actual release heights whenever available. Plume rise from buoyancy and mechanical jet effects may be considered in establishing the release height if the analyst can demonstrate with reasonable assurance that the vertical velocity of the release will be maintained during the course of the accident. If actual release height is not available, set release height equal to intake height.	Conforms For the ten source-receptor locations, the actual release height was utilized or the release height was conservatively assumed to be 0 (assume release height equals intake height). Plume rise from buoyancy and mechanical jet effects was not considered in establishing the release height.
Building Area, meters^2 ARCON96 uses the value of the building area in the high speed wind speed adjustment for ground-level and vent release models. Use the actual building vertical cross-sectional area perpendicular to the wind direction. Use default of 2000 m^2 if the area is not readily available. Do not enter zero. Use 0.01 m^2 if a zero entry is desired. *Note: This building area is for the building(s) that has the largest impact on the building wake within the wind direction window. This is usually, but need not always be, the reactor containment. With regard to the diffuse area source option, the building area entered here may be different from that used to establish the diffuse source.*	Conforms The actual building vertical cross-sectional area perpendicular to the wind direction was conservatively calculated. A value of 1,740 m^2 was utilized.

Vertical Velocity, meters/second In ARCON96, the value of the vertical velocity is used only in vent and stack release models. It is used for the downwash calculation. In the vent release model the velocity is used in the mixed-mode calculation. If the vertical velocity is set to zero, the maximum downwash will be calculated and the release height will be reduced by an amount equal to six times the stack radius. *Note: The vent release model should not be used for DBA accident calculations.* For stack release calculations only, use the actual vertical velocity if the licensee can demonstrate with reasonable assurance that the value will be maintained during the course of the accident (e.g., addressed by technical specifications), otherwise, enter zero. If the vertical velocity is set to zero, ARCON96 will reduce the stack height by 6 times the stack radius for all wind speeds. If this reduction is not desired, the stack radius should also be set to zero.	N/A Vent and stack release models were not utilized. All releases were ground level.
Stack Flow, meters3/second ARCON96 uses the value of the stack flow in X/Q calculations for all 3 release types to ensure that the near field concentrations are no greater than the concentration at the release point. The impact diminishes with increasing distance. Use actual flow if it can be demonstrated with reasonable assurance that the value will be maintained during the course of the accident (e.g., addressed by Technical Specifications). Otherwise, enter zero. The flow is used in both elevated and ground-level release modes to establish a maximum X/Q value. This value is significant only if the flow is large and the distance from the release point to the receptor is small.	N/A Vent and stack release models were not utilized. All releases were ground level.

Stack Radius, meters ARCON96 uses the value of the stack radius in downwash calculations in the vent and stack release modes. Use the actual stack internal radius when both the stack radius and vertical velocity are available. If the stack flow is zero, the radius should be set to zero.	N/A Vent and stack release models were not utilized. All releases were ground level.
Distance to Receptor, meters The value of horizontal distance to the receptor from the release point is used in ARCON96 for calculating the slant range for ground level releases and the off-centerline correction factors for stack release models. Use the actual straight-line horizontal distance between the release point and the Control Room intake. For ground-level releases, it may be appropriate to consider flow around an intervening building if the building is sufficiently tall that it is unrealistic to expect flow from the release point to go over the building. *Note: If the distance to receptor is less than about 10 meters, ARCON96 should not be used to assess relative concentrations.*	Conforms The actual straight-line horizontal distances between the release point and the Control Room are utilized (no taut string distances even for releases in the building complex) to calculate the χ/Qs. All source to receptor distances are greater than 10 meters.
Intake Height, meters The value of the intake height is used in ARCON96 for calculating the slant range for ground level releases and the off-centerline correction factors for stack release models. Use the actual intake height. If the intake height is not available for ground level releases, assume the intake height is equal to the release height. For elevated releases, assume the height of the tallest site building.	Conforms The actual Control Room intake heights were utilized to calculate the χ/Qs.

Elevation Difference, meters The value of this parameter is used by ARCON96 to normalize the release heights and the intake heights when the two heights are specified as "above grade" with different grades for the release point and intake height, or when one measurement is referenced to "above grade" and the other to "above sea level." Use zero unless it is known that the release heights are reported relative to different grades or reference data.	Conforms Actual release heights and the intake heights and terrain elevation differences were utilized as appropriate to determine χ/Qs.
Direction to Source, degrees ARCON96 uses the value of this parameter and the Wind Direction Window to establish which range of wind directions should be included in the assessment of the X/Q. Use the direction FROM the intake back TO the release point. (Wind directions are reported as the direction from which the wind is blowing. Thus, if the direction from the intake to the release point is north, a north wind will carry the plume from the release point to the intake). Note: Some facilities have a "plant north" shown on site arrangement drawings that is different from "true north." The direction entered must have the same point of reference as the wind directions reported in the meteorological data. For ground-level releases, if the plume is assumed to flow around a building rather than over it, the direction may need to be modified to account for the redirected flow. In this case, the X/Q should be calculated assuming flow around and flow over (through) the building and the higher of the two X/Q s should be used.	Conforms Wind direction from the intakes back to the release point was utilized to calculate the χ/Qs. Plant north and true north are considered the same at VCSNS. No analyses considered ground-level releases that flow around a building rather than over it.

Surface Roughness Length, meters ARCON96 uses the value of this parameter in adjusting wind speeds to account for differences in meteorological instrumentation height and release height. Use a value of 0.2 in lieu of the default value of 0.1 for most sites. (Reasonable values range from 0.1 for sites with low surface vegetation to 0.5 for forest-covered sites).	Conforms A surface roughness length of 0.2 in lieu of the default value of 0.1 was utilized to calculate the χ/Qs.
Wind Direction Window, degrees Code Default ARCON96 uses the value of this parameter and the Direction to Source to establish which range of wind directions should be included in the assessment of the X/Q. Use the default window of 90 degrees (45 degrees on either side of line of sight from the source to the receptor).	Conforms Used the default values.
Minimum Wind Speed, meters/second Code Default ARCON96 uses the value of this parameter to identify calm conditions. Use the default wind speed of 0.5 m·s-1 (regardless of the wind speed units entered earlier), unless there is some indication that the anemometer threshold is greater than 0.6 $m \cdot s^{-1}$.	Conforms Used the default values.

Averaging Sector Width Constant Code Default ARCON96 uses the value of this parameter to prevent inconsistency between the centerline and sector average X/Q s for wide plumes. Has largest effect on ground level plumes. Although the default value is 4, a value of 4.3 is preferred. (A future revision to ARCON96 will change the default to 4.3).	Conforms An averaging sector width constant of 4.3 (the preferred value) was utilized to calculate the χ/Qs.
Initial Diffusion Coefficients, meters ARCON96 uses these parameters in modeling a diffuse source. These values will normally be set to zero. If the diffuse source option is being used, see Regulatory Position 2.2.4.	Conforms The initial diffusion coefficients are set to zero since only point sources were utilized in the evaluation.
Hours in Averages Code Default The values of this parameter were selected to provide results for desired periods and to provide a smooth X/Q curve. Use the default values.	Conforms Used the default values.
Minimum Number of Hours Code Default The default values of this parameter will allow processing with up to 10% missing data. Use the default values.	Conforms Used the default values.

Attachment 5

Safety Assessment for the Proposed Technical Specification And Bases Changes

Proposed Technical Specification and Bases Changes

A description of each proposed TS change and the associated basis/safety assessment are included in Table 5-1.

The majority of the Technical Specification changes are revised in accordance with the Improved Standard Technical Specifications Change Traveler (TSTF-51, Revision 2) which permits removal of the Technical Specification requirements for ESF features to be OPERABLE after sufficient radioactive decay has occurred to ensure off-site doses remain below the SRP limits.

Associated with this change is the deletion of OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features. This change allows flexibility to move personnel and equipment and perform work which would affect containment OPERABILITY during the handling of irradiated fuel.

Following reactor shutdown, decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed changes are based on performing analyses assuming a longer decay period to take advantage of the reduced radionuclide inventory available for release in the event of a fuel handling accident. Following sufficient decay, the primary success path for mitigating the fuel handling accident no longer includes the functioning of the active containment systems. Therefore, the OPERABILITY requirements of the Technical Specifications are modified to reflect that water level (23') and decay time (72 hours after shutdown) are the primary elements for the success path for mitigating a fuel handling accident.

To support this change in requirements during the handling of irradiated fuel, the OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features are deleted. The accidents postulated to occur during core alterations, in addition to fuel handling accidents, are: inadvertent criticality (due to a control rod removal error or continuous control rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage.

Also, the Technical Specifications only allow the handling of irradiated fuel in the reactor vessel when the water level in the reactor cavity is at the high water level. Therefore, the proposed changes only affect containment requirements during periods of relatively low shutdown risk during refueling outages. Therefore, the proposed changes do not significantly increase the shutdown risk.

Other Technical Specification revisions reflect the update of the accident source term and associated design basis accidents utilizing the guidance provided in USNRC Regulatory Guide 1.183 and the associated control room and offsite dose requirements of 10 CFR 50.67.

The preceding discussion in Attachment 1, the Safety Assessment in Attachment 2, and the calculations in Attachment 10 support these changes.

Technical Specification changes to the following sections and tables are proposed:

Index
Table 3.3-6
Table 4.3-3
Section 3/4.7.11
Section 3/4.9.4
Section 3/4.9.8
BASES 3/4.4.5
BASES 3/4.4.6.2
BASES 3/4.6.1.1
BASES 3/4.7.1.4
BASES 3/4.7.10
BASES 3/4.7.11
BASES 3/4.9.4
BASES 3/4.9.8
BASES 3/4.9.9
ADMINISTRATIVE CONTROLS 6.8.4.1

Table 5-1: Proposed Technical Specification and Bases Changes

Description and Safety Assessment for Specific Changes to TS and TS Bases		
Change #1	**Current Technical Specification:** INDEX ENTRIES MAIN SECTION ENTRIES: 3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM, 3/4.9.4 REACTOR BUILDING PENETRATIONS and 3/4.9.8, REACTOR BUILDING PURGE AND EXHAUST ISOLATION SYSTEM BASES SECTION ENTRIES: 3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM, 3/4.9.4 REACTOR BUILDING PENETRATIONS and 3/4.9.8, REACTOR BUILDING PURGE AND EXHAUST ISOLATION SYSTEM	**Proposed Change:** INDEX, MAIN SECTION ENTRIES: 3/4.7.11 deleted, 3/4.9.4 deleted and 3/4.9.8 deleted. INDEX, BASES SECTION ENTRIES: 3/4.7.11 deleted, 3/4.9.4 deleted and 3/4.9.8 deleted.
Change #1	**Basis / Safety Assessment:** The index changes are administrative and reflect changes to the VCSNS Limiting Conditions for Operations and Surveillance Requirements and their associated Bases that are supported by this submittal. The basis for each technical change is discussed below.	
Change #2	**Current Technical Specification:** TABLE 3.3-6, RADIATION MONITORING INSTRUMENTATION, INSTRUMENT, 1, AREA MONITORS, b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B) and associated Action 28	**Proposed Change:** Delete these entries from the Table.
Change #2	**Basis / Safety Assessment:** Calculation DC00040-102, "Fuel Handling Accidents – AST" (see Attachment 10). This new dose analysis, performed in accordance with Regulatory Guide 1.183, assumes (1) no filtration or radionuclide removal for the release and (2) the containment is open for the duration of the event. Under these assumptions, the resulting Control Room and offsite doses are within the regulatory limits of 10CFR50.67. Other accidents considered during Mode 6 are inadvertent criticality due to boron dilution and inadvertent loading of a fuel assembly in an improper location. Neither of these accidents are postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during Mode 6 that results in a significant radioactive release is the Fuel Handling Accident, the proposed Technical Specification change omitting the use of RM-G17A or RM-G17B to initiate automatic isolation of the Containment Purge and Exhaust is justified. This change is consistent with TSTF-51 which allows deletion of OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features previously credited for the FHA.	

<table>
<tr><th colspan="3">Description and Safety Assessment for Specific Changes to TS and TS Bases</th></tr>
<tr><td>Change #3</td><td>Current Technical Specification:
TABLE 3.3-6, RADIATION MONITORING INSTRUMENTATION, INSTRUMENT, 2, PROCESS MONITORS, a. Spent Fuel Pool Exhaust – Ventilation System (RM-A6),
i. Gaseous Activity, ii. Particulate Activity and associated Action 27</td><td>Proposed Change:
Delete these entries from the Table.</td></tr>
<tr><td>Change #3</td><td colspan="2">Basis / Safety Assessment:
Calculation DC00040-102, “Fuel Handling Accidents – AST” (see Attachment 10). This new dose analysis, performed in accordance with Regulatory Guide 1.183, assumes no filtration or radionuclide removal for the Fuel Handling Building release, and the resulting Control Room and offsite doses are within the regulatory limits of 10CFR50.67. Other than the Fuel Handling Accident, no other accident involving movement of irradiated fuel in the spent fuel pool and during crane operation with loads over the pool are predicted/postulated to result in a significant radioactive release. Therefore, removal of the OPERABILITY requirements for the Spent Fuel Pool Exhaust Ventilation System radiation monitor (RM-A6) is justified. This is consistent with intent of TSTF-51 which allows deletion of OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features previously credited for the FHA.</td></tr>
<tr><td>Change #4</td><td>Current Technical Specification:
TABLE 3.3-6, RADIATION MONITORING INSTRUMENTATION, INSTRUMENT, 2, PROCESS MONITORS, b. Containment, i. Gaseous Activity – Purge & Exhaust Isolation (RM-A4)</td><td>Proposed Change:
Delete this entry from the Table.</td></tr>
<tr><td>Change #4</td><td colspan="2">Basis / Safety Assessment:
New dose analyses (see Attachment 10) for the six analyzed accidents are enclosed and proposed for use in Chapter 15 of the FSAR. The new analyses are performed utilizing the guidance of Regulatory Guide 1.183 to meet the 10 CFR 50.67 regulatory requirements. During the FHA, the containment is assumed to be open for the duration of the event. Other accidents considered during Mode 6 are inadvertent criticality due to boron dilution and inadvertent loading of a fuel assembly in an improper location. Neither of these accidents are postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during Mode 6 that results in a significant radioactive release is the Fuel Handling Accident, the proposed Technical Specification change omitting the use of RM-A4 to initiate automatic isolation of the Containment Purge and Exhaust is justified. This change is consistent with TSTF-51 which allows deletion of OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features previously credited for the FHA.</td></tr>
</table>

Description and Safety Assessment for Specific Changes to TS and TS Bases		
Change #5	**Current Technical Specification:** TABLE 4.3-3, RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS, INSTRUMENT 1, AREA Monitors, b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	**Proposed Change:** Delete these entries from the Table.
Change #5	**Basis / Safety Assessment:** Change 2 above deletes the operational requirements of process radiation monitors RM-G17A and RM-G17B since automatic isolation of the Containment Purge and Exhaust is no longer required for mitigation of the Fuel Handling Accident. For the same reasons documented under Change 2, the associated surveillance requirements for RM-G17A and RM-G17B in Table 4.3-3 are no longer required.	
Change #6	**Current Technical Specification:** TABLE 4.3-3, RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS, INSTRUMENT, 2, PROCESS MONITORS, a. Spent Fuel Pool Exhaust – Ventilation System (RM-A6), i. Gaseous Activity and ii. Particulate Activity	**Proposed Change:** Delete these entries from the Table.
Change #6	**Basis / Safety Assessment:** Change 3 above deletes the operational requirements of process radiation monitor RM-A6 since filtration or radionuclide removal for the Fuel Handling Building release is no longer required for mitigation of the Fuel Handling Accident. For the same reasons documented under Change 3, the associated surveillance requirements for RM-A6 in Table 4.3-3 are no longer required.	
Change #7	**Current Technical Specification:** TABLE 4.3-3, RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS, INSTRUMENT, 2, PROCESS MONITORS, b. Containment, i. Gaseous Activity – Purge & Exhaust Isolation (RM-A4)	**Proposed Change:** Delete this entry from the Table.
Change #7	**Basis / Safety Assessment:** Change 4 above deletes the operational requirements of process radiation monitor RM-A4 since automatic isolation of the Containment Purge and Exhaust is no longer required for mitigation of the Fuel Handling Accident. For the same reasons documented under Change 4, the associated Mode 6 surveillance requirements for RM-A4 in Table 4.3-3 are no longer required.	

Description and Safety Assessment for Specific Changes to TS and TS Bases		
Change #8	**Current Technical Specification:** 3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM, LIMITING CONDITIONS FOR OPERATION and SURVEILLANCE REQUIREMENTS	**Proposed Change:** Delete this section in its entirety.
Change #8	**Basis / Safety Assessment:** Calculation DC00040-102, "Fuel Handling Accidents – AST" (see Attachment 10). This new dose analysis, performed in accordance with Regulatory Guide 1.183, assumes no filtration or radionuclide removal for the Fuel Handling Building release, and the resulting Control Room and offsite doses are within the regulatory limits of 10CFR50.67. Other than the Fuel Handling Accident, no other accident involving movement of irradiated fuel in the spent fuel pool and during crane operation with loads over the pool are predicted/postulated to result in a significant radioactive release. Therefore, removal of the OPERABILITY requirements for the Spent Fuel Pool Ventilation Sub-Systems is justified. This is consistent with intent of TSTF-51 which allows deletion of OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features previously credited for the FHA.	
Change #9	**Current Technical Specification:** REFUELING OPERATIONS, 3/4.9.4 REACTOR BUILDING PENETRATIONS	**Proposed Change:** Delete this section in its entirety.
Change #9	**Basis / Safety Assessment:** Calculation DC00040-102, "Fuel Handling Accidents – AST" (see Attachment 10). This new dose analysis, performed in accordance with Regulatory Guide 1.183, assumes no filtration, holdup, or radionuclide removal for the Reactor Building release, and the resulting Control Room and offsite doses are within the regulatory limits of 10CFR50.67. Other accidents considered during CORE ALTERATIONS or movement of irradiated fuel within the reactor building are inadvertent criticality due to boron dilution and inadvertent loading of a fuel assembly in an improper location. Neither of these accidents is postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur that results in a significant radioactive release is the Fuel Handling Accident, the proposed Technical Specification change omitting OPERABILITY requirements for Reactor Building Penetrations is justified. This change is consistent with TSTF-51 which allows deletion of OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features previously credited for the FHA.	

<table>
<tr><th colspan="3">Description and Safety Assessment for Specific Changes to TS and TS Bases</th></tr>
<tr><td>Change #10</td><td>Current Technical Specification:
REFUELING OPERATIONS, 3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM</td><td>Proposed Change:
Delete this section in its entirety.</td></tr>
<tr><td>Change #10</td><td colspan="2">Basis / Safety Assessment:
Calculation DC00040-102, "Fuel Handling Accidents – AST" (see Attachment 10). This new dose analysis, performed in accordance with Regulatory Guide 1.183, assumes no filtration, holdup, or radionuclide removal for the Reactor Building release, and the resulting Control Room and offsite doses are within the regulatory limits of 10CFR50.67. Other accidents considered during CORE ALTERATIONS or movement of irradiated fuel within containment are inadvertent criticality due to boron dilution and inadvertent loading of a fuel assembly in an improper location. Neither of these accidents is postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur that results in a significant radioactive release is the Fuel Handling Accident, the proposed Technical Specification change omitting OPERABILITY requirements for the Reactor Building Purge Supply and Exhaust Isolation System is justified. This change is consistent with TSTF-51 which allows deletion of OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features previously credited for the FHA.</td></tr>
<tr><td>Change #11</td><td>Current Technical Specification:
REACTOR COOLANT SYSTEM, BASES, STEAM GENERATOR TUBE INTEGRITY, Applicable Safety Analyses, 2nd paragraph, "The dose consequences of these events are within the limits of GDC 19 (Reference 2), 10 CFR 100 (Reference 3) or the NRC approved licensing basis (e.g., a small fraction of these limits)."</td><td>Proposed Change:
REACTOR COOLANT SYSTEM, BASES, STEAM GENERATOR TUBE INTEGRITY, Applicable Safety Analyses, 2nd paragraph, "The dose consequences of these events are within the limits of GDC 19 (Reference 2), 10 CFR 50.67 (Reference 3) or the NRC approved licensing basis (e.g., a small fraction of these limits)."</td></tr>
<tr><td>Change #11</td><td colspan="2">Basis / Safety Assessment:
The enclosed steam generator tube rupture dose (see Attachment 10, Calculation DC00040-98, "Steam Generator Tube Rupture – AST") analysis is performed utilizing the guidance of Regulatory Guide 1.183 to meet the 10 CFR 50.67 regulatory requirements. This change updates the BASES section to reflect the appropriate regulatory requirements.</td></tr>
<tr><td>Change #12</td><td>Current Technical Specification:
REACTOR COOLANT SYSTEM, BASES, STEAM GENERATOR TUBE INTEGRITY, References, 3, 10 CFR 100, "Reactor Site Criteria"</td><td>Proposed Change:
REACTOR COOLANT SYSTEM, BASES, STEAM GENERATOR TUBE INTEGRITY, References, 3, 10 CFR 50.67, "Accident Source Term"</td></tr>
<tr><td>Change #12</td><td colspan="2">Basis / Safety Assessment:
The enclosed steam generator tube rupture dose (see Attachment 10, Calculation DC00040-98, "Steam Generator Tube Rupture – AST") analysis is performed utilizing the guidance of Regulatory Guide 1.183 to meet the 10 CFR 50.67 regulatory requirements. This change supports change #11 above by updating Reference 3 to reflect the appropriate regulatory requirements.</td></tr>
</table>

<table>
<tr><th colspan="3">Description and Safety Assessment for Specific Changes to TS and TS Bases</th></tr>
<tr><td>Change #13</td><td>Current Technical Specification:
REACTOR COOLANT SYSTEM, BASES, OPERATIONAL LEAKAGE, Applicable Safety Analyses, 4th paragraph, "The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100, or the staff approved licensing basis (i.e., a small fraction of these limits)."</td><td>Proposed Change:
REACTOR COOLANT SYSTEM, BASES, OPERATIONAL LEAKAGE, Applicable Safety Analyses, 4th paragraph, "The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50.67, or the staff approved licensing basis (i.e., a small fraction of these limits)."</td></tr>
<tr><td>Change #13</td><td colspan="2">Basis / Safety Assessment:
The enclosed main stream line break dose (see Attachment 10, Calculation DC00040-99, "Main Steam Line Break – AST") analysis is performed utilizing the guidance of Regulatory Guide 1.183 to meet the 10 CFR 50.67 regulatory requirements. This change updates the BASES section to reflect the appropriate regulatory requirements.</td></tr>
<tr><td>Change #14</td><td>Current Technical Specification:
CONTAINMENT SYSTEMS, BASES, 3/4.6.1.1 CONTAINMENT INTEGRITY, "This restriction, in conjunction with the leak rate limitation, will limit site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions."</td><td>Proposed Change:
CONTAINMENT SYSTEMS, BASES, 3/4.6.1.1 CONTAINMENT INTEGRITY, "This restriction, in conjunction with the leak rate limitation, will limit site boundary radiation doses to within the limits of 10 CFR 50.67 during accident conditions."</td></tr>
<tr><td>Change #14</td><td colspan="2">Basis / Safety Assessment:
New dose analyses (see Attachment 10) for the six analyzed accidents are enclosed and proposed for use in Chapter 15 of the FSAR. The new analyses are performed utilizing the guidance of Regulatory Guide 1.183 to meet the 10 CFR 50.67 regulatory requirements. This change updates the BASES section to reflect the appropriate regulatory requirements.</td></tr>
<tr><td>Change #15</td><td>Current Technical Specification:
PLANT SYSTEMS, BASES, 3/4.7.1.4 ACTIVITY, "The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR 100 limits in the event of a steam line rupture."</td><td>Proposed Change:
PLANT SYSTEMS, BASES, 3/4.7.1.4 ACTIVITY, "The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to 10 CFR 50.67 limits in the event of a steam line rupture."</td></tr>
<tr><td>Change #15</td><td colspan="2">Basis / Safety Assessment:
New dose analyses (see Attachment 10) for the six analyzed accidents are enclosed and proposed for use in Chapter 15 of the FSAR. The new analyses are performed utilizing the guidance of Regulatory Guide 1.183 to meet the 10 CFR 50.67 regulatory requirements. This change updates the BASES section to reflect the appropriate regulatory requirements.</td></tr>
</table>

Description and Safety Assessment for Specific Changes to TS and TS Bases		
Change #16	**Current Technical Specification:** PLANT SYSTEMS, BASES, 3/4.7.10 WATER LEVEL – SPENT FUEL POOL, "The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis."	**Proposed Change:** PLANT SYSTEMS, BASES, 3/4.7.10 WATER LEVEL – SPENT FUEL POOL, "The restrictions on minimum water level ensure that sufficient water depth is available to remove 99.5% of the assumed 16% I-131 and 10% other halogens gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis."
Change #16	**Basis / Safety Assessment:** A new dose analysis (see Attachment 10, DC00040-102, "Fuel Handling Accidents – AST") for the Fuel Handling Accident is enclosed and proposed for use in Chapter 15 of the FSAR. The analysis is performed utilizing the guidance of Regulatory Guide 1.183 to meet the 10 CFR 50.67 regulatory requirements. The analysis utilizes iodine removal factors and gap activities that differ from the CLB assumptions currently shown in BASES Section 3/4.7.10. This change updates the BASES section to reflect the revised analysis assumptions.	
Change #17	**Current Technical Specification:** PLANT SYSTEMS, BASES, 3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM	**Proposed Change:** Delete this section in its entirety.
Change #17	**Basis / Safety Assessment:** Change #8 deletes all Limiting Conditions for Operation and Surveillance Requirements for the Spent Fuel Pool Ventilation System since they are no longer needed for mitigation of the Fuel Handling Accident. This change deletes the associated BASES section (3/4.7.11) for this specification.	
Change #18	**Current Technical Specification:** REFUELING OPERATIONS, BASES, 3/4.9.4 REACTOR BUILDING PENETRATIONS	**Proposed Change:** Delete this section in its entirety.
Change #18	**Basis / Safety Assessment:** Change #9 deletes all Limiting Conditions for Operation and Surveillance Requirements for Reactor Building Penetrations since they are no longer needed for mitigation of the Fuel Handling Accident. This change deletes the associated BASES section (3/4.9.4).	
Change #19	**Current Technical Specification:** REFUELING OPERATIONS, BASES, 3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM	**Proposed Change:** Delete this section in its entirety.
Change #19	**Basis / Safety Assessment:** Change #10 deletes all Limiting Conditions for Operation and Surveillance Requirements for the Reactor Building Purge Supply and Exhaust Isolation System since the isolation function is no longer needed for mitigation of the Fuel Handling Accident. This change deletes the associated BASES section (3/4.9.8).	

<table>
<tr><th colspan="3">Description and Safety Assessment for Specific Changes to TS and TS Bases</th></tr>
<tr><td>Change #20</td><td>Current Technical Specification:
REFUELING OPERATIONS, BASES, 3/4.9.9 WATER LEVEL – REACTOR VESSEL, "The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis."</td><td>Proposed Change:
REFUELING OPERATIONS, BASES, 3/4.9.9 WATER LEVEL – REACTOR VESSEL, "The restrictions on minimum water level ensure that sufficient water depth is available to remove 99.5% of the assumed 16% I-131 and 10% other halogens gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis."</td></tr>
<tr><td>Change #20</td><td colspan="2">Basis / Safety Assessment:
A new dose analysis (see Attachment 10, DC00040-102, "Fuel Handling Accidents – AST") for the Fuel Handling Accident is enclosed and proposed for use in Chapter 15 of the FSAR. The analysis is performed utilizing the guidance of Regulatory Guide 1.183 to meet the 10 CFR 50.67 regulatory requirements. The analysis utilizes iodine removal factors and gap activities that differ from the CLB assumptions currently shown in BASES Section 3/4.9.9. This change updates the BASES section to reflect the revised analysis assumptions.</td></tr>
<tr><td>Change #21</td><td>Current Technical Specification:
ADMINISTRATIVE CONTROLS, 6.8.4.1 Ventilation Filter Testing Program (VFTP)</td><td>Proposed Change:
The Spent Fuel Ventilation System is removed from the scope of the VFTP.</td></tr>
<tr><td>Change #21</td><td colspan="2">Basis / Safety Assessment:
Change #8 deletes all Limiting Conditions for Operation and Surveillance Requirements for the Spent Fuel Pool Ventilation System since they are no longer needed for mitigation of the Fuel Handling Accident. This change deletes the Spent Fuel Pool Ventilation System from the VFTP defined by ADMININSTRATIVE CONTROL 6.8.4.l. Removal of current inplace and laboratory filter tests requirements for the Spent Fuel Pool Ventilation Sub-Systems is, likewise, justified since the system is no longer needed for mitigation of the Fuel Handling Accident.</td></tr>
</table>

Attachment 6

Proposed Technical Specification Changes (Mark-ups)

Table 6-1: List of Proposed Technical Specification Changes (Marked ups)

Table 5-1 Change #	Sections	Title
1	TOC	Index
2, 3 & 4	Table 3.3-6	RADIATION MONITORING INSTRUMENTATION
5, 6 & 7	Table 4.3-3	RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS
8	3/4.7.11	SPENT FUEL POOL VENTILATION SYSTEM
9	3/4.9.4	REFUELING OPERATIONS REACTOR BUILDING PENETRATIONS
10	3/4.9.8	REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM
21	6.8.4.1	ADMINISTRATIVE CONTROLS –Ventilation Filter Testing Program (VFTP)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

REPLACE

DELETED

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DELETED

REPLACE

INDEX

BASES

DELETED REPLACE

INDEX

BASES

REPLACE

DELETED

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area (RM-G8)	1	*	≤ 15 mR/hr	10^{-1} - 10^{4} mR/hr	25
b. ~~Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)~~	~~1~~	~~6~~	~~≤ 1 R/hr~~	~~1 - 10^{5} mR/hr~~	~~28~~ delete
2. PROCESS MONITORS					
a. ~~Spent Fuel Pool Exhaust - Ventilation System (RM-A6)~~					
i. ~~Gaseous Activity~~	~~1~~	~~**~~	~~≤ 1 x 10^{-5} μCi/cc (Kr-85)~~	~~10 - 10^{6} cpm~~	~~27~~ delete
ii. ~~Particulate Activity~~	~~1~~	~~**~~	~~N/A~~	~~10 - 10^{6} cpm~~	~~27~~
b. Containment					
i. ~~Gaseous Activity - Purge & Exhaust Isolation (RM-A4)~~	~~1~~	~~6~~	~~≤ 2 x background***~~	~~10 - 10^{6} cpm~~	~~28~~ delete
ii. Particulate and Gaseous Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{6} cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	≤ 2 x background	10 - 10^{6} cpm	29

Replace

DELETED

Replace

* With fuel in the storage pool or building

~~** With irradiated fuel in the storage pool~~

~~*** Alarm/trip setpoint will be per the Operational Dose Calculation Manual when purge exhaust operations are in progress.~~ delete

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

DELETED

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

Replace

ACTION 27 - ~~With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.11.~~

ACTION 28 - ~~With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.8.~~

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the emergency mode of operation.

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE Status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	S	R	M	*
b. ~~Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)~~ Replace, DELETED	~~S~~	~~R~~	~~M~~	~~6~~ delete
2. PROCESS MONITORS				
a. ~~Spent Fuel Pool Exhaust Area - Ventilation System (RM-A6)~~				
~~i. Gaseous Activity~~	~~S~~	~~R~~	~~M~~	~~**~~ delete
~~ii. Particulate Activity~~	~~S~~	~~R~~	~~M~~	~~**~~
b. Containment				
i. ~~Gaseous Activity - Purge & Exhaust Isolation (RM-A4)~~	~~S~~	~~R~~	~~M~~	~~6~~ delete
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	S	R	M	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	S	R	M	All MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	S	R	M	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	S	R	M	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	S	R	M	1, 2, 3 & 4

* With fuel in the storage pool or building

~~** With irradiated fuel in the storage pool~~ delete

PLANT SYSTEMS

3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11 Two independent spent fuel pool ventilation sub-systems shall be OPERABLE with at least one sub-system in operation.

APPLICABILITY: Whenever irradiated fuel is being moved in the spent fuel pool and during crane operation with loads over the pool.

ACTION:

a. With one spent fuel pool ventilation sub-system inoperable, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the OPERABLE spent fuel pool ventilation sub-system is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no spent fuel pool ventilation sub-system OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11 The above required spent fuel pool ventilation sub-systems shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that each sub-system operates for at least 15 minutes.

b. By performing required filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

c. At least once per 18 months by:

1. Verifying that on a loss of offsite power test signal, the system automatically starts.

2. Verifying that the system maintains the spent fuel pool area at a negative pressure greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

THIS SECTION DELETED

REFUELING OPERATIONS

3/4.9.4 REACTOR BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the reactor building atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed by an OPERABLE automatic Reactor Building Purge and Exhaust isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the reactor building.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the reactor building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required reactor building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic Reactor Building Purge and Exhaust isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the reactor building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the Reactor Building Purge and Exhaust isolation valves per the applicable portions of Specification 4.6.4.2.

THIS SECTION DELETED REPLACE

REFUELING OPERATIONS

3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.8 The Reactor Building Purge Supply and Exhaust Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the Reactor Building Purge Supply and Exhaust Isolation System inoperable, close each of the Purge and Exhaust penetrations providing direct access from the reactor building atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 The Reactor Building Purge Supply and Exhaust Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that Reactor Building Purge Supply and Exhaust isolation occurs on manual initiation, on a high radiation test signal from each of the containment radiation monitoring instrumentation channels, and by verifying that isolation occurs on the 36-inch lines of the Purge Supply and Exhaust Isolation System on a high radiation test signal from the reactor building manipulator crane area channels.

REPLACE

THIS SECTION DELETED

location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

a) Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

b) Inspect 100% of the tubes at sequential periods of 144, 108, 72 and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

c) If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

5. Provisions for monitoring operational primary-to-secondary leakage.

I. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with Regulatory Guide 1.52, Revision 2 and ASME N510-1989.

1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate	
Control Room Emergency Filtration System	21,270 SCFM	
~~Spent Fuel Pool Ventilation System~~	~~30,000 ACFM~~	DELETE
Reactor Building Cooling Units	60,270 ACFM	

2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate	
Control Room Emergency Filtration System	21,270 SCFM	
~~Spent Fuel Pool Ventilation System~~	~~30,000 ACFM~~	DELETE

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below.

ESF Ventilation System	Penetration	RH	Face Velocity (fps)	
Control Room	<2.5%	70%	0.667	
~~Spent Fuel Pool~~	~~<2.5%~~	~~95%~~	~~0.667~~	DELETE

4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Delta P	Flowrate	
Control Room	<6 in. W.G.	21,270 SCFM	
~~Spent Fuel Pool~~	~~<6 in. W.G.~~	~~30,000 ACFM~~	DELETE
Reactor Building Cooling Units	<3 in. W.G.	60,270 ACFM	

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

m. 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 Filtration System (CREFS), 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 whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

Attachment 7

For Information - Proposed Technical Specification Bases Changes Mark-ups

Table 7-1: List of Proposed Technical Specification Bases Changes (Marked ups)

Table 5-1 Change #	Sections	Title
11 & 12	3/4.4.5	STEAM GENERATOR TUBE INTEGRITY
13	3/4.4.6.2	OPERATIONAL LEAKAGE
14	3/4.6.1.1	CONTAINMENT INTEGRITY
15	3/4.7.1.4	ACTIVITY
16	3/4.7.10	WATER LEVEL SPENT FUEL POOL
17	3/4.7.11	SPENT FUEL POOL VENTILATION SYSTEM
18	3/4.9.4	REACTOR BUILDING PENETRATIONS
19	3/4.9.8	REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM
20	3/4.9.9	WATER LEVEL – REACTOR VESSEL

REACTOR COOLANT SYSTEM

BASES

STEAM GENERATOR TUBE INTEGRITY (Continued)

Applicable Safety Analyses

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding a SGTR is the basis for this Specification. The accident analysis for a SGTR event accounts for a bounding primary-to-secondary leakage rate equal to 1 gpm and the leakage rate associated with a double-ended rupture of a single tube. Contaminated fluid in a ruptured steam generator is only briefly released to the atmosphere as steam via the main steam safety valves. To maximize its contribution to the dose releases, the entire 1 gpm primary-to-secondary leakage is assumed to occur in the intact steam generators where it can be released during the subsequent cooldown of the plant.

The analyses for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 1 gpm, or is assumed to increase to 1 gpm as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be greater than or equal to the limits in LCO 3.4.8, "Reactor Coolant System, Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Reference 2), 10 CFR 100 (Reference 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

REPLACE 50.67

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity. Refer to Action a. below.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.k and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

REACTOR COOLANT SYSTEM

BASES

STEAM GENERATOR TUBE INTEGRITY (Continued)

Surveillance Requirements (Continued)

4.4.5.2 During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.8.4.k are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The frequency of "Prior to entering MODE 4 following a SG inspection" ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

References

1. NEI 97-06, "Steam Generator Program Guidelines"
2. 10 CFR 50, Appendix A, GDC 19, "Control Room"
3. 10 CFR ~~100, "Reactor Site Criteria"~~ REPLACE 50.67, "Accident Source Term"
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976
6. EPRI TR-107569, "Pressurized Water Reactor Steam Generator Examination Guidelines"

REACTOR COOLANT SYSTEM

BASES

OPERATIONAL LEAKAGE (Continued)

Applicable Safety Analyses

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary-to-secondary leakage from all steam generators is 1 gpm or increases to 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR analysis for SGTR accounts for a bounding primary-to-secondary leakage rate equal to 1 gpm and the leakage rate associated with a double-ended rupture of a single tube. Leakage through the ruptured tube is the dominate contributor to dose releases. Since contaminated fluid in the ruptured steam generator is only briefly released to the atmosphere as steam via the main steam safety valves, the entire 1 gpm primary-to-secondary leakage is assumed to occur in the intact steam generators where it can be released during the subsequent cooldown of the plant. Overall, this pathway is a small contributor to dose releases.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes the entire 1 gpm primary-to-secondary leakage is through the effected steam generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits). REPLACE 50.67

The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the Reactor Coolant Pressure Boundary. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR ~~100~~ during accident conditions.

REPLACE 50.67

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates (including those used in demonstrating a 30 day water seal) ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the Containment Leakage Rate Testing Program.

3/4.6.1.3 REACTOR BUILDING AIR LOCKS

The limitations on closure for the reactor building air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

PLANT SYSTEMS

BASES

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

The OPERABILITY of the emergency feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each emergency feedwater pump is capable of delivering a total feedwater flow of 380 gpm at a pressure of 1211 psig to the entrance of two out of three steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F at which point the Residual Heat Removal System may be placed into operation.

Also, each Emergency Feedwater (EFW) pump is capable of supplying 400 gpm to all 3 steam generators while the steam generators are pressurized to 1211 psig. This capacity is sufficient to ensure that the pressurizer does not overfill during a loss of normal feedwater event. The total head criteria of 3800 feet for the motor driven EFW pumps and 3140 feet for the turbine driven EFW pump includes margin that allows for a maximum EFW flow control valve leakage of 5 gpm for any one of 6 EFW flow control valves.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 11 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

REPLACE 50.67

PLANT SYSTEMS

BASES

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of 2°F.

3/4.7.10 WATER LEVEL - SPENT FUEL POOL

INSERT 16% I-131 and 10% other halogens

99.5% REPLACE

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

DELETE

~~3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM~~

~~The limitations on the spent fuel pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analysis.~~

THIS SECTION DELETED REPLACE

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. Valves in the reactor makeup system are required to be closed to minimize the possibility of a boron dilution accident.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum time of 72 hours for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. The minimum decay time of 72 hours is consistent with the assumptions used in the accident analysis.

The tabulated hold times associated with Component Cooling Water (CCW) temperature ensure that the spent fuel heat load is reduced sufficiently to allow the spent fuel pool cooling system to maintain the bulk pool temperature below 170°F. These hold times ensure that adequate cooling is provided to the Spent Fuel Pool under the highest possible heat load conditions. The hold times are based on the performance of the cooling system, which is dependent upon CCW temperature and recognizes that the spent fuel pool cooling system is capable of increased flow rates up to 2400 gpm during single loop operation. This higher flow rate may be required when only a single cooling loop is operable during a refueling outage.

The CCW temperature limits defined in Figure 3.9-1 are adjusted for uncertainty in the implementing procedure.

3/4.9.4 ~~REACTOR BUILDING PENETRATIONS~~

~~The requirements on reactor building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of reactor building pressurization potential while in the REFUELING MODE.~~

REPLACE THIS SECTION DELETED

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

REFUELING OPERATIONS

BASES

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod and fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and 2) sufficient coolant circulation is maintained thru the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

THIS SECTION DELETED REPLACE

~~3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM~~

~~The OPERABILITY of this system ensures that the reactor building vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the reactor building. The OPERABILITY of this system is required to restrict the release of radioactive material from the reactor building atmosphere to the environment.~~

3/4.9.9 WATER LEVEL - REACTOR VESSEL

99.5 REPLACE The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

REPLACE 16% I-131 and 10% other halogens

Attachment 8

Procedure Changes to be Completed Before AST Implementation

Procedure Changes to be Completed Before AST Implementation

Change	Basis
Revise EOP-2.2, "Transfer to Cold Leg Recirculation", to further isolate the RWST and NAOH tanks following the transition the cold leg recirculation.	Leakage through the RWST and NAOH is neglected in the AST calculations for LOCA based on the assumption that the EOPs will require closure of the 20" RWST outlet valve (6700) and closure of the 3" NAOH outlet valve (3012) following the transition to cold leg recirculation. This results in 3 valve isolation and a minimum of 2 valves isolation in the long term with a single failure.
Revise EOPs and other plants procedures as appropriate to require 4-hours of Spray Operation following a LOCA.	The LOCA AST analysis credits the iodine removal capabilities of the RB Spray for a period of 4-hours.
Revise EOPs and other plant procedures as appropriate to ensure the CR ventilation is run in the emergency mode when required for non-LOCA events that do not result in a SI.	Placing the CR ventilation in the emergency mode has been credited within 2-hours for non-LOCA events (LRA & CREA) that may not result in a SI. Although this is expected to occur automatically with high radiation on RMA-1, this feature has not been credited due to lack of redundancy. Consequently, as a backup for RMA-1,the operator will be asked to assess the need for protection actions based on existing radiation indications and atmospheric releases.

Attachment 9

No Significant Hazards Consideration Determination & Environmental Consideration for the Proposed Changes

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Description of Amendment Request

South Carolina Gas and Electric Co. (SCE&G) is proposing to amend the operating license for Virgil C. Summer Nuclear Station (VCSNS), by revising the Technical Specifications (TS) and incorporating an alternative source term (AST) methodology into the facility's licensing basis. The proposed license amendment involves a full implementation of an AST methodology by revising the current accident source term and replacing it with an AST, as prescribed in 10 CFR 50.67.

AST analyses were performed using the guidance provided by Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms." The six PWR limiting design basis accidents (DBAs) identified in RG 1.183 considered were the loss of coolant accident (LOCA), the main steam line break accident (MSLB), the refueling accident (FHA), the steam generator tube rupture (SGTR), reactor coolant pump locked rotor (RCP LRA) and the control rod ejection accident (CREA). As a result of the application of a revised accident source term, changes are proposed to the TS to implement Improved Standard Technical Specifications Change Traveler (Reference 12.1 - TSTF-51, Revision 2) which permits removal of the Technical Specification requirements for engineered safety features (ESF) features to be OPERABLE after sufficient radioactive decay has occurred to ensure off-site doses remain below the SRP limits. Other Technical Specification revisions reflect the update of the accident source term and associated design basis accidents utilizing the guidance provided in USNRC Regulatory Guide 1.183 and the associated control room and offsite dose requirements of 10 CFR 50.67.

The AST analyses are based on new control room habitability (CRHE) atmospheric dispersion coefficients (χ/Qs) based on site specific meteorological data in accordance with Regulatory Guides 1.194.

Basis for No Significant Hazards Determination:

Pursuant to 10 CFR 50.92, VCSNS has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration, since the proposed change satisfies the criteria in l0 CFR 50.92(c). These criteria require that the operation of the facility in accordance with the proposed amendment will not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

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

Response: No.

Adoptions of the AST and pursuant TS changes and the changes to the atmospheric dispersion factors have no impact to the initiation of DBAs. Once the occurrence of an accident has been postulated, the new accident source term and atmospheric dispersion factors are an input to analyses that evaluate the radiological consequences. Some of the proposed changes do affect the design or manner in which the facility is operated following an accident; however, the proposed changes do not involve a revision to the design or manner in which the facility is operated that could increase the probability of an accident previously evaluated in Chapter 15 of the FSAR.

Therefore, the proposed change does not involve an increase in the probability of an accident previously evaluated.

The structures, systems and components affected by the proposed changes act as mitigators to the consequences of accidents. Based on the AST analyses, the proposed changes do revise certain performance requirements; however, the proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of an accident previously discussed in Chapter 15 of the FSAR.

Plant-specific radiological analyses have been performed using the AST methodology and new atmospheric dispersion factors. Based on the results of these analyses, it has been demonstrated that the CRHE dose consequences of the limiting events considered in the analyses meet the regulatory guidance provided for use with the AST, and the offsite doses are within acceptable limits. This guidance is presented in 10 CFR 50.67, RG 1.183, and Standard Review Plan Section (SRP) 15.0.1.

Therefore, the proposed amendment does not result in a significant increase in the consequences of any previously evaluated accident.

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

Response: No.

Implementation of AST and the associated proposed TS changes and new atmospheric dispersion factors do not alter or involve any design basis accident initiators. With the exception of the fuel handling accident, these changes do not affect the design function or mode of operations of structures, systems and components in the facility prior to a postulated accident. Since structures, systems and components are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed change. The alternative source term change itself does not have the capability to initiate accidents.

For the fuel handling accident, the Improved Standard Technical Specifications Change Traveler (TSTF-51, Revision 2) permits removal of the Technical Specification requirements for ESF features to be OPERABLE after sufficient radioactive decay has occurred to ensure off-site doses remain below the SRP limits. As noted in this submittal no credit is taken for the accident mitigation of the ESF features associated with the fuel handling accidents to meet these limits. Since these are not associated with accident initiators the proposed license amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The results of the AST analyses are subject to the acceptance criteria in 10 CFR 50.67. The analyzed events have been carefully selected, and the analyses supporting these changes have been performed using approved methodologies to ensure that analyzed events are bounding and safety margin has not been reduced. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, RG 1.183, and SRP 15.0.1. Thus, by meeting the applicable regulatory limits for AST, there is no significant reduction in a margin of safety.

New Control Room atmospheric dispersion factors (χ/Qs) based on site specific meteorological data, calculated in accordance with the guidance of RG 1.194, utilizes more recent data and improved calculational methodologies.

For the fuel handling accident, the Improved Standard Technical Specifications Change Traveler (TSTF-51, Revision 2) permits removal of the Technical Specification requirements for ESF features to be OPERABLE after sufficient radioactive decay has occurred to ensure off-site doses remain below the SRP limits. Following sufficient decay, the primary success path for mitigating the fuel handling accident no longer includes the functioning of the active containment or fuel handling building systems. With the proposed changes, the OPERABILITY requirements of the Technical Specifications will reflect that water level

(23') and decay time (72 hours after shutdown) are the primary success path for mitigating a fuel handling accident.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to not result in a significant reduction in a margin of safety.

Conclusion

On the basis of the above, VCSNS has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(C), in that it: (1) does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL CONSIDERATION

VCSNS has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21.

VCSNS has determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). 10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. VCSNS has evaluated the proposed change and has determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the amendment. The basis for this determination, using the above criteria, follows:

Basis

As demonstrated in the No Significant Hazards Consideration Evaluation, the proposed amendment does not involve a significant hazards consideration.

There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure. Furthermore, the proposed change does not involve any unreviewed safety questions concerning the physical alteration of the plant or change in methods governing normal plant operation except for the fuel handling accident. For the fuel handling accident, the Improved Standard Technical Specifications Change Traveler (TSTF-51, Revision 2) permits removal of the Technical Specification requirements for ESF features to be OPERABLE after sufficient radioactive decay has occurred to ensure off-site doses remain below the SRP limits. Following sufficient decay, the primary success path for mitigating the fuel handling accident no longer includes the functioning of the active containment or fuel handling building systems. With the proposed changes, the OPERABILITY requirements of the Technical Specifications will reflect that water level (23') and decay time (72 hours after shutdown) are the primary success path for mitigating a fuel handling accident.

Conclusion

The alternative source term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated, the alternative source term is an input to evaluate the consequences of accidents. The implementation of the alternative source term has been evaluated in AST analyses of the limiting design basis accidents at VCSNS (loss of coolant accident, the main steam line break accident, the refueling accident, the steam generator tube rupture, reactor coolant pump locked rotor and the control rod ejection accident). Based upon the results of these analyses it has been demonstrated that, with the requested changes, the dose consequences are within NRC regulatory limits for alternative source term (i.e., 10 CFR 50.67 and 10 CFR 50, Appendix A, General Design Criterion 19).

On the basis of the above, VCSNS has determined that operation of the facility in accordance with the proposed change does not involve an environmental consideration as defined in 10 CFR 51.22(c)(9), in that it does not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

Attachment 10

CD of Non-Proprietary Versions of Supporting Calculations and Meteorological Data Used to Determine New Control Room χ/Qs

The following non-proprietary Supporting Calculations are included in the "Supporting Calculation" directory of the Attachment 10 CD.

1. VCSNS Calculation DC00040-079, "Atmospheric Dispersion Coefficients for Control Room", Revision 1.

2. VCSNS Calculation DC00040-097, "Loss of Coolant Accident - AST", Revision 0.

3. VCSNS Calculation DC00040-098, "Steam Generator Tube Rupture - AST", Revision 0.

4. VCSNS Calculation DC00040-099, "Main Steam Line Break - AST", Revision 0.

5. VCSNS Calculation DC00040-100, "Reactor Coolant Pump Locked Rotor - AST", Revision 0.

6. VCSNS Calculation DC00040-101, "Rod Ejection - AST", Revision 0.

7. VCSNS Calculation DC00040-102, "Fuel Handling Accidents - AST", Revision 0.

The following non-proprietary information is included in the "Meteorological Data" directory of the Attachment 10 CD.

1. VCSNS Calculation DC00040-080, "Post DBA CR Dose - Met Data Input to ARCON96", Revision 2.

2. ARCON96 Input File (VCS_R2.MET).

3. 6-EXCEL Files Containing Raw Met Tower Data from 2002 to 2006.

Attachment 11

Proposed Technical Specification Changes (Re-Typed)

BASES

BASES

SECTION PAGE

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area (RM-G8)	1	*	≤ 15 mR/hr	10^{-1} - 10^{4} mR/hr	25
b. Deleted					
2. PROCESS MONITORS					
a. Deleted					
b. Containment					
i. Deleted					
ii. Particulate and Gaseous Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{6} cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	≤ 2 x background	10 - 10^{6} cpm	29

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* With fuel in the storage pool or building

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 27 - Deleted

ACTION 28 - Deleted

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the emergency mode of operation.

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	S	R	M	*
b. Deleted				
2. PROCESS MONITORS				
a. Deleted				
b. Containment				
i. Deleted				
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	S	R	M	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	S	R	M	ALL MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	S	R	M	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	S	R	M	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	S	R	M	1, 2, 3 & 4

* With fuel in the storage pool or building

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location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

a) Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

b) Inspect 100% of the tubes at sequential periods of 144, 108, 72 and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

c) If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

5. Provisions for monitoring operational primary-to-secondary leakage.

l. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989.

1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
Control Room Emergency Filtration System	21,270 SCFM
Reactor Building Cooling Units	60,270 ACFM

2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
Control Room Emergency Filtration System	21,270 SCFM

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below.

ESF Ventilation System	Penetration	RH	Face Velocity (fps)
Control Room	<2.5%	70%	0.667

4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Delta P	Flowrate
Control Room	<6 in. W.G.	21,270 SCFM
Reactor Building Cooling Units	<3 in. W.G.	60,270 ACFM

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

m. <u>Control Room Envelope Habitability Program</u>

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 Filtration System (CREFS), 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 whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

Attachment 12

Regulatory Commitments

There are no commitments contained in this submittal.