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Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Alternative Source Term, TSTF-51, TSTF-471, and TSTF-490

Ladies and Gentlemen:

Pursuant to the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations*, Southern Nuclear Operating Company (SNC) hereby submits license amendment requests (LARs) to renewed facility operating licenses NPF-68 and NPF-81 to revise the Technical Specifications (TS) for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2, respectively.

SNC requests Nuclear Regulatory Commission (NRC) review and approval of the proposed revisions to the licensing basis of VEGP that support a selected scope application of an Alternative Source Term (AST) methodology. The proposed changes, which are supported by the AST Design Basis Accident radiological consequences analysis, are included in this license amendment request (LAR). In addition, the proposed amendment incorporates Technical Specification Task Force (TSTF) Travelers TSTF-51-A, "Revise containment requirements during handling irradiated fuel and core alterations," Revision 2; TSTF-471-A, "Eliminate use of term CORE ALTERATIONS in ACTIONS and Notes," Revision 1; and TSTF-490-A, "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec," Revision 0.

The Enclosure to this letter contains SNC's evaluation of the proposed changes. Attachments to this enclosure provide supporting documentation, as required. Attachment 1 of the Enclosure provides the existing TS pages marked-up to show the proposed changes. Note that existing TS pages 3.4.16-3 and 3.4.16-4 are proposed to be deleted in their entirety.

SNC requests approval of this LAR one year from acceptance. The proposed changes would be implemented within 120 days of issuance of the amendment.

This document contains NRC regulatory commitments as stated in Attachment 12 of the Enclosure.

In accordance with 10 CFR 50.91(b)(1), SNC is notifying the State of Georgia of this LAR by transmitting a copy of this letter and enclosure to the designated State Official.

If you have any questions regarding this submittal, please contact Ryan Joyce at 205.992.6468.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 30th day of June 2022.

Respectfully submitted,

Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Company

CAG/kgl/cg

Enclosure: Basis for Proposed Change

cc: Regional Administrator, Region II
NRR Project Manager – Vogtle 1 and 2
Senior Resident Inspector – Vogtle 1 and 2
Director, Environmental Protection Division – State of Georgia
RType: CVC7000

Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Alternative Source Term,
TSTF-51, TSTF-471, and TSTF-490

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1.0 Summary Description

This evaluation supports a request to revise Operating Licenses NPF-68 and NFP-81 for Vogtle Electric Generating Plant (VEGP), Units 1 and 2, respectively. Southern Nuclear Operating Company (SNC) requests Nuclear Regulatory Commission (NRC) review and approval of proposed revisions to the licensing basis of VEGP that support a selective scope application of an Alternative Source Term (AST) methodology. The proposed amendment also incorporates Revision 2 of Technical Specification Task Force (TSTF) Traveler, TSTF-51, “Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations,” Revision 1 of TSTF-471, “Eliminate use of term CORE ALTERATIONS in ACTIONS and Notes,” and Revision 0 of TSTF-490, “Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec,” into the VEGP Technical Specifications.

2.0 Detailed Description

2.1 Background

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, "Accident Source Term," which provided a mechanism for licensed power reactors to voluntarily replace the traditional accident source term used in their Design Basis Accident (DBA) analyses with an AST. Regulatory guidance for the implementation of the AST is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 1). 10 CFR 50.67 requires a licensee seeking to use an AST to submit a license amendment request (LAR) and requires that the application contain an evaluation of the consequences of DBAs.

This LAR addresses the applicable requirements and guidance in proposing to use an AST in evaluating the offsite and Control Room (CR) radiological consequences of the VEGP DBAs. This reanalysis involves several changes in selected analysis assumptions. As part of the implementation of the AST, the Total Effective Dose Equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11. This will also replace the whole body (and its equivalent to any part of the body) dose criteria of 10 CFR 50, Appendix A, General Design Criterion (GDC) 19.

2.2 TSTF-51

The proposed amendment would revise certain Technical Specifications (TSs) to remove the requirements for certain engineered safety features (ESF) systems to operate after sufficient radioactive decay of irradiated fuel has occurred following a plant shutdown. Following sufficient radioactive decay, these ESF systems are no longer required during a fuel handling accident (FHA) to ensure main control

room (MCR) personnel dose remains below the 10 CFR 50.67(b)(2)(iii) dose limit and of-site dose remains below the accident dose limit specified in the NRC standard review plan, which represents a small fraction of the 10 CFR 50.67 does limits.

Associated with this change is the deletion of OPERABILITY requirements during CORE ALTERATIONS for certain ESF mitigation features. This change will allow 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 an FHA. Following sufficient decay occurring, the primary success path for mitigating the FHA no longer includes the functioning of the active containment systems. Therefore, the OPERABILITY requirements of the TS are modified to reflect that water level and decay time are the primary success path for mitigating an FHA (which meets General Design Criterion 3).

To support this change in requirements during the handling of irradiated fuel, the OPERABILITY requirements during CORE ALTERATIONS for certain ESF mitigation features are deleted. The accidents postulated to occur during core alterations, in addition to fuel handling accidents, are: inadvertent criticality (due to 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. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the FHA, the proposed TS requirements omitting CORE ALTERATIONS is justified.

Also, the TS 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.

2.3 TSTF-471

Suspending CORE ALTERATIONS or exempting testing except during CORE ALTERATIONS has no effect on the initial conditions or mitigation of any Design Basis Accident (DBA) or transient, and these requirements apply an operational burden with no corresponding safety benefit. Therefore, the proposed amendment eliminates the use of the defined term CORE ALTERATIONS from only the following select Technical Specifications.

TS 3.9.1 Boron Concentration

TS 3.9.2 Unborated Water Source Isolation Valves

TS 3.9.4 Containment Penetrations

2.4 TSTF-490

The proposed amendment would replace the current limits on primary coolant gross specific activity with limits on primary coolant noble gas activity. The noble gas activity would be based on DOSE EQUIVALENT XE-133 and would take into account only the noble gas activity in the primary coolant.

The background for this proposed change is as stated in the model SE in the NRC Notice of Availability published on March 19, 2007 (72 FR 12838) (Reference 2), the NRC Notice for Comment published on November 20, 2006 (71 FR 67170 (Reference 3), and TSTF-490, Revision 0, “Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec.”

3.0 Technical Evaluation

Alternative Source Term

3.1 Meteorology and Atmospheric Dispersion

The AST application uses atmospheric dispersion (X/Q) values for the Exclusion Area Boundary (EAB), the Low Population Zone (LPZ), and the CR receptors. As described below, the EAB and LPZ X/Q values are consistent with the current licensing basis, as given in VEGP Final Safety Analysis Report (FSAR) Table 2.3.4-1 and Table 15A-2. New and revised X/Q values for the CR have been developed to address potential leakage from the Refueling Water Storage Tank (RWST) vent and releases from the secondary side for evaluation of non-LOCA radiological consequences. The resulting X/Q values at the CR intakes are calculated using the NRC-sponsored computer code ARCON96 (NUREG/CR-6331) consistent with the procedures in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," (Reference 4). Information used to develop the new X/Q values is included in Attachment 11.

3.1.1 Meteorological Data

The same meteorological data used to calculate the X/Q values applied in the current licensing basis radiological consequence analyses was determined to remain representative of the site and was used to calculate new CR X/Q values.

3.1.2 EAB and LPZ Atmospheric Dispersion Factors

RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," Section 5.3, "Meteorology Assumptions," states:

> Atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the CR that were approved by the staff during the initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide.

For the AST analyses, X/Q values for the EAB and the LPZ are consistent with the current licensing basis.

The X/Q values for the EAB and the LPZ used in the radiological consequence analyses are shown in Table 3.1.

Table 3.1 - EAB and LPZ X/Q values (sec/m^3)

Location	Time Period	X/Q Value
EAB	0 – 2 hours	1.8×10^{-4}
LPZ	0 – 2 hours	7.2×10^{-5}
LPZ	2 – 8 hours	3.3×10^{-5}
LPZ	8 – 24 hours	2.2×10^{-5}
LPZ	24 – 96 hours	9.2×10^{-6}
LPZ	96 – 720 hours	2.7×10^{-6}

3.1.3 Control Room Atmospheric Dispersion Factors

X/Q values for onsite release-receptor combinations were developed using the ARCON96 computer code and the guidance in RG 1.194. Various release-receptor combinations were considered for the onsite CR X/Q values. These different cases are considered to determine the limiting release-receptor combination for the events postulated in RG 1.183. Existing X/Q values included releases from the Containment, Containment Hatch Door, and Fuel Handling Building. New X/Q values were developed for the RWST release points for the Loss-of-Coolant Accident (LOCA) as well as the North and South Main Steam Valve Rooms for secondary side releases in non-LOCA events such as the Control Rod Ejection Accident (CREA), Locked Rotor Accident (LRA), Steam Generator Tube Rupture (SGTR), and Main Steam Line Break (MSLB).

Figure 3.1 provides a sketch of the general layout of Vogtle 1 & 2 that has been annotated to highlight the onsite release and receptor point locations. All releases are taken as ground level releases per RG 1.194 Position 3.2.1.

Table 3.2 provides information related to the relative elevations of the release-receptor combinations, the straight-line horizontal distance between the release point and the receptor location, and the direction (azimuth) from the receptor location to the release points. Angles are calculated based on trigonometric layout of release and receptor points in relation to the North-South and East-West axes.

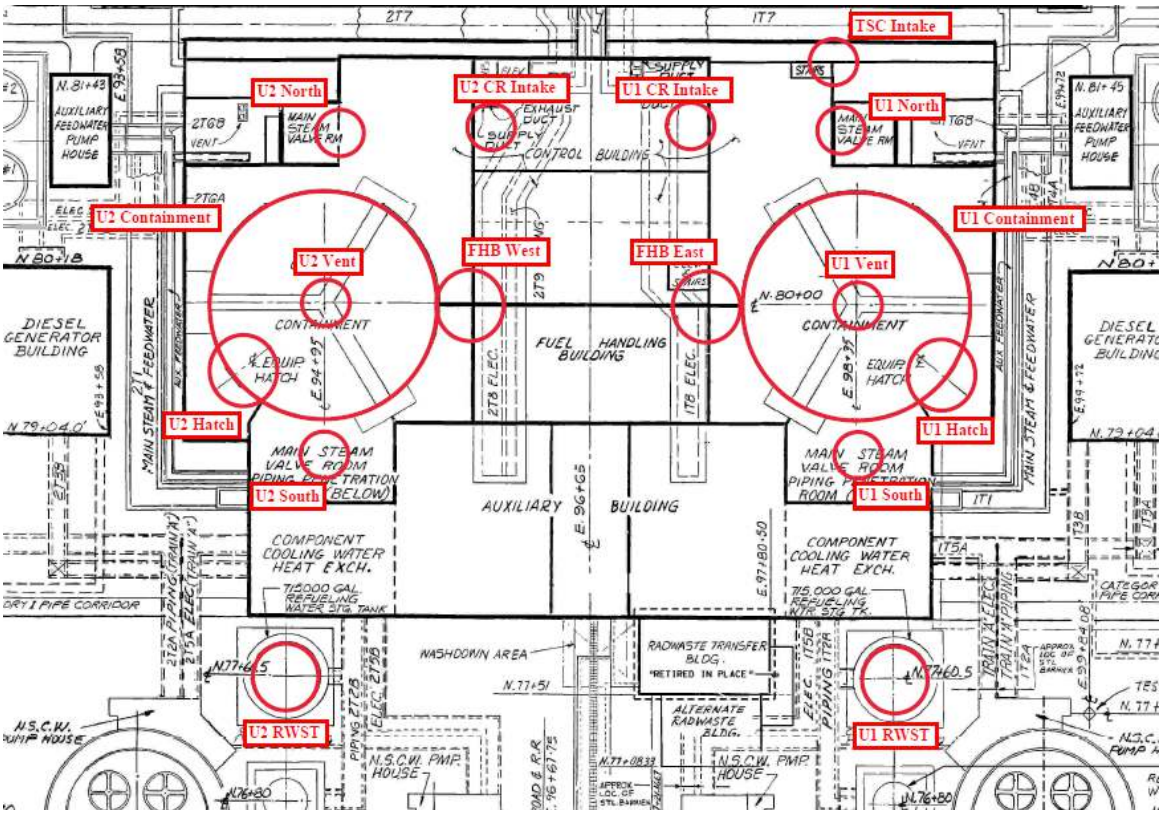

Figure 3.1 - Air Intake Locations and Release Points

Table 3.2 - Distance and Geometry of Release and Receptor Locations

Receptor: Unit 1 CR Air Intake						
Release Point	**Horizontal Distance (ft)**	**Horizontal Distance (m)**	**Modeled Release Elevation (ft)**	**Modeled Release Elevation above Grade**		**Direction to Source (degree)**
Unit 1				(ft)	(m)	
Containment	70.9	21.6	283.0*	63.0	19.2	136
RWST	365.0	111.2	283.0*	63.0	19.2	160
North Release	91.9	28.0	283.0*	63.0	19.2	90
South Release	207.6	63.3	283.0*	63.0	19.2	151
Plant Vent	144.6	44.1	420.1	200.1	61.0	136
Hatch Door	275.1	83.8	228.5	8.5	2.6	135
FHB	106.0	32.3	288.2	68.2	20.8	180
Unit 2						
Containment	188.1	57.3	283.0*	63.0	19.2	246
RWST	432.4	131.8	283.0*	63.0	19.2	217
North Release	232.0	70.7	283.0*	63.0	19.2	270
South Release	301.2	91.8	283.0*	63.0	19.2	233
Plant Vent	261.8	79.8	420.1	200.1	61.0	246
Hatch Door	363.0	110.6	228.5	8.5	2.6	243
Receptor: Unit 2 CR Air Intake						
Release Point	**Horizontal Distance (ft)**	**Horizontal Distance (m)**	**Modeled Release Elevation (ft)**	**Modeled Release Elevation above Grade**		**Direction to Source (degree)**
Unit 1				(ft)	(m)	
Containment	188.1	57.3	283.0*	63.0	19.2	114
RWST	432.4	131.8	283.0*	63.0	19.2	143
North Release	232.0	70.7	283.0*	63.0	19.2	90
South Release	301.2	91.8	283.0*	63.0	19.2	127
Plant Vent	261.8	79.8	420.1	200.1	61.0	114
Hatch Door	363.0	110.6	228.5	8.5	2.6	118
Unit 2						
Containment	70.9	21.6	283.0*	63.0	19.2	224
RWST	365.0	111.2	283.0*	63.0	19.2	200
North Release	91.9	28.0	283.0*	63.0	19.2	270
South Release	207.6	63.3	283.0*	63.0	19.2	209
Plant Vent	144.6	44.1	420.1	200.1	61.0	224
Hatch Door	275.1	83.8	228.5	8.5	2.6	225
FHB	106.0	32.3	288.2	68.2	20.8	180

* Release heights are conservatively made the same as the Control Room intakes.

Table 3.3 provides the ARCON96 modeling outputs for releases originating at the Containments, reactor vents, hatch doors, RWSTs, FHB, and North and South Main Steam Valve Rooms. The individual AST analyses use X/Q values that bound both units and correspond to the limiting release points applicable to the event. Refer to the individual accident Attachments for the CR X/Q values used in each analysis.

Table 3.3 - X/Q (s/m^3) Values at the Control Room Air Intakes

Release Point	Receptor	0 – 2 hours	2 – 8 hours	8 – 24 hours	1 – 4 days	4 – 30 days
U1 Reactor	U1 CR	2.20E-03	1.31E-03	5.26E-04	4.64E-04	3.32E-04
U1 Hatch Door	U1 CR	6.82E-04	4.85E-04	1.75E-04	1.43E-04	1.03E-04
U1 RWST	U1 CR	5.23E-04	3.81E-04	1.44E-04	1.15E-04	8.48E-05
U1 North	U1 CR	7.64E-03	6.17E-03	2.57E-03	1.74E-03	1.33E-03
U1 South	U1 CR	1.57E-03	1.12E-03	4.10E-04	3.32E-04	2.41E-04
U1 Vent	U1 CR	1.70E-03	1.23E-03	4.30E-04	3.50E-04	2.43E-04
U2 Reactor	U1 CR	8.46E-04	6.63E-04	2.96E-04	2.36E-04	1.71E-04
U2 Hatch Door	U1 CR	4.42E-04	3.64E-04	1.61E-04	1.28E-04	9.07E-05
U2 RWST	U1 CR	4.02E-04	3.21E-04	1.30E-04	1.08E-04	7.88E-05
U2 North	U1 CR	1.31E-03	1.05E-03	4.70E-04	3.22E-04	2.63E-04
U2 South	U1 CR	8.31E-04	6.62E-04	2.76E-04	2.31E-04	1.57E-04
U2 Vent	U1 CR	7.71E-04	5.59E-04	2.35E-04	1.96E-04	1.41E-04
FHB (east)	U1 CR	6.01E-03	4.44E-03	1.71E-03	1.40E-03	1.07E-03
U1 Reactor	U2 CR	8.01E-04	5.52E-04	2.34E-04	1.78E-04	1.43E-04
U1 Hatch Door	U2 CR	4.30E-04	3.06E-04	1.19E-04	9.11E-05	6.85E-05
U1 RWST	U2 CR	3.74E-04	2.65E-04	9.49E-05	7.91E-05	5.71E-05
U1 North	U2 CR	1.31E-03	1.06E-03	4.43E-04	3.02E-04	2.32E-04
U1 South	U2 CR	7.83E-04	5.30E-04	1.98E-04	1.52E-04	1.12E-04
U1 Vent	U2 CR	7.91E-04	5.59E-04	2.16E-04	1.60E-04	1.17E-04
U2 Reactor	U2 CR	2.22E-03	1.55E-03	6.57E-04	5.80E-04	4.47E-04
U2 Hatch Door	U2 CR	7.16E-04	5.83E-04	2.45E-04	2.05E-04	1.42E-04
U2 RWST	U2 CR	5.42E-04	4.27E-04	1.69E-04	1.37E-04	1.01E-04
U2 North	U2 CR	7.58E-03	6.10E-03	2.72E-03	1.86E-03	1.52E-03
U2 South	U2 CR	1.67E-03	1.32E-03	5.18E-04	4.21E-04	3.15E-04
U2 Vent	U2 CR	1.67E-03	1.20E-03	5.16E-04	4.39E-04	3.25E-04
FHB (west)	U2 CR	6.01E-03	4.44E-03	1.71E-03	1.40E-03	1.07E-03

Notes: U1 – Unit 1; U2 – Unit 2; CR – Control Room; FHB – Fuel Handling Building; RWST – Refueling Water Storage Tank

3.2 Radionuclide Inventory

3.2.1 Fission Product Inventory

The nominal inventory of fission products in the reactor core was calculated using ORIGEN-ARP based on the full power operation of the core plus uncertainty. The nominal inventory was based on an equilibrium cycle modeled with lead rod burnup of 62 GWD/MTU and variable enrichment regions. Parametric studies were performed varying enrichment (up to 5 weight percent), lead rod burnup, and beginning and end of cycle burnup. The fission product inventory used in the AST analyses applies a fuel design margin to the nominal inventory that ensures the AST dose consequences bound the dose consequences of the parametric study maxima.

3.2.2 Equilibrium RCS and Secondary Source Terms

For the AST dose analyses that consider it, the equilibrium RCS source term is assumed to consist of halogens (I, Br), noble gases (Kr, Xe), and alkali metals (Cs, Rb). The concentrations of I-131, I-132, I-133, I-134, and I-135 are assumed to be based on 1.0 µCi/g dose equivalent (DE) I-131, consistent with the Technical Specification 3.4.16 limit and revised

definition of DE I-131 from TSTF-490. The concentrations of Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 are be based on 280 µCi/g DE Xe-133, consistent with the proposed Technical Specification 3.4.16B limit and definition of DE Xe-133 from TSTF-490. Remaining nuclides are assumed to be present in the equilibrium RCS source term at concentrations corresponding to 1% of the fuel containing cladding defects.

For the AST dose analyses that consider it, the equilibrium secondary side source term is assumed to consist of halogens (I, Br) and alkali metals (Cs, Rb). The concentrations of I-131, I-132, I-133, I-134, and I-135 are based on 0.1 µCi/g dose equivalent (DE) I-131, consistent with Technical Specification LCO 3.7.16. The concentrations of other halogens and alkali metals in the equilibrium secondary side source term are assumed to be 10% of the RCS equilibrium concentrations, based on the ratio of the secondary to RCS DE I-131 concentrations.

The individual accident sections and Attachments describe the RCS and secondary side source term considerations applicable to that event (e.g., iodine spiking in the SGTR and MSLB).

3.3 Analytical Models

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

RADTRAD is used to determine the CR and offsite doses for the LOCA, FHA, CREA, LRA, SGTR, and MSLB using the source term, X/Q inputs, and pertinent scenario specific requirements in Regulatory Guide 1.183 as described in the individual event sections and Attachments. The code considers the radionuclide release timing and chemical form, hold-up and removal (e.g., spray, deposition) within compartments, filtration, transport to receptors, and dose calculation.

ARCON96 was used to determine the X/Qs at the CR intakes for selected release locations using plant meteorological data.

ORIGEN-ARP was used to calculate bounding plant-specific fission product inventories for use in the dose calculations that postulate fuel damage (LOCA, FHA, CREA, LRA).

3.4 Loss of Coolant Accident

The LOCA is a postulated rupture in the reactor coolant system that results in expulsion of the coolant to containment. Even though the emergency core cooling system (ECCS) is designed to maintain cooling of the fuel assemblies in this event, the dose consequence analysis is performed assuming a significant release of the radionuclides from the fuel assemblies.

3.4.1 Methodology Overview

The LOCA is modeled as a release of nuclides from the reactor core into the containment building. The Containment release paths modeled are:

1) the Containment Mini-Purge System,

2) Containment leakage,

3) ECCS leakage, and

4) RWST backleakage.

The radiological source term characteristics and release timing are based on the AST methodology in RG 1.183. Atmospheric dispersion factors from Section 3.1, above, are used in this analysis. Doses to the public at the EAB and the LPZ, and occupants in the CR are determined.

3.4.2 Radiological Dose Models

The RADTRAD (Version 3.10) code was used to calculate the immersion and inhalation dose contributions to both the onsite and offsite radiological dose consequences. Models were developed for both the containment leakage and ECCS leakage cases.

The analysis used assumptions and inputs that follow the guidance in RG 1.183. The key parameters and assumptions are listed in Attachment 5. The calculated dose results are given in Table 3.4. The calculated doses are within the RG 1.183 radiological dose acceptance criteria for a LOCA. These TEDE criteria are 25 rem at the EAB (worst 2 hour interval) and LPZ (cumulative for the 30 day accident duration), and 5 rem in the CR (cumulative for the 30 day accident duration).

Table 3.4 - Calculated LOCA Radiological Consequences

	TEDE (rem)		
	EAB*	LPZ	Control Room
Calculated results	8.4	9.6	4.4
Dose acceptance criteria	25	25	5

*Worst 2-hour dose

3.5 Fuel Handling Accident

The limiting case of a fuel handling accident (FHA) in the fuel building is analyzed.

Analysis of the FHA in the fuel building takes no credit for either filtration, or holdup in the fuel building. The dispersion factors for a release from the fuel building are much larger than those for a release from the containment

building (containment open configuration) equipment hatch and personnel air locks (PALs). In addition, a release through the containment PAL would result in a torturous path to the CR through the auxiliary building out to the atmosphere and finally into the CR from the control room emergency filtration system (CREFS) supply intake.

For the containment closed case where the purge system is operating, the purge flow rates would result in an 8 hour release duration as compared to the 2 hour duration required by RG 1.183 Revision 0 for a release from the fuel building. This release pathway would discharge through the plant vent which has a much smaller dispersion factor as compared to that of the fuel handling building. For these reasons, the FHA in the fuel building is bounding compared to any release from containment (open or closed). Certain sections in the VEGP FSAR are retained for historical purposes, but it is important to note that an FHA inside containment is not limiting regardless of containment configuration compared to a release from the fuel building.

Table 3.5a shows a comparison of atmospheric dispersion factors which illustrates this further.

High radiation in the CR makeup air intake results in isolation of the CR. A delay time of 10 minutes is conservatively assumed for radiation levels to reach the CREFS actuation setpoint.

The FHA analysis used assumptions and inputs that follow the guidance in RG 1.183. The key parameters and assumptions are listed in Attachment 6. An exception to the RG-1.183 linear heat generation limit (LHGR) of 6.3 kW/ft (footnote 11) has been requested. For Vogtle Units 1 and 2 it is requested that 40% of the rods be allowed to exceed the 6.3kW/ft limit and those 40% of rods be approved for a LHGR limit of 7.4 kW/ft. This is consistent with the LHGR limit depicted in Figure A.1 of PNNL-18212 Revision 1. To demonstrate that this request is acceptable, the gap release fractions assumed in the FHA analysis were taken from Table 2.9 of PNNL-18212 Revision 1. These values reflect increases in release fractions for Kr-85, I-132, and other Noble Gases. For conservatism, the source term developed for the FHA analysis assumed 100% of the failed rods exceeded the RG 1.183 footnote 11 limit. The source term at 70 hours after reactor shutdown used in the FHA analysis is provided in Attachment 6.

The calculated dose results are given in Table 3.5b. The calculated doses are within the RG 1.183 radiological dose acceptance criteria for an FHA. These TEDE criteria are 6.3 rem at the EAB for the worst two hours, 6.3 rem at the LPZ for the duration of the accident (24 hours) and 5 rem in the CR for the duration of the accident.

Table 3.5a: Comparison of X/Qs (s/m^3) for the FHA Analysis

Release	Receptor	0-2 Hours	2-8 Hours	8-24 Hours
U1 Hatch Door	U1 MCR	6.82E-04	4.85E-04	1.75E-04
U1 Vent	U1 MCR	1.70E-03	1.23E-03	4.30E-04
FHB (east)	U1 MCR	6.01E-03	4.44E-03	1.71E-03
U2 Hatch Door	U1 MCR	4.42E-04	3.64E-04	1.61E-04
U2 Vent	U1 MCR	7.71E-04	5.59E-04	2.35E-04
U1 Hatch Door	U2 MCR	4.30E-04	3.06E-04	1.19E-04
U1 Vent	U2 MCR	7.91E-04	5.59E-04	2.16E-04
FHB (west)	U2 MCR	6.01E-03	4.44E-03	1.71E-03
U2 Hatch Door	U2 MCR	7.16E-04	5.83E-04	2.45E-04
U2 Vent	U2 MCR	1.67E-03	1.20E-03	5.16E-04
Max		6.01E-03	4.44E-03	1.71E-03

Table 3.5b: FHA Analysis Results

Location/Dose Point	TEDE (REM)	Acceptance Criteria (REM)
Exclusion Area Boundary (EAB)	1.0	6.3
Low Population Zone (LPZ)	0.4	6.3
Control Room (CR)	3.9	5

3.6 Main Steam Line Break Accident

This event consists of a break in one main steam line outside of containment in which the faulted steam generator (SG) completely depressurizes and instantly releases the initial contents of the faulted SG secondary side to the environment. The plant cooldown continues by dumping steam with the intact SGs. In addition to the release of nuclides that are initially present in the SG secondary side, leakage of primary coolant into the SG secondary side occurs at a rate equal to 0.35 gpm to the faulted SG, and 0.65 gpm to the intact SGs (1.0 gpm total). This is conservative relative to the TS limit of 150 gallons per day per SG.

Two iodine spike cases are considered. In the pre-accident iodine spike case, a reactor transient is assumed to occur prior to the event in which the primary coolant iodine concentration has increased to the maximum TS value of 60 µCi/gm. For the concurrent spike case, the initial primary iodine activity release rate is at 500 times the equilibrium TS value of 1.0 µCi/g Dose Equivalent Iodine. This concurrent iodine spike is assumed to have a duration of 8 hours. No fuel damage is postulated to occur for the MSLB event.

Leakage from the RCS into all of the SGs, and steam release from the intact SGs, continues until the RCS is placed on RHR cooling after 20 hours. The leakage to the faulted SG is modeled as a direct flow from the RCS to the environment without partitioning. In the leakage to the intact SGs, noble gases are assumed to leak directly to the environment. A partition factor of 100 is applied to the iodine nuclides in the intact SGs.

The release locations from the faulted and intact SGs are conservatively taken as the most limiting MSIV area release locations. The CR is automatically realigned into the emergency ventilation mode upon receipt of a safety injection signal.

The analysis used assumptions and inputs that follow the guidance in RG 1.183. The key parameters and assumptions are listed in Attachment 7. The calculated dose results are given in Table 3.6. The calculated doses are within the RG 1.183 radiological dose acceptance criteria for a MSLB. These TEDE criteria are 25 rem at the EAB and LPZ for the fuel damage or pre-incident spike case, and 2.5 rem at the EAB and LPZ for the concurrent iodine spike case. The TEDE criteria is 5 rem for the CR occupant in both cases. The duration of the release is until the residual heat removal (RHR) system is placed in service (20 hours); the resultant doses are calculated for 30 days post-accident.

Table 3.6 - Calculated MSLB Accident Radiological Consequences

	TEDE (rem)		
	EAB	LPZ	Control Room
Calculated results			
Pre-Incident Spike	<0.1	<0.1	<0.1
Concurrent Iodine Spike	0.2	0.2	0.2

	TEDE (rem)		
	EAB	LPZ	Control Room
Dose acceptance criteria			
Fuel Damage or			
Pre-Incident Spike	25	25	5
Concurrent Iodine Spike	2.5	2.5	5

3.7 Steam Generator Tube Rupture Accident

The SGTR event represents an instantaneous rupture of a SG tube that releases primary coolant into the lower pressure secondary system. In addition to the break flow rate, primary-to-secondary leakage (consistent with CLB) occurs at a rate equal to 0.3 gpm to the ruptured and 0.7 gpm for the intact SGs (1.0 gpm total). Leakage into the SGs continues until the RCS is placed on residual heat removal (RHR) system cooling at 20 hours.

A portion of the break flow to the ruptured SG flashes to vapor based upon the thermodynamic conditions in the reactor and secondary coolant. The portion of the primary coolant that does flash in the SG secondary is released directly to the environment without mitigation. The break flow that does not flash mixes with the bulk water in the SG where the activity is released based upon the steaming rate and a partition factor. A SG partition factor of 100 is applied to the iodine nuclides.

Two iodine spike cases are considered. In the pre-accident iodine spike case, a reactor transient is assumed to occur prior to the event in which the primary coolant iodine concentration has increased to the maximum TS value of 60 µCi/gm. For the concurrent spike case, the initial primary iodine activity concentration is at the equilibrium TS value of 1.0 µCi/g Dose Equivalent Iodine. This concurrent iodine spike is assumed to have a duration of 8 hours. In both cases, as an initial condition, RCS activity includes an equilibrium noble gas concentration of 280 µCi/g dose equivalent Xe-133 and consideration of clad defects in 1% of the fuel rods for the other radioactive nuclides.

The release locations from the faulted and intact SGs are conservatively taken as a release from the main steam line room closest to the CR. This is conservative as this is where the main steam safety relief valves and (power operated) atmospheric relief valves (ARVs) are located. The CR is automatically realigned into the emergency ventilation mode upon receipt of a safety injection signal.

The SG blowdown sample lines are assumed open in the analysis and discharge from the faulted and intact steam generators for the duration of the event. There is no credit for partitioning of nobles, iodines, or particulates from the blowdown sample line flow. All of the blowdown sample line flow is assumed to flash and be released from the limiting location in the main steam line room in the auxiliary building (discussed above) for conservatism.

The analysis used assumptions and inputs that follow the guidance in RG 1.183. The key parameters and assumptions are listed in Attachment 8. The calculated dose results are given in Table 3.7. The analysis used a release duration of 20 hours and an exposure duration of 30 days. The calculated doses are within the RG 1.183 radiological dose acceptance criteria for a SGTR. These TEDE criteria are 25 rem at the EAB and LPZ for the pre-accident spike case, and 2.5 rem at the EAB and LPZ for the concurrent iodine spike case. The TEDE criteria is 5 rem for the CR occupant in both cases.

Table 3.7 - Calculated SGTR Accident Radiological Consequences

	TEDE (rem)		
	EAB	LPZ	Control Room
Calculated results			
Pre-Accident Spike	1.6	0.9	0.6
Concurrent Spike	1.4	0.8	0.5

	TEDE (rem)		
	EAB	LPZ	Control Room
Dose acceptance criteria			
Pre-Accident Spike	25	25	5
Concurrent Spike	2.5	2.5	5

3.8 Control Rod Ejection Accident

The Control Rod Ejection event involves a reactivity insertion that produces a short, rapid core power level increase, which results in fuel rod damage and localized melting. Two separate release pathways are evaluated independently: a release from containment and a release from the secondary system. In both cases, 10% of the noble gases and 10% of the iodine isotopes in the core are available for release from the fuel gap of the damaged fuel rods. In addition, 12% of the alkali metals and 5% of the other halogens (i.e., Br) are also assumed to be located in the fuel rod gap. It is assumed that the reactor coolant contains the maximum equilibrium iodine concentration (1.0 μCi/gm DE I-131) as well as noble gases based on 280 μCi/gm DE Xe-133 and other radionuclides corresponding to 1% defective fuel. The initial secondary side activity is assumed to be 10% of the primary side activity.

For releases from containment, 10% of the fuel rods in the core experience cladding failure and 0.25% of the fuel experiences melting. The activity in the fuel rod gap of the damaged fuel is instantaneously and uniformly mixed throughout the containment atmosphere. Moreover, 100% of the noble gases and 25% of the iodine isotopes in the melted fuel are also added to the fission product inventory in containment.

No credit is taken for removal by containment sprays or for deposition of elemental iodine on containment surfaces. Credit is taken for natural deposition of aerosols in containment based on a conservatively low removal coefficient. Activity is released from containment at the TS leak rate limit plus 5% for the first 24 hours and at half that rate after that.

For releases from the secondary system, 10% of the fuel rods in the core are breached and 0.25% of the fuel experiences melting. Activity released from the fuel is completely dissolved in the primary coolant and is available for release to the secondary system. In this case, 100% of the noble gases and 50% of the iodine isotopes in the melted fuel are also released into the reactor coolant. The noble gases are assumed to be released directly to the environment, and the remaining fission products are transported from the RCS to the SGs at 1 gpm, which is conservative relative to the TS limit of 150 gallons per day per SG. The leakage duration is assumed to continue until the termination of steaming from the SGs. The secondary side mass releases are conservatively assumed to last for 20 hours.

The atmospheric dispersion factors are conservatively taken as the most limiting release and receptor locations for each pathway and each averaging period. The CR ventilation system is automatically realigned into the emergency ventilation mode following receipt of a safety injection signal.

The analysis used assumptions and inputs that follow the guidance in RG 1.183. The key parameters and assumptions are listed in Attachment 9. The calculated dose results are given in Table 3.8. The calculated doses are within the RG 1.183 radiological dose acceptance criteria for a Control Rod Ejection. These

TEDE criteria are 6.3 rem at the EAB and LPZ, and 5 rem for the CR occupant. The accident release duration is 30 days for the Containment pathway, and until the residual heat removal (RHR) system is placed in service for the secondary pathway.

Table 3.8 - Calculated Control Rod Ejection Accident Radiological Consequences

	TEDE (rem)		
	EAB	LPZ	Control Room
Containment Release	1.5	1.8	0.6
Secondary Release	0.5	0.4	1.1
Dose acceptance criteria	6.3	6.3	5

3.9 Locked Rotor Accident

The Locked Rotor Accident dose analysis is defined by the 5% of the fuel rods which become damaged by the event. A radial peaking factor of 1.7 is assumed. Radionuclides released from the fuel are instantaneously and uniformly mixed throughout the primary coolant. Noble gases are released directly to the environment, and the remaining isotopes are transported to the SGs at a rate of 1 gpm. This continues for 20 hours, by which time the RCS is placed on residual heat removal (RHR) system cooling.

The quantity of the fission products released from the failed fuel dominates the RCS activity during the event; however, the initial nuclide concentration in the RCS prior to the event is considered. The analysis also includes the dose contribution from the release of iodine initially present in the SG secondary side. The release locations are conservatively taken as the most limiting release locations from the MSIV area. The analysis assumes that the CR isolates (608 seconds from transient initiation) on high radiation from the control room intake radiation monitors and then enters the emergency ventilation mode (698 seconds from transient initiation).

The analysis used assumptions and inputs that follow the guidance in RG 1.183. The key parameters and assumptions are listed in Attachment 10. The calculated dose results are given in Table 3.9. The calculated doses are within the RG 1.183 radiological dose acceptance criteria for a Locked Rotor Accident. These TEDE criteria are 2.5 rem at the EAB and LPZ, and 5 rem for the CR occupant. The duration of the release is 20 hours and the dose results are calculated for 30 days.

Table 3.9 - Calculated Locked Rotor Accident Radiological Consequences

	TEDE (rem)		
	EAB	LPZ	Control Room
Calculated results	<0.1	<0.1	0.3
Dose acceptance criteria	2.5	2.5	5

3.10 Equipment Qualification

Impacts to environmental qualification of equipment were evaluated against the legacy approved approach (TID-14844) for Vogtle. The evaluation concluded that environmental qualification of equipment at Vogtle is unaffected by the adoption of RG-1.183.

TSTF- 51 and TSTF-471

3.11 Accident Analysis

The control of movement of loads heavier than a fuel assembly over irradiated fuel is described in VEGP FSAR Section 9.1.5. Section 9.1.5 also demonstrates conformance to NUREG-0612 and includes the following topics:

- Safe Load Paths
- Establishment of load handling procedures
- Implementation of standards with respect to: training of crane operators, use of special lifting devices, use of slings, and design, inspection, testing, and maintenance of cranes
- Heavy load drop analyses

As summarized in FSAR sub-subsection 9.1.5.3.1, "Postulated Loads Inside Containment," and Section 9.1.5.3.2 “Postulated Loads Inside Fuel Handling Building” the effects of heavy load drops have been evaluated and as presented in FSAR Table 9.1.5-3 satisfy NUREG-0612 criteria. It has been verified that the buckling load on affected fuel assemblies would not exceed design limits and that there will be no consequential damage to the structural integrity of the reactor vessel, reactor vessel nozzles, or RCS loop piping. Therefore, core cooling capability and the integrity of the fuel cladding will be maintained. Thus, the inadvertent drop of a heavy load at VEGP would have no impact on the health or safety of the public. The proposed license amendment does not impact or alter the VEGP load drop analysis described in Subsection 9.1.5 of the FSAR.

3.12 Summary of FHA Dose Results

A summary of the AST FHA dose results is located Section 3.5 of this Enclosure.

3.13 Acceptability of the Proposed Change

Following reactor shutdown, decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed change is based on the results of the VEGP FHA analyses that assumes a fuel decay period of at least 70 hours reducing the radionuclide inventory available for release to the environment in the event of an FHA. Following sufficient decay occurring, the primary success path for mitigating the FHA no longer includes the functioning of the active containment systems to ensure off-site and MCR doses remain below the 10 CFR 50.67 dose limits.

When referring to movement of recently irradiated fuel in the proposed change, the term “recently” is described in the associated TS Bases, consistent with TSTF-51, as fuel that has occupied part of a critical reactor core within the previous 70 hours. This time is based on the input assumption in the FHA analysis, which shows that, following this fuel decay period, off-site and MCR doses remain below the 10 CFR 50.67 dose limits without reliance on containment closure, auxiliary building closure, or SFP room filtration from the PPAFES and associated actuation instrumentation.

The operability requirements of the Technical Specifications specified herein are modified to reflect that reactor vessel water level, SFP water level, and decay time are the primary success path for mitigating an FHA. The isolation, pressurization, and filtration of the MCR continues to be assumed in the FHA analysis, and therefore, these requirements are not modified by the proposed amendment request.

As specified in Attachment 12 of this Enclosure, SNC will establish administrative controls that ensure the following guidelines specified in Sub-subsection 11.3.6.5 of NUMARC 93-01 (Reference 8), will be included in the assessment of systems removed from service during fuel handling or core alterations:

- Ventilation system and radiation monitor availability should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the proposed license amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.

- A single normal or contingency method to promptly close containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure; rather the prompt methods should enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

Regarding proposed deletion of TS 3.9.1, Required Action A.1, suspending core alterations has no effect on the initial conditions or mitigation of refueling mode (i.e., Mode 6) design basis accidents or transients, and this requirement applies an operational burden with no corresponding safety benefit. The purpose of maintaining boron concentration within limits in Mode 6 ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. If boron concentration is not within the required limit, the appropriate action is to immediately suspend positive reactivity additions (current Required Action A.2). There are two evolutions encompassed under the term Core Alterations that could negatively affect the shutdown margin; the addition of fuel and the withdrawal of control rods. However, Required Action A.2 (proposed Required Action A.1), requires immediate suspension of positive reactivity changes. The immediate suspension of positive reactivity changes would include both the addition of fuel to the reactor vessel and the withdrawal of control rods. Another accident considered in MODE 6 that could affect shutdown margin is a dilution event. A boron dilution accident is mitigated by stopping the dilution. Additionally, allowing continuation of some core alterations may, in fact, increase core shutdown margin. For example, removal of one or more irradiated fuel assemblies from the core in the proper sequence or inserting a reactivity control component can increase overall shutdown margin. Therefore, prohibiting core alterations in this condition is unnecessary and possibly eliminates an option to restore core shutdown margin. As a result, the requirement to immediately suspend core alterations when boron concentration is not within the required limit in Mode 6 is deleted.

Regarding the addition of the proposed Note to TS 3.9.3 Required Action A.1: the existing Required Actions A.1 and A.2, which require immediately suspending core alterations and positive reactivity additions, are unchanged. Suspending positive reactivity additions prohibits diluting the boron concentration of the coolant in the RCS, the loading of fuel assemblies or sources into the core, or the removal of reactivity control components. Suspending core alterations also prohibits any movement of fuel, sources, or reactivity control components in the reactor core. A proposed Note to Required Action A.1 permits fuel assemblies, sources, and reactivity control components to be moved, if necessary, to restore an inoperable source range neutron flux monitor. The source range neutron flux monitors are located outside the reactor core in wells in the concrete reactor shield. The radiation levels in these wells can be very high if fuel assemblies are nearby. Troubleshooting, repair, or replacement of the inoperable source range neutron flux monitors may require moving fuel, sources, or reactivity control components away from the source range neutron flux monitor location to minimize the radiation dose to the workers. Also, in accordance with the definition of Core Alterations specified in TS Section 1.1, if movement of a fuel assembly, source, or reactivity control component is in progress when it is discovered that the required source range neutron flux monitor is inoperable, the component may be placed in a safe location. Therefore, in the event one or more source range neutron flux monitors are inoperable, the required actions continue to minimize actions that could result in reactivity changes within the core, while providing the ability to safely restore source range neutron monitoring capability.

3.14 Variations from TSTF-51 and TSTF-471

The proposed amendment is based on the STS changes described in TSTF-51, Revision 2, and TSTF-471, Revision 1, but SNC proposes variations from the NUREG-1431 markups in TSTF-51 and TSTF-471, as identified below and include differing TS numbers and TS titles, where applicable.

1. The definition of CORE ALTERATION is being retained in TS Section 1.1, "Definitions," because this terminology continues to be used in a number of TSs, which are not being modified as a result of this amendment request. This is an administrative variation from TSTF-471.

2. The control room emergency filtration system (CREFS) actuation instrumentation and the CREFS continue to be assumed to provide isolation, pressurization, and filtration of the MCR in the event of an FHA. Since this system and associated isolation instrumentation are mitigation systems necessary to maintain dose to personnel in the MCR below the regulatory and regulatory guidance limits for an FHA, the following TSs and support TSs and associated Bases are not modified:

   - TS 3.3.7, "Control Room Emergency Filtration/Pressurization System (CREFS) Actuation Instrumentation,"
   - TS 3.7.10, "Control Room Emergency Filtration/Pressurization System (CREFS) - Both Units Operating,"
   - TS 3.7.11, "Control Room Emergency Filtration/Pressurization System (CREFS) - One Unit Operating,"
   - TS 3.8.2, "AC Sources – Shutdown,"
   - TS 3.8.5, "DC Sources – Shutdown,"
   - TS 3.8.8, "Inverters – Shutdown,"
   - TS 3.8.10, "Distribution Systems – Shutdown," and
   - TS 3.9.7, "Refueling Cavity Water Level."

   This is a plant-specific variation from TSTF-51 and 471.

3. The applicability requirements associated with the containment ventilation isolation instrumentation are shown in TS Table 3.3.6-1. This is a presentation difference from the applicability requirements shown in the NUREG-1431 TS 3.3.6 marked up pages in TSTF-51. However, the proposed changes to footnote (c) in TS Table 3.3.6-1 are consistent with those shown in TSTF-51. These proposed changes are administrative variations from TSTF-51.

4. TS 3.9.3, "Nuclear Instrumentation," Required Actions were not modified in accordance with TSTF-471. However, proposed Note added to Required Action A.1 is consistent with the intent of the proposed Note in TSTF-571-T, "Revise Actions for Inoperable Source Range Neutron Flux Monitor" (Reference 6). TSTF-571-T was accepted for use by the NRC as documented in a letter to the TSTF dated October 4, 2018 (Reference 7). Movement of fuel sources and reactivity control components within the reactor vessel is currently covered by the Core Alteration definition. Since

the VEGP TSs retain the definition of Core Alteration, the required action continues to require suspension of core alterations and the note was modified to use the term Core Alterations. These proposed changes are considered administrative variations from TSTF-471.

SNC considers the differences from TSTF-51 and TSTF-471 listed herein to be either: 1) necessary variations to maintain the requirements for required safety systems assumed in the VEGP FHA analysis; or 2) minor variations or deviations that are administrative in nature.

TSTF-490

3.15 TSTF-490

SNC has reviewed References 2 and 3. SNC has also applied the methodology in Reference 1 to develop the proposed TS changes. SNC has also concluded that the justifications presented in TSTF-490, Revision 0, and the model SE prepared by the NRC staff are applicable to VEGP, Units 1 and 2, and justify this amendment for the incorporation of the changes to the VEGP TS.

3.15.1 Variations

SNC is not proposing any variations or deviations from the applicable parts of the NRC staff's model safety evaluation. SNC is proposing the following variations from the TS changes described in the TSTF-490, Revision 0.

The VEGP TS include a Surveillance Frequency Control Program. Therefore, the periodic Surveillance Frequencies shown in TSTF-490 are replaced with the statement, "In accordance with the Surveillance Frequency Control Program."

Additionally, SNC is proposing an administrative change which deletes the global NOTE regarding LCO 3.0.4c applicability. This NOTE is replaced by the same NOTE added to Action A (administrative change) and to Action B (TSTF-490). The NOTE does not apply to Action C (shutdown action).

3.15.2 Calculation of Dose Equivalent Xenon-133

TSTF-490 Revision 0 (described in USNRC ADAMS ML052630462, ML070250176, ML16113A402, and ML18256A027) defines Dose Equivalent Xenon-133 (DE Xe-133) as that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of the Noble Gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. The dose consequences of a given radionuclide is proportional to the product of its RCS concentration and its dose conversion factor (DCF):

Dose Consequences α [Concentration] * [Dose Conversion Factor]
The definition of DE Xe-133 can be developed as follows:

$$[\text{DE Xe-133 } (\mu\text{Ci/gm})] * [DCF_{Xe\text{-}133}] = \Sigma[C_{NOM\text{-}i} * DCF_i]$$

$$\text{DE Xe-133 } (\mu\text{Ci/gm}) = \left\{\Sigma[C_i * DCF_i]\right\} / DCF_{Xe\text{-}133}$$

where:

$C_{NOM\text{-}i}$ = Nominal RCS concentration of individual noble gas isotope i (µCi/g) based on clad defects in 1% of the fuel rods

DCF_i = DCF for individual isotope i [(REM/sec)/(µCi/m^3)]

$DCF_{Xe\text{-}133}$ = DCF for Xe-133 [(REM/sec)/(µCi/m^3)]

The determination of Dose Equivalent Xe-133 is performed using the effective DCFs for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

Since the nominal Noble Gas concentrations based upon 1% cladding defects may differ from their Dose Equivalent Xe-133 concentrations, the nominal concentrations are adjusted to meet the RCS Dose Equivalent Xe-133 specific activity limit as follows:

Dose Equivalent Xe-133 Concentration for Noble Gas Isotope i = Adjustment Factor * Noble Gas Isotope i Nominal Concentration

$$C_{DEX\text{-}i}\ (\mu\text{Ci/g}) = \frac{\text{Dose Equivalent Xe-133 Limit } (\mu\text{Ci/g})}{\{\Sigma\ [C_{NOM\text{-}i} * DCF_i]\}/DCF_{Xe\text{-}133}} * C_{NOM\text{-}i}\ (\mu\text{Ci/g})$$

The Dose Equivalent Xe-133 Limit per TSTF-490 is 280 µCi/g. The calculation of the individual noble gases' dose equivalent Xe-133 concentrations are shown in the table below. The fuel design margin applied to the core source term is not applied here because as is apparent from the above equation for $C_{DEX\text{-}i}$, it will cancel out.

Nuclide	**$C_{NOM\text{-}i}$ (µCi/g)**	**DCF_i (REM/sec)/ (µCi/m^3)**	**$C_{NOM\text{-}i}$ * DCF_i (REM/sec)/ (g/m^3)**	**$C_{DEX\text{-}i}$ (µCi/g)**
Kr-85	9.03E+00	4.40E-10	3.98E-09	3.84E+00
Kr-85m	1.75E+00	2.77E-08	4.85E-08	7.46E-01

Nuclide	C_{NOM-i} (μCi/g)	DCF_i (REM/sec)/ (μCi/m^3)	C_{NOM-i} * DCF_i (REM/sec)/ (g/m^3)	C_{DEX-i} (μCi/g)
Kr-87	1.15E+00	1.52E-07	1.75E-07	4.87E-01
Kr-88	3.85E+00	3.77E-07	1.45E-06	1.64E+00
Xe-131m	3.43E+00	1.44E-09	4.93E-09	1.46E+00
Xe-133	2.70E+02	5.77E-09	1.56E-06	1.15E+02
Xe-133m	3.75E+00	5.07E-09	1.90E-08	1.59E+00
Xe-135	8.16E+00	4.40E-08	3.59E-07	3.47E+00
Xe-135m	5.15E-01	7.55E-08	3.88E-08	2.19E-01
Xe-138	6.50E-01	2.13E-07	1.39E-07	2.77E-01

C_{NOM-i} = Nominal RCS Noble Gas concentration for isotope i (μCi/g)

DCF_i = Effective DCF for isotope i, Table III.1, FGR-12 (REM/sec)/(μCi/m^3)

DE Xe-133 = Dose Equivalent Xe-133 concentration (μCi/g)

DE Xe-133 = $[\sum (C_{NOM-i} * DCF_i]/DCF_{Xe-133}$

$\sum (C_{NOM} * DCF]_i$ = 3.80E-06 (REM/sec)/(g/m^3)
DCF_{Xe-133} = 5.77E-09 (REM/sec)/(μCi/m^3)
DE Xe-133 = 659 μCi/g

$$Adjustment\ Factor = \frac{Dose\ Equivalent\ Xe - 133\ Limit}{DE\ Xe - 133} = \frac{280\mu Ci/g}{659\mu Ci/g} = 0.425$$

C_{DEX-i} = 0.425 * C_{NOM-i}

4.0 Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

Title 10 Code of Federal Regulations Section 50.36, "Technical Specifications"

Changes to the VEGP TSs are proposed for the adoption of TSTF-51, TSTF-471, and TSTF-490. A description of these proposed changes and their relationship to applicable regulatory requirements and guidance are provided herein.

Title 10 Code of Federal Regulations Section 50.67, "Accident Source Term"

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 the DBA analysis with an AST. The direction provided in 10 CFR 50.67 is that

licenses who seek to revise their current accident source term in design basis radiological consequence analyses shall apply for a LAR under 10 CFR 50.90.

4.1.1 Additional Applicable Regulatory Criteria for TSTF-51 and TSTF-471

The TSs satisfy 10 CFR 50.36, “Technical specifications.” The following systems and parameters meet one or more of the criteria of 10 CFR 50.36(c)(2)(ii):

- Containment and associated containment purge and exhaust isolation instrumentation,
- Piping Penetration Area Filtration and Exhaust System (PPAFES) and associated actuation instrumentation,
- Boron concentration requirement in Mode 6, and
- Neutron instrumentation requirements in Mode 6.

The proposed amendment revises the TS applicability of these systems and parameters to eliminate the requirements during core alterations, and during movement of irradiated fuel assemblies that have decayed beyond the decay period assumed in the VEGP FHA analysis because these requirements are no longer assumed in the mitigation of an FHA or the potential radioactive release as a result of dropping of a non-irradiated fuel assembly, source, or reactivity control component onto the reactor core during core alterations. The proposed amendment does not alter requirements associated with the CREFS and associated instrumentation, which are assumed to mitigate the effects of a radiological release to MCR personnel due to an FHA, and continues to maintain requirements associated with structures, systems, and components that are part of the primary success path and actuate to mitigate the related design basis accidents and transients. The proposed amendment continues to provide appropriate remedial actions and shutdown requirements required by 10 CFR 50.36(c)(2)(i) for any system requiring an LCO pursuant the criteria of 10 CFR 50.36(c)(2)(ii).

10 CFR 50.67, “Accident source term” – Note that this License Amendment request includes conversion of the VEGP source terms to the ASTs consistent with 10 CFR 50.67. Thus, the modified The VEGP FHA analysis of record meets the requirements of 10 CFR 50.67. Accident source terms have not been modified as a result of the proposed amendment. SNC has determined that the inputs and assumptions related to atmospheric dispersion related to the FHA analysis are not changed as a result of the proposed license amendment. Therefore, the revised VEGP FHA analysis continues to meet the requirements of 10 CFR 50.67.

In addition, the following 10 CFR Part 50, Appendix A General Design Criteria (GDCs) are related to the proposed change:

GDC 13: Instrumentation and control. The proposed amendment does not alter the design of the applicable instrumentation that monitor

variables and systems over their anticipated ranges for normal operation for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety.

GDC 16: Containment design. The proposed amendment does not alter the containment design or the associated systems' design. The containment and associated systems, when required, will continue to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment as previously licensed and approved by the NRC. During movement of recently irradiated fuel assemblies when containment integrity is relaxed, SNC, as previously committed, will continue to establish and implement administrative controls that ensure that the open personnel airlock and any open penetrations can and will be promptly closed, following containment evacuation, in the event of an FHA inside containment.

GDC 19: Control room. The proposed amendment does not alter the design or operation of the control room envelope or the CREFS. To support the proposed amendment, the input assumptions have been revised in the FHA analysis. However, FHA analysis results show that the radiological dose to the MCR personnel continues to be within the requirements of GDC-19 as updated for consistency with the TEDE criterion in 10 CFR 50.67.b.2.iii. Adequate radiation protection continues to be provided permitting access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

GDC 20: Protection system functions. The proposed amendment does not alter the design of reactivity control protection systems or instrumentation that sense accident conditions to initiate systems or components important to safety. The change relaxes the requirements for instrumentation of systems not assumed in the mitigation of an FHA.

GDC 21: Protection system reliability and testability. The proposed amendment does not alter the design of any protection system, including the containment ventilation instrumentation and the CREFS actuation instrumentation. Therefore, the protection system design continues to provide high functional reliability and inservice testability commensurate with the safety functions to be performed and continues to be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy. The containment ventilation instrumentation and the CREFS actuation instrumentation designs continue to permit periodic testing of its functioning when the reactor is in operation as previously licensed and approved by the NRC.

GDC 22: Protection system independence. The proposed amendment does not alter the design of any protection system, including the containment ventilation instrumentation and the CREFS actuation instrumentation. Therefore, the protection system design continues to

assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function to the extent previously licensed and approved by the NRC.

GDC 23: Protection system failure modes. The proposed amendment does not alter the design of any protection system, including the containment ventilation instrumentation and the CREFS actuation instrumentation. Therefore, the protection system design continues to fail into a safe state or into a state demonstrated to be acceptable as previously licensed and approved by the NRC.

GDC 24: Separation of protection and control systems. The proposed amendment does not alter the design of any protection system, including the containment ventilation instrumentation and the CREFS actuation instrumentation. Therefore, the protection system design continues to be separated from control systems as previously licensed and approved by the NRC.

GDC 64: Monitoring radioactivity releases. The proposed amendment does not alter the design of any radioactivity monitoring instrumentation, including the containment ventilation instrumentation and the CREFS actuation instrumentation. Means continue to be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents (e.g., FHA).

4.1.2 Additional Applicable Regulatory Criteria for TSTF-490

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on March 19, 2017 (Reference 2), the NRC Notice for Comment published on November 20, 2006 (Reference 3), and TSTF-490, Revision 0.

4.2 Precedent

4.2.1 A number of AST submittals have been reviewed and approved by the NRC since RG 1.183 was published and have helped to inform the content of this application, specifically Joseph M. Farley Nuclear Plant – Units 1 and 2, License Amendments 216 and 213, respectively (NRC ADAMS Accession No. ML17271A265).

4.2.2 STS Traveler TSTF-51 was approved by the NRC staff and incorporated into the STS NUREGs, Revision 2, published in June 2001, which was also approved by the NRC staff. A number of facilities have adopted, as technically practical, TSTF-51. For example: Joseph M. Farley Nuclear Plant, Units 1 and 2, License Amendments 223 and 220, respectively (NRC ADAMS Accession No. ML19071A138), Indian Point Nuclear

Generating Unit 2, License Amendment 238 (NRC ADAMS Accession Nos. ML033160528 and ML033210260), North Anna Power Station, Units 1 and 2, License Amendments 231 and 212, respectively (NRC ADAMS Accession Nos. ML021200265, ML021220108, and ML021220166), Beaver Valley Units 1 and 2, License Amendments 278 and 161, respectively (NRC ADAMS Accession Nos. ML070160593 and ML070390284), Watts Bar Nuclear Plant, Unit 1, License Amendment 35 (NRC ADAMS Accession Nos. ML020100062 and ML020280264), and Byron Units 1 and 2, License Amendments 147 and Braidwood Units 1 and 2, License Amendments 140 (NRC ADAMS Accession No. ML062340420).

4.2.3 STS Traveler TSTF-471 was approved for use by the NRC staff and incorporated into the applicable STS NUREGs, Revision 4, published in April 2012. Some facilities have adopted, as technically practical, TSTF-471. For example: Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Amendments 279 and 256, respectively (NRC ADAMS Accession Nos. ML062350447 and ML062690054).

4.2.4 STS Traveler TSTF-490 was approved for use by the NRC staff and incorporated into the applicable STS NUREGs, Revision 5, published in September 2021. Some facilities have adopted as technically practical, TSTF-490. For example: Salem Generating Station, Units 1 and 2, Amendments 337 and 318, respectively (NRC ADAMS Accession No. ML21110A052).

4.3 Significant Hazards Consideration

Southern Nuclear Operating Company (SNC) has evaluated the proposed changes to the Technical Specifications (TS) using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration.

The proposed amendment revises certain Technical Specifications (TSs) to remove the requirements for engineered safety feature (ESF) systems to be operable after sufficient radioactive decay of irradiated fuel has occurred following a plant shutdown. Following sufficient radioactive decay, these systems are no longer required during a fuel handling accident (FHA) to ensure main control room (MCR) personnel dose remains below the 10 CFR 50.67(b)(2)(iii) dose limit and off-site dose remains below the accident dose limit specified in the NRC standard review plan (SRP), which represent a small fraction of the 10 CFR 50.67 dose limits.

The proposed amendment also revises certain TS actions that are not needed to mitigate accidents postulated during shutdown. Specifically, the requirement to immediately suspend core alterations when boron concentration is not within the required limit in refueling condition is deleted. In addition, when one or more required source range neutron flux monitors are inoperable in the refueling condition, a note added to the actions will permit fuel assemblies, sources, and reactivity control components to be moved if necessary to restore an inoperable source range neutron flux monitor to operable status.

Furthermore, SNC has reviewed the proposed no significant hazards consideration determination for TSTF-490 published in the Federal Register on March 19, 2007 (72 FR 12838) as part of the Consolidated Line Item Improvement Process (CLIIP). SNC has concluded that the proposed determination presented in the notice is applicable to VEGP and the determination is hereby incorporated by reference.

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "issuance of amendment," as discussed below:

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

   Response: No

   There are no physical changes to the plant being introduced by the proposed changes to the accident source term. Implementation of Alternative Source Term (AST) and the new atmospheric dispersion factors have no impact on the probability for initiation of any Design Basis Accidents (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. 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 Final Safety Analysis Report (FSAR).

   Based on the AST analyses, there are no proposed changes to performance requirements and no proposed revision to the parameters or conditions that could contribute to the consequences 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 (X/Qs) have been established. Based on the results of these analyses, it has been demonstrated that the Control Room and off-site dose consequences of the limiting events considered in the analyses meet the regulatory guidance provided for use with the AST, and the doses are within the limits established by 10 CFR 50.67.

   The proposed amendment does not involve a physical change to the containment or spent fuel area systems, nor does it change the safety function of the containment or associated instrumentation. The subject ESF systems are not assumed in the mitigation of an FHA after sufficient radioactive decay of irradiated fuel has occurred. The revised FHA dose analysis shows that MCR dose remains below the 10 CFR 50.67(b)(2)(iii) dose limit and off-site dose remains below the accident dose limit specified in the NRC SRP, which represents a small fraction of the 10 CFR 50.67 dose limits.

Elimination of the action to suspend core alterations in the event boron concentration is not within the required limit in refueling condition does not alter the initiation or consequences of a boron dilution event and the required actions continue to prohibit positive reactivity additions until reactor core shutdown margin can be restored to within the required limit.

Permitting fuel assemblies, sources, and reactivity control components to be moved to restore an inoperable source range neutron flux monitor to operable status when one or more required source range neutron flux monitors are inoperable does not significantly alter the probability or consequences of any previously evaluated refueling accident or transient. The required actions continue to minimize actions that could result in reactivity changes within the core, while providing the ability to safely restore source range neutron monitoring capability.

Therefore, it is concluded that the proposed amendment does not involve a significant increase in the probability or the consequences of an accident previously evaluated.

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

   Response: No

   No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Implementation of the AST and the associated Technical Specification changes and new X/Qs have no impact to the initiation of any DBAs. These changes do not affect the design function of modes of operation of structures, systems and components in the facility prior to a postulated accident. Since structures, systems and components are operated no differently after the AST implementation, no new failure modes are created by this proposed change. The AST change itself does not have the capability to initiate accidents.

   With respect to a new or different kind of accident, there are no proposed design changes to the safety related plant structures, systems, and components (SSCs); nor are there any changes in the method by which safety related plant SSCs perform their specified safety functions. The proposed amendment will not affect the normal method of plant operation or revise any operating parameters. No new accident scenarios, transient precursor, failure mechanisms, or limiting single failures will be introduced as a result of this proposed change and the failure modes and effects analyses of SSCs important to safety are not altered as a result of this proposed change. The proposed amendment does not alter the design or performance of the related SSCs, and, therefore, does not constitute a new type of test.

   No changes are being proposed to the procedures that operate the plant equipment and the change does not have a detrimental impact on the

manner in which plant equipment operates or responds to an actuation signal.

Consequently, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   The AST analyses 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. Thus, by meeting the applicable regulatory limits for AST, there is no significant reduction in a margin of safety.

   The proposed amendment does not involve a physical change to the containment, nor does it change the safety function of the containment or associated instrumentation. The subject ESF systems are not assumed in the mitigation of an FHA after sufficient radioactive decay of irradiated fuel has occurred. The revised VEGP FHA dose analysis shows that MCR dose remains below the 10 CFR 50.67(b)(2)(iii) dose limit and off-site dose remains below the accident dose limit specified in the NRC SRP, which represents a small fraction of the 10 CFR 50.67 dose limits.

   Elimination of the action to suspend core alterations does not reduce the margin of safety in the event boron concentration is not within the required limit in refueling condition because the remaining required actions continue to prohibit positive reactivity additions until reactor core shutdown margin can be restored to within the required limit.

   Permitting fuel assemblies, sources, and reactivity control components to be moved to restore an inoperable source range neutron flux monitor to operable status when one or more required source range neutron flux monitors are inoperable does not significantly reduce the margin of safety. The required actions continue to minimize actions that could result in reactivity changes within the core, while providing the ability to safely restore source range neutron monitoring capability.

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

4.4 Conclusion

In conclusion, based on the considerations discussed above, SNC concludes: (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 amendment will not be inimical to the common defense and security or the health and safety of the public.

5.0 Environmental Consideration

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

Furthermore, in regard to the implementation of TSTF-490, SNC has reviewed the environmental consideration included in the model SE published in the Federal Register on March 19, 2007 (72 FR 12838) as part of the CLIIP. SNC has concluded that the staff's findings presented therein are applicable to VEGP and the determination is hereby incorporated by reference for this application.

6.0 References

1. Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

2. Federal Register 71 FR 12838, "Notice of Availability of Model Application Concerning Technical Specification (TS) Improvement Regarding TSTF-490, "Deletion of E Bar Definition and Revision to Reactor Coolant System Specific Activity TS Using Consolidated Line Item Improvement Process,"" dated March 19, 2007 (NRC ADAMS Accession No. ML070250176)

3. Federal Register 71 FR 67170, "Notice of Opportunity to Comment on Model Safety Evaluation and Model License Amendment Request on Technical Specification Improvement Regarding Deletion of E Bar Definition and Revision to Reactor Coolant System Specific Activity Technical Specification Using the Consolidated Line Item Improvement Process," dated November 20, 2006

4. Regulatory Guide (RG) 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"

5. STS Change Traveler TSTF-286, "Define Operations Involving Positive Reactivity Additions," Revision 2, dates April 3, 2000.

6. Letter from Technical Specification Task Force to NRC, “TSTF Input to Lifting the Suspension of Acceptance of Amendment Requests to Adopt TSTF-51, TSTF-471, and TSTF-286,” dated August 9. 2018 (NRC ADAMS Accession No. ML18221A561).

7. Letter from V.G. Cusumano (NRC) to Technical Specification Task Force, “Plant Specific Adoption of Travelers TSTF-51, Revision 2, “Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations,” TSTF-471, Revision 1, “Eliminate Use of Term Core Alterations in Actions and Notes,” and TSTF-286, Revision 2, “Operations Involving Positive Reactivity Additions,” dated October 4, 2018 (NRC ADAMS Accession No. ML17346A587).

8. Nuclear Energy Institute NUMARC 93-01, Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 4A, April 2011 (NRC ADAMS Accession No. ML11116A198).

Vogtle Electric Generating Plant – Units 1 & 2
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Attachment 1

Technical Specification Pages (Markup)

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or other reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, September 1988.~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine

(continued)

	isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11.
DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

(continued)

1.1 Definitions (continued)

~~Ē – AVERAGE DISINTEGRATION ENERGY~~	~~Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 14 minutes, making up at least 95% of the total noniodine activity in the coolant.~~
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known to not interfere with the operation of leakage detection systems; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- Only applicable during ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment. ----------	C.1 Place and maintain containment purge and exhaust valves in closed position. OR	Immediately
No radiation monitoring channels OPERABLE. OR Required Action and associated Completion Time for Condition A not met.	C.2 Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge supply and exhaust isolation penetrations not in required status.	Immediately

Table 3.3.6-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1,2,3,4	2	SR 3.3.6.6	NA
2. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Radiation	1,2,3,4,6[c]	2[a]	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	
a. Gaseous (RE-2565C)				(b)
b. Particulate (RE-2565A)				(b)
c. Iodine (RE-2565B)				(b)
d. Area Low Range (RE-0002, RE-0003)				≤ 15 mr/h[c] ≤ 50x background[d]
4. Safety Injection[d]	1,2,3,4	Refer to LCO 3.3.2, "ESFAS Instrumentation,"Function 1, for all initiation functions and requirements.		

(a) Containment ventilation radiation (RE-2565) is treated as one channel and is considered OPERABLE if the particulate (RE-2565A) and iodine monitors (RE-2565B) are OPERABLE or the noble gas monitor (RE-2565C) is OPERABLE.

(b) Setpoints will not exceed the limits of Specifications 5.5.4.h and 5.5.4.i of the Radioactive Effluent Controls Program.

(c) During ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies within containment.

(d) During MODES 1, 2, 3, and 4.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 ~~The specific activity of the reactor coolant shall be within limits.~~ RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, ~~and~~ 2, 3, and 4
~~MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.~~

ACTIONS

~~----------NOTE----------~~
~~LCO 3.0.4c is applicable.~~
~~----------~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit ~~> 1.0 µCi/gm~~.	----------NOTE---------- LCO 3.0.4.c is applicable. ---------- A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 µCi/gm ~~within the acceptable region of Figure 3.4.16-1~~. AND A.2 Restore DOSE EQUIVALENT I-131 to within limit.	Once per 4 hours 48 hours
B. ~~Gross specific activity of the reactor coolant~~ DOSE EQUIVALENT XE-133 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ---------- B.1 ~~Be in MODE 3 with T_{avg} < 500°F~~ Restore DOSE EQUIVALENT XE-133 to within limit.	~~6~~ 48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > 60 µCI/gm~~in the unacceptable region of Figure 3.4.16-1~~.	C.1 Be in MODE 3~~ with T_{avg} < 500°F~~. AND C.2 Be in MODE 5	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 280 µCi/gm.~~Verify reactor coolant gross specific activity ≤ 100/Ē µCi/gm.~~	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 µCi/gm.	In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	----------NOTE---------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ---------- Determine Ē from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	In accordance with the Surveillance Frequency Control Program

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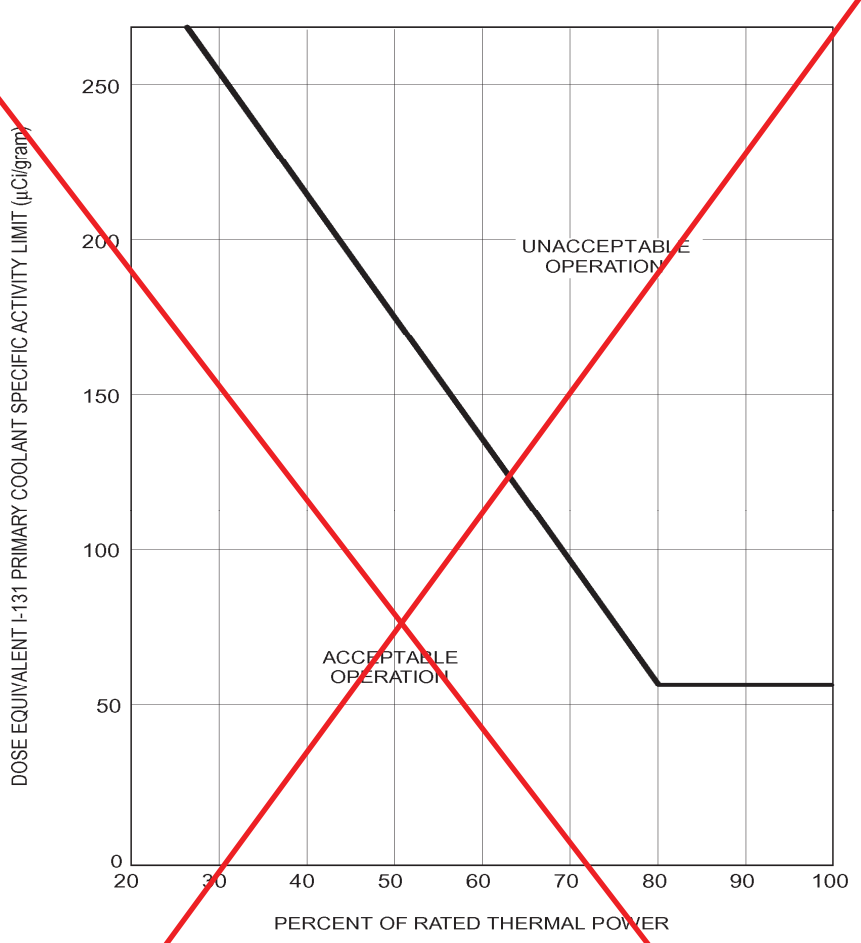

FIGURE 3.4.16-1
REACTOR COOLANT DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 µCi/gram DOSE EQUIVALENT I-131

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

----------NOTE----------
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	A.~~2~~1 Suspend positive reactivity additions.	Immediately
	AND	
	A.~~3~~2 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Isolation Valves

LCO 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.

----------NOTE----------
Valves in the flowpath from the RMWST, through the chemical mixing tank, to the suction of the charging pumps may be opened under administrative control provided the reactor coolant system is in compliance with Specification 3.9.1 and the high flux at shutdown alarm is OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Required Action A.~~3~~2 must be completed whenever Condition A is entered. ---------- One or more valves not secured in closed position.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	A.~~2~~1 Initiate actions to secure valve in closed position.	Immediately
	AND	
	A.~~3~~2 Perform SR 3.9.1.1 (verify boron concentration).	12 hours

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One source range neutron flux monitor inoperable.	A.1 ----------NOTE---------- CORE ALTERATIONS may continue to restore an inoperable source range neutron flux monitor. ----------	Immediately
	~~A.1~~ Suspend CORE ALTERATIONS.	
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
B. ----------NOTE---------- Condition A entry is required when Condition B is entered. ---------- Two source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1 (verify boron concentration).	Once per 12 hours

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. The emergency and personnel air locks are isolated by at least one air lock door, or if open, the emergency and personnel air locks are isolable by at least one air lock door with a designated individual available to close the open air lock door(s); and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by at least two OPERABLE Containment Ventilation Isolation valves

----------NOTE----------
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: ~~During CORE ALTERATIONS,~~
During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	A.~~2~~1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

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TSTF-51, TSTF-471, and TSTF-490

Attachment 2

Technical Specification (Clean Typed Pages)

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or other reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11.

(continued)

1.1 Definitions (continued)

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

(continued)

1.1 Definitions (continued)

LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known to not interfere with the operation of leakage detection systems; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except primary to secondary LEAKAGE) through a fault in an RCS component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

(continued)

1.1 Definitions (continued)

OPERABLE — OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:) a. Described in Chapter 14 of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, Cold Overpressure Protection System (COPS) arming temperature and the nominal PORV setpoints for the COPS, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these operating limits is addressed in individual specifications.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3625.6 MWt.

(continued)

1.1 Definitions (continued)

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

1.1 Definitions (continued)

THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- Only applicable during movement of recently irradiated fuel assemblies within containment. ----------	C.1 Place and maintain containment purge and exhaust valves in closed position.	Immediately
	OR	
No radiation monitoring channels OPERABLE.	C.2 Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge supply and exhaust isolation penetrations not in required status.	Immediately
OR		
Required Action and associated Completion Time for Condition A not met.		

Table 3.3.6-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1,2,3,4	2	SR 3.3.6.6	NA
2. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Radiation	1,2,3,4,6[c]	2[a]	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	
a. Gaseous (RE-2565C)				(b)
b. Particulate (RE-2565A)				(b)
c. Iodine (RE-2565B)				(b)
d. Area Low Range (RE-0002, RE-0003)				≤ 15 mr/h[c] ≤ 50x background[d]
4. Safety Injection[d]	1,2,3,4	Refer to LCO 3.3.2, "ESFAS Instrumentation,"Function 1, for all initiation functions and requirements.		

(a) Containment ventilation radiation (RE-2565) is treated as one channel and is considered OPERABLE if the particulate (RE-2565A) and iodine monitors (RE-2565B) are OPERABLE or the noble gas monitor (RE-2565C) is OPERABLE.

(b) Setpoints will not exceed the limits of Specifications 5.5.4.h and 5.5.4.i of the Radioactive Effluent Controls Program.

(c) During movement of recently irradiated fuel assemblies within containment.

(d) During MODES 1, 2, 3, and 4.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ---------- A.1 Verify DOSE EQUIVALENT I-131≤ 60 µCi/gm. <u>AND</u> A.2 Restore DOSE EQUIVALENT I-131 to within limit.	Once per 4 hours 48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ---------- B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > 60 µCI/gm.	C.1 Be in MODE 3. AND C.2 Be in MODE 5	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 280 µCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 µCi/gm.	In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

----------NOTE----------
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend positive reactivity additions. AND A.2 Initiate action to restore boron concentration to within limit.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Isolation Valves

LCO 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.

----------NOTE----------
Valves in the flowpath from the RMWST, through the chemical mixing tank, to the suction of the charging pumps may be opened under administrative control provided the reactor coolant system is in compliance with Specification 3.9.1 and the high flux at shutdown alarm is OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Action A.2 must be completed whenever Condition A is entered. ---------- One or more valves not secured in closed position.	A.1	Initiate actions to secure valve in closed position.	Immediately
	AND		
	A.2	Perform SR 3.9.1.1 (verify boron concentration).	12 hours

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One source range neutron flux monitor inoperable.	A.1 ----------NOTE---------- CORE ALTERATIONS may continue to restore an inoperable source range neutron flux monitor. ---------- Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
B. ----------NOTE---------- Condition A entry is required when Condition B is entered. ---------- Two source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1 (verify boron concentration).	Once per 12 hours

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. The emergency and personnel air locks are isolated by at least one air lock door, or if open, the emergency and personnel air locks are isolable by at least one air lock door with a designated individual available to close the open air lock door(s); and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by at least two OPERABLE Containment Ventilation Isolation valves

----------NOTE----------
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Alternative Source Term,
TSTF-51, TSTF-471, and TSTF-490

Attachment 3

Technical Specification Bases Pages (Markup) (For Information Only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term" (Ref. 4).

BASES

SAFETY LIMITS (continued)

Code, Section III, is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL 2.2.2 is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWB-5000.

4. 10 CFR ~~100~~50.67.

5. FSAR, Section 7.2.

BASES

APPLICABLE SAFETY ANALYSES (continued)

SDM satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable. The required SDM is specified in the COLR.

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits."

ACTIONS

The ACTIONS table is modified by a Note prohibiting transition to a lower MODE within the Applicability. LCO 3.0.4 already prohibits entry into MODE 5 from MODE 6, MODE 4 from MODE 5 and into MODE 3 from MODE 4 when SDM requirements are not met.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. FSAR, Subsection 15.4.9.

3. FSAR, Subsection 15.4.6.

4. 10 CFR ~~100~~50.67.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

The LSSS, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 20 and 10 CFR ~~100~~ 50.67 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~50.67

(continued)

BASES

BACKGROUND (continued)

the "as-found" value of a protection channel setting during a surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left tolerance around the NTSP to account for further drift during the next surveillance interval.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling (DNB),

2. Fuel centerline melt shall not occur, and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR ~~100~~ 50.67 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~ 50.67 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The ESFAS instrumentation is segmented into four distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured;

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event, within approximately 60 seconds. The isolation of the purge supply and exhaust valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment purge supply and exhaust isolation radiation monitors act as backup to the SI signal to ensure closing of the purge supply and exhaust valves for events occurring in MODES 1 through 4. Although not credited in the fuel handling accident analysis, ~~Manual~~ manual isolation (using individual valve handswitches) following a radiation alarm is the assumed means for isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses less than the acceptance criteria in Regulatory Guide 1.183 Revision 0~~are below 10 CFR 100 (Ref. 1) limits~~. Due to radioactive decay, containment is only required to isolate during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 70 hours).

The containment ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment ventilation isolation at any time by using either of two switches in the control room (containment isolation Phase A switches). Either switch actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one CIA handswitch and the interconnecting wiring to the actuation logic cabinet.

(continued)

BASES

LCO (continued)

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two channels of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI. The applicable MODES and specified conditions for the Containment ventilation isolation portion of these Functions are different and less restrictive than those for their SI roles. If one or more of the SI Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to their SI Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment ventilation isolation remains OPERABLE. During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies in containment, the required channels provide input to control room alarms to ensure prompt operator action to manually close the containment purge and exhaust valves. It is also acceptable during ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel to meet the requirements of this LCO by maintaining the radiation monitoring instrumentation necessary to initiate containment ventilation isolation OPERABLE, in accordance with the requirements stated for MODES 1, 2, 3, and 4 operability. The purge exhaust radiation detectors (RE-2565A, B&C) are treated as one channel which is considered OPERABLE if the particulate (RE-2565A) and iodine (RE-2565B) monitors are OPERABLE or the noble gas monitor (RE-2565C) is OPERABLE. In addition, two individual channels of containment area low range gamma monitors (RE-0002 & RE-0003) are provided. The two required radiation monitoring channels may be made up of any combination of the above described channels.

(continued)

BASES

LCO (continued)

4. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements. The safety injection initiation function is applicable in MODES 1, 2, 3, and 4 only.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays, Containment Radiation, and Safety Injection Functions are required OPERABLE in MODES 1, 2, 3, and 4. Under these conditions, the potential exists for an accident that could release significant fission product radioactivity into containment. Therefore, the Containment ventilation isolation instrumentation must be OPERABLE in these MODES.

During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 70 hours) in containment, the air locks may be open provided they are isolable per LCO 3.9.4. Since the air locks can only be closed manually, it is assumed that containment ventilation isolation is accomplished by manually closing the purge and exhaust ventilation valves. Therefore, only OPERABLE radiation monitors are required to alert the operators of the need for containment ventilation isolation.

While in MODES 5 and 6 without fuel handling in progress, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Required Action A.1. If no radiation monitoring channels are operable or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place and maintain containment purge supply and exhaust isolation valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each penetration not in the required status. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable ~~during CORE ALTERATIONS and~~ during movement of recently irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.4

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. For MODES 1, 2, 3, and 4, this test verifies the capability of the instrumentation to provide the containment purge and exhaust system isolation. During ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel in containment, this test verifies the capability of the required channels to generate the signals required for input to the control room alarm. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

SR 3.3.6.5

SR 3.3.6.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay.

For slave relays and associated auxiliary relays in the CVI actuation system circuit that are Potter and Brumfield (P&B) type Motor Driven Relays (MDR), the SLAVE RELAY TEST is performed on an 18-month frequency. This test frequency is based on relay reliability assessments presented in WCAP-13878, “Reliability Assessment of Potter and Brumfield MDR Series Relays.” The reliability assessments are relay specific and apply only to Potter and Brumfield MDR series relays. Quarterly testing of the slave relays associated with non-P&B MDR auxiliary relays will be administratively controlled until an alternate method of testing the auxiliary relays is developed or until they are replaced by P&B MDR series relays.

SR 3.3.6.6

SR 3.3.6.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test

(continued)

BASES

REFERENCES 1. 10 CFR ~~100.11~~50.67.

Complete Replacement of the Existing 3.4.16 Bases

BASES

BACKGROUND

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.18, "Secondary Specific Activity."

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of 335), respectively. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or an RCS transient prior

APPLICABLE SAFETY ANALYSES (continued)

to the accident. In both cases, the noble gas specific activity is assumed to be 280 µCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis also assumes a loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 60.0 µCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The iodine specific activity in the reactor coolant is limited to 1.0 µCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 280 µCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

APPLICABILITY

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to

limit the potential consequences of a SGTR to within the SRP acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is ≤ 60.0 μCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

ACTIONS (continued)

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODES(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

C.1 and C.2

If the Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > 60.0 µCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in the noble gas specific activity. A Frequency Note limits this periodic measurement to MODE 1 when reactor power produces the fission products.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the low probability of a gross fuel failure during this time.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the noble gas specific activity levels peak during this time; samples at other times would provide inaccurate results.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. A Frequency Note limits this periodic measurement to MODE 1 when reactor power produces the fission products. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

REFERENCES

1. 10 CFR 50.67.

2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms."

3. FSAR, Section 15.1.5.

4. FSAR, Section 15.6.3.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis 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 gallon per minute or is assumed to increase to 1 gallon per minute 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 equal to the LCO 3.4.16, "RCS Specific Activity," limits. 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 (Ref. 2), 10 CFR ~~100~~ 50.67 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

LCO

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

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

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. Portions of the tube below 15.2 inches below the top of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.2 (continued)

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 plugging 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.

3. 10 CFR ~~100~~50.67.

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, “Pressurized Water Reactor Steam Generator Examination Guidelines.”

7. License Amendment Nos. 167 and 149, "Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Revision to Technical Specifications 5.5.9, "Steam Generator (SG) Program," and 5.6.10, "Steam Generator Tube Inspection Report," (TAC Nos. ME8313 and ME8314),” September 10, 2012.

BASES

BACKGROUND (continued)

A long term recirculation solution pH of 7.0 to 10.5 also serves to minimize the hydrogen produced by the corrosion of galvanized surfaces and zinc-based paints.

In addition, the determination of this pH range also considered the environmental qualification of equipment in containment that may be subjected to the containment spray.

In order to achieve the desired pH range of 7.0 to 10.5 in the post-LOCA recirculation solution, a total of between 11,484 pounds (220 ft^3) and 14,612 pounds (260 ft^3) of TSP is required. The three TSP storage baskets are designed and located to permit the TSP to be dissolved into the containment recirculation sump solution as the post-LOCA water level rises. The stainless steel mesh screen storage baskets are located in the containment sump area anchored to the filler slab at elevation 171-ft 9-in. The post-LOCA ECCS recirculation and containment spray provide mixing to achieve a uniform solution pH.

TSP, because of its stability when exposed to radiation and elevated temperature and its nontoxic nature, is the preferred buffer material. The dodecahydrate form of TSP is used because of the high humidity in the containment during normal operation. Since the TSP is hydrated, it will not absorb large amounts of water from the humid atmosphere and will be less susceptible to physical and chemical change than the anhydrous form of TSP.

APPLICABLE SAFETY ANALYSES

Following the assumed release of radioactive material from a DBA to the containment atmosphere, the containment is assumed to leak at its design value. The LOCA radiological dose analysis assumes the amount of radioactive material available for release to the outside atmosphere is reduced by the operation of the containment spray system. The analysis also assumes the long term pH control of the recirculation fluid retains the dissolved iodine in solution which prevents the iodine from becoming available for release to the atmosphere (Ref. 2). The radiological consequences of a LOCA may be increased if the long term pH of the recirculation solution is not adjusted to 7.0 or greater. Therefore, long term pH control of the post-LOCA recirculation fluid helps ensure the offsite and control room thyroid doses are within the limits of 10 CFR ~~100~~ 50.67 and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

containment was designed with an allowable leakage rate of 0.2% of containment air weight per day for the first 24 hours and 0.1% per day thereafter (Ref. 2). This leakage rate is defined as L_a = 0.2% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 37 psig following a DBA. This allowable leakage rate (0.2%) forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leaktight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. The pressure and temperature limitations of MODES 5 and 6 reduce the probability and consequences of the events events considered for MODES 1, 2, 3, and 4. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. In MODE 6, the limiting location for dose consequences resulting from a fuel handling accident is in the fuel building and not containment. As a result, no credit is taken for the closure of the airlocks.~~the requirements for~~

(continued)

BASES

APPLICABILITY (continued)

~~the containment air locks are based on a fuel handling accident inside containment.~~ The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door Is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through the outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

limiting with respect to the steam releases used in meeting equipment qualification criteria. The failure of an MSIV has no effect on the results of these events.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs. This is not a limiting scenario with respect to doses or with respect to the core response analyses.

d. For a steam generator tube rupture, closure of the MSIVs in the faulted loop isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO requires that two MSIV systems in each steam line be OPERABLE. The MSIV systems are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal. An OPERABLE MSIV system may consist of an OPERABLE MSIV and inoperable associated bypass valve provided the inoperable bypass valve is maintained closed.

This LCO provides assurance that the MSIV systems will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR ~~100~~ 50.67 (Ref. 6) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIV systems must be OPERABLE in MODE 1, and in MODES 2 and 3 except when one MSIV system in each steam line is closed, when there is significant mass and energy in the RCS and steam generators. When the MSIV systems are closed, they are already performing the safety function.

In MODE 4, normally most of the MSIV systems are closed, and the steam generator energy is low.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV system is within the limit given in Reference 9 on an actual or simulated actuation signal and is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the INSERVICE TESTING PROGRAM. This SR is normally performed upon returning the unit to operation following a refueling outage.

The Frequency is in accordance with the INSERVICE TESTING PROGRAM. Operating experience has shown that these components usually pass the Surveillance when performed in accordance with the INSERVICE TESTING PROGRAM. Therefore, the Frequency is acceptable from a reliability standpoint.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. If desired, this allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 6.2.
3. FSAR, Subsection 15.1.5.
4. FSAR, Subsection 15.4.9.
5. FSAR, Subsection 15.2.8.
6. 10 CFR ~~100.11~~50.67.
7. ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code).
8. Vogtle Electric Generating Plant, Units 1 and 2 - Issuance of Amendments Regarding Implementation of Topical Report Nuclear Energy Institute NEI 06-09, “Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines,” Revision 0-A (CAC Nos. ME9555 and ME9556).
9. TRM, Section 13.7.6.

BASES

APPLICABLE SAFETY ANALYSES (continued)

event and other accident analyses. After primary to secondary break flow termination, it is assumed that one ARV on an intact SG is used to cool the RCS down to 350°F, at the maximum allowable cooldown rate of 100°F/hour.

The offsite radiological dose analyses show that the failure open of the ARV on the ruptured SG represents the limiting single failure. The resulting offsite radiological doses at the exclusion area boundary, low population zone, and control room are well within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10 CFR ~~100~~50.67. A detailed description of the SGTR analyses can be found in WCAP-11731 and associated supplements (Ref. 3).

The ARVs are equipped with manual block valves in the event an ARV spuriously fails open or fails to close during use.

The ARVs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Three ARV lines are required to be OPERABLE. One ARV line is required from each of three steam generators to ensure that at least one ARV line is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second ARV line on an unaffected steam generator. A block valve for each required ARV must be OPERABLE to isolate a failed open ARV line.

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an SGTR event in which the condenser is unavailable for use with the Steam Dump System.

An ARV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand. Additionally, it is required that at least two of the three OPERABLE ARVs maintain the capability for local manual actuation via their associated handpumps.

APPLICABILITY

In MODES 1, 2, and 3, the ARVs are required to be OPERABLE.

In MODE 4, the pressure and temperature limitations are such that the probability of an SGTR event requiring ARV operation is low. In addition, the RHR system is available

(continued)

BASES

BACKGROUND (continued)

moisture removal. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level; however, the VEGP dose analysis assumes no heater operation and an iodine removal efficiency consistent with the iodine removal efficiency in Regulatory Guide 1.52 (Ref. 4) for systems designed to operate inside primary containment (i.e., no humidity control). Therefore, the heaters are not required for PPAFES OPERABILITY.

APPLICABLE SAFETY ANALYSES

The PPAFES design basis is established by the large break loss of coolant accident (LOCA). The system evaluation assumes 2 gpm continuous leakage and a 50 gpm leak for 30 minutes due to a passive failure during a Design Basis Accident (DBA). The system restricts the radioactive release to within the 10 CFR ~~100~~ 50.67 (Ref. 4) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR ~~100~~ 50.67 limits). The analysis of the effects and consequences of a large break LOCA are presented in Reference 3.

The PPAFES satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Two independent and redundant trains of the PPAFES are required to be OPERABLE to ensure that at least one train is available, assuming there is a single failure disabling the other train coincident with a loss of offsite power.

The PPAFES is considered OPERABLE when the individual components necessary to control radioactive releases are OPERABLE in both trains. A PPAFES train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Demister, ductwork, valves, and dampers are OPERABLE and air circulation can be maintained.

The LCO is modified by a Note allowing the PPAFES boundary to be opened intermittently under administrative controls without requiring entry into the Condition for an inoperable pressure boundary. For

(continued)

BASES

LCO (continued)

entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for PPAFES isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, the PPAFES is required to be OPERABLE, consistent with the OPERABILITY requirements of the ECCS.

In MODE 5 or 6, the PPAFES is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one PPAFES train inoperable, the action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the PPAFES function. The 7 day Completion Time is appropriate because the risk contribution of the PPAFES is less than that of the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of a DBA occurring during this period, and the remaining train providing the required capability.

B.1

If the PPAFES boundary is inoperable, the PPAFES trains cannot perform their intended function. Actions must be taken to restore an OPERABLE PPAFES boundary within 24 hours. During the period that the PPAFES boundary is inoperable, appropriate compensatory measures (consistent with the intent, as applicable, of GDC 19, 60, 64 and 10 CFR ~~100~~50.67) will be utilized to ensure the necessary physical security and to minimize the release of radioactive material to the atmosphere outside the building. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24-hour Completion Time is reasonable based on the low

(continued)

BASES (continued)

REFERENCES

1. FSAR, Subsection 6.5.1.

2. FSAR, Subsection 9.4.3.

3. FSAR, Subsection 15.6.5.

4. 10 CFR ~~100~~50.67.

5. WCAP-16294-NP-A, Rev. 1, “Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs,” June 2010.

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the FSAR, Subsection 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Subsection 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Subsection 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.~~195~~ 183 (Ref. 4). The resultant 2 hour ~~thyroid~~ dose per person at the exclusion area boundary is ~~a small fraction of the 10 CFR 100 (Ref. 5) limits~~less than the acceptance criteria specified in Regulatory Guide 1.183, Revision 0 (Reference 4).

~~According to Reference 4,~~If there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident, then~~. With 23 ft of water,~~ the ~~assumptions of~~decontamination factors specified in Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small non-conservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop. The analyses also assume a limited number of fuel rods are damaged in a second fuel bundle.

The fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1 (continued)

water level in the fuel storage pool must be checked periodically. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

During refueling operations, the level in the fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.

REFERENCES

1. FSAR, Subsection 9.1.2.

2. FSAR, Subsection 9.1.3.

3. FSAR, Subsection 15.7.4.

4. Regulatory Guide 1.~~195~~183 Revision 0, " Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors~~,~~ July~~May~~ 2003~~3~~.

~~5. 10 CFR 100.11.~~

B 3.7 PLANT SYSTEMS

B 3.7.16 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR ~~100~~ 50.67 (Ref. 1) limits.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR ~~100.11~~50.67.
2. FSAR, Chapter 15.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in all filled portions of the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. In MODES 1 and 2, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," ensure an adequate amount of negative reactivity is available to shut down the reactor. In MODES 3, 4, and 5, LCO 3.1.1, "SHUTDOWN MARGIN" ensures an adequate amount of negative reactivity is available to shut down the reactor.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the Reactor Coolant System. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

ACTIONS

A.1 ~~and A.2~~

Continuation of ~~CORE ALTERATIONS or~~ positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the filled portions of the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving ~~CORE ALTERATIONS or~~ positive reactivity additions must be suspended immediately.

(continued)

BASES

ACTIONS

A.1 ~~and A.2~~ (continued)

Suspension of ~~CORE ALTERATIONS and~~ positive reactivity additions shall not preclude moving a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.

A.~~3~~2

In addition to immediately suspending ~~CORE ALTERATIONS or~~ positive reactivity additions, boration to restore the concentration must be initiated immediately.

There are no safety analysis assumptions of boration flow rate and concentration that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in all filled portions of the RCS, and connected portions of the refueling canal and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.4. If any dilution has occurred while the cavity or canal were disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. FSAR, Subsection 15.4.6.

BASES

LCO (continued)

administrative control provided the reactor coolant system boron concentration is within the limit specified in the COLR and the high flux at shutdown alarm is OPERABLE. The high flux at shutdown alarm is not normally required OPERABLE in MODE 6, however for the purpose of meeting the requirement stated in this Note, the high flux at shutdown alarm is considered OPERABLE if the applicable surveillance requirements of LCO 3.3.8, High Flux at Shutdown Alarm and LCO 3.9.3, Nuclear Instrumentation are met.

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated.

ACTIONS

The ACTIONS do not apply to valves in the flow path from the RMWST, through the chemical mixing tank, to the suction of the charging pumps, when opened under administrative control in accordance with the Note in the LCO. The ACTIONS table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve.

~~A.1~~

~~Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.~~

~~Condition A has been modified by a Note to require that Required Action A.3 be completed whenever Condition A is entered.~~

(continued)

BASES

ACTIONS (continued)

A.~~2~~1

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valve(s) secured closed. Securing the valve(s) in the closed position ensures that the valve(s) cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

A.~~3~~2

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate that the required boron concentration exists. The Completion Time of 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

These valve(s) are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed through a system walkdown. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Subsection 15.4.6.

2. NUREG-0800, Section 15.4.6.

BASES

LCO (continued)

source of power, provided the detector for the opposite source range neutron flux monitor is powered from its normal source.

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, the operability requirements for the installed source range detectors and circuitry are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control. Suspending the movement of fuel, sources, and reactivity control components ensures that positive reactivity is not inadvertently added to the reactor core while the source range neutron flux monitor is inoperable. Required Action A.1 is modified by a Note that states that fuel assemblies, sources, and reactivity control components may be moved if necessary to facilitate repair or replacement of the inoperable source range neutron flux monitor. It may be necessary to move these items away from the locations in the core close to the source range neutron flux monitor to minimize personnel radiation dose during troubleshooting or repair. The Note also permits completion of movement of a component to a safe position, should the source range neutron flux monitor be discovered inoperable during component movement.

B.1

Condition B is modified by a Note to clarify the requirement that entry into or continued operation in accordance with Condition A is required for any entry into Condition B. The Note reinforces conventions of LCO applicability as stated in LCO 3.0.2 and as reflected in examples in 1.3, Completion Times.

(continued)

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, actions shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS (except as allowed by the Note to Required Action A.1) and positive reactivity additions are not to be

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the 10 CFR 50, Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the ~~requirements~~ limits of ~~10 CFR 100~~Regulatory Guide 1.183 (Reference 3). Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. If closed, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. Alternatively, the equipment hatch can be open provided it can be installed with a minimum of four bolts holding it in place.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is

(continued)

BASES

BACKGROUND (continued)

required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment, the door interlock mechanism may remain disabled, but one air lock door must always must be isolable by at least one air lock door with a designated individual available to close the air lock door, or at least one air lock door must be closed.

The emergency air lock will not normally be open during ~~core alterations or~~ fuel movement inside containment. Therefore, in the event the emergency air lock is open at the same time the personnel air lock is open, a separate individual shall be responsible for closing the emergency air lock (within 15 minutes) in addition to the individual designated to close the personnel air lock.

The requirements for containment penetration closure are sufficient to ensure fission product ~~radiactivity~~radioactivity release from containment due to a fuel handling accident involving handling of recently irradiated fuel during refueling is maintained to within the acceptance criteria of Standard Review Plan Section 15.7.4 and General Design Criteria 19.

The Containment Ventilation System consists of two 24 inch penetrations for purge and exhaust of the containment atmosphere. Each main or shutdown purge and exhaust system contains one motor operated 24 inch valve inside containment and one motor operated 24 inch valve outside containment (HV-2626A, HV-2627A, HV-2628A, and HV-2629A). A second 14 inch mini-purge and exhaust system shares each 24 inch penetration and consists of one 14 inch pneumatically operated valve inside containment and one outside of containment (HV-2626B, HV-2627B, HV-2628B, and HV-2629B). A 14 inch mini-purge line is connected to each 24 inch line between the 24 inch isolation valve and the penetration both inside and outside containment.

In MODES 1, 2, 3 and 4 the 24 inch main or shutdown purge and exhaust valves are secured in the closed position. The 14 inch mini-purge and exhaust valves may be opened in these MODES in accordance with LCO 3.6.3, Containment Isolation Valves, and are automatically closed by a Containment Ventilation Isolation signal. The instrumentation that provides the automatic isolation function for these valves is listed in LCO 3.3.6, Containment Ventilation Isolation Instrumentation.

(continued)

BASES

BACKGROUND (continued)

In MODE 6, the 24 inch main or shutdown purge and exhaust valves are used to exchange large volumes of containment air to support refueling operations or other maintenance activities. During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment any open 24 inch valves are capable of being closed (LCO 3.3.6). The 14 inch mini-purge and exhaust valves, though typically not opened during ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment, if opened are also capable of being closed (LCO 3.3.6).

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods allowed under the provisions of 10 CFR 50.59 may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during ~~CORE ALTERATIONS~~ or movement of recently irradiated fuel assemblies within containment (Ref. 1).

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to recently irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly onto another irradiated fuel assembly.

To support the plant configuration of both air lock doors open (personnel and/or emergency air locks), and to further minimize an unmonitored, untreated release, the designated individual for closure of the air lock will have the air lock closed within 15 minutes of the fuel handling accident. The 15 minute duration was chosen as the limit for the response capability for the person who is designated for closing the air lock door. The NRC

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

acceptance of this specification was based on doses for a 2 hour release as well as a licensee commitment for a person designated to close the door quickly.

The requirements of LCO 3.9.7, "Refueling Cavity Water Level," ~~and the~~in conjunction with minimum decay time of ~~90~~ 70 hours prior to ~~CORE ALTERATIONS~~ irradiated fuel movement with containment closure capability or a minimum decay time of 70 hours without containment closure capability ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are ~~well within the guideline values specified in 10 CFR 100~~less than the acceptance criteria specified in Regulatory Guide 1.183, Revision 0 (Reference 3). ~~The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values as specified in Regulatory Guide 1.195 (Ref. 3).~~ The radiological consequences of a fuel handling accident in containment ~~have been evaluated assuming that the containment is open to the outside atmosphere~~is non-limiting as compared to a fuel handling accident in the fuel building. ~~All airborne activity reaching the containment atmosphere is assumed to be exhausted to the environment within 2 hours of the accident. The calculated offsite and control room operator doses are within the acceptance criteria of Regulatory Guide 1.195 and GDC 19.~~ Therefore, although the containment penetrations do not satisfy any of the 10 CFR 50.36 (c)(2)(ii) criteria, LCO 3.9.4 provides containment closure capability to minimize potential offsite doses.

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires the equipment hatch, the air locks, and any penetration providing direct access to the outside atmosphere to be closed or capable of being closed. Personnel air lock closure capability is provided by the availability of at least one door and a designated individual to close it. Emergency air lock closure capability is provided by the availability of at least one door and a designated individual to close it. Equipment hatch closure capability is provided by a designated trained hatch closure crew and the necessary equipment. For the OPERABLE containment ventilation penetrations, this LCO ensures that each penetration is isolable by the Containment Ventilation Isolation valves. The OPERABILITY requirements for LCO 3.3.6, Containment Ventilation Isolation Instrumentation ensure that radiation monitor inputs to the control room alarm exist so that operators can take timely

(continued)

BASES

LCO (continued)

action to close containment penetrations to minimize potential offsite doses. The LCO requirements for penetration closure may also be met by the automatic isolation capability of the CVI system. Temporary non-1E power may be supplied to the air operated and/or solenoid operated CVI valves. The temporary non-1E power must be connected in such a way that it cannot affect the capability of the valves to close either automatically or manually from the control room handswitch.

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

Item b of this LCO includes requirements for both the emergency air lock and the personnel air lock. The personnel and emergency air locks are required by Item b of this LCO to be isolable by at least one air lock door in each air lock. Both containment personnel and emergency air lock doors may be open during movement of recently irradiated fuel in the containment ~~and during CORE ALTERATIONS~~ provided at least one air lock door is isolable in each air lock. An air lock is isolable when the following criteria are satisfied:

1. one air lock door is OPERABLE,

2. at least 23 feet of water shall be maintained over the top of the reactor vessel flange in accordance with Specification 3.9.7,

3. a designated individual is available to close the door.

OPERABILITY of a containment air lock door requires that the door seal protectors are easily removed, that no cables or hoses are being run through the air lock, and that the air lock door is capable of being quickly closed.

The equipment hatch is considered isolable when the following criteria are satisfied:

1. the necessary equipment required to close the hatch is available.

(continued)

BASES

LCO (continued)

2. at least 23 feet of water is maintained over the top of the reactor vessel flange in accordance with Specification 3.9.7,

3. a designated trained hatch closure crew is available.

Similar to the air locks, the equipment hatch opening must be capable of being cleared of any obstruction so that closure can be achieved as soon as possible.

APPLICABILITY

The containment penetration requirements are applicable during ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment because this is when there is a potential for ~~a~~ the limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment ~~are~~ is not being conducted, the potential for a fuel handling accident does not exist. Additionally, due to radioactive decay, a fuel handling accident involving handing recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 70 hours) will result in does that are well within the guideline values specified in 10 CFR 100 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 ~~and A.2~~

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the required open containment ventilation isolation valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each required

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.4.2

This Surveillance demonstrates that each containment ventilation isolation valve in each open containment ventilation penetration actuates to its isolation position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note stating that this surveillance is not required to be met for valves in isolated penetrations. LCO 3.9.4.c.1 provides the option to close penetrations in lieu of requiring automatic actuation capability.

SR 3.9.4.3

The equipment hatch is provided with a set of hardware, tools, and equipment for moving the hatch from its storage location and installing it in the opening. The required set of hardware, tools, and equipment shall be inspected to ensure that they can perform the required functions.

The 7 day frequency is adequate considering that the hardware, tools, and equipment are dedicated to the equipment hatch and not used for any other functions.

The SR is modified by a Note which only requires that the surveillance be met for an open equipment hatch. If the equipment hatch is installed in its opening, the availability of the means to install the hatch is not required.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.

2. FSAR, Subsection 15.7.4.

3. ~~Regulatory Guide 1.195, May 2003~~Regulatory Guide 1.183, Revision 0, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents At Nuclear Reactors,” July 2000.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Ref~~s~~. 1~~ and 2~~). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~< 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3~~less than the acceptance criteria in Reference 1.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.~~195~~ 183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water.

The fuel handling accident analysis inside containment is non-limiting compared to a fuel handling accident in the fuel building as described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of ~~90~~ 70 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained less than the acceptance criteria specified in Reference 1~~within allowable limits (Refs. 3 and 4)~~.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the acceptance criteria in Reference 1 ~~guidance of Reference 3~~.

APPLICABILITY

LCO 3.9.7 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and when moving irradiated fuel assemblies within containment. Unlatching and latching of control rod drive shafts includes drag testing of the associated rod cluster control assembly. The LCO ensures a sufficient level of water is present in the reactor cavity to minimize the radiological consequences of a fuel handling accident in containment. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.15, "Fuel Storage Pool Water Level."

ACTIONS

A.1 and A.2

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Regulatory Guide 1.~~195, May 2003~~183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. FSAR, Subsection 15.7.4.

~~3. 10 CFR 100.11~~

~~4. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-7828, Radiological Consequences of a Fuel Handling Accident, December 1971.~~

Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Alternative Source Term,
TSTF-51, TSTF-471, and TSTF-490

Attachment 4

Regulatory Guide 1.183 Conformance Tables

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
1.1.1	The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. The safety margins are products of specific values and limits contained in the technical specifications (which cannot be changed without NRC approval) and other values, such as assumed accident or transient initial conditions or assumed safety system response times. Changes, or the net effects of multiple changes, that result in a reduction in safety margins may require prior NRC approval. Once the initial AST implementation has been approved by the staff and has become part of the facility design basis, the licensee may use 10 CFR 50.59 and its supporting guidance in assessing safety margins related to subsequent facility modifications and changes to procedures.	Conforms- Adequate safety margins are maintained, as discussed in the No Significant Hazards Consideration. Future changes will be evaluated under the provisions of 10 CFR 50.59.
1.1.2	The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained to compensate for uncertainties in accident progression and analysis data. Consistency with the defense-in-depth philosophy is maintained if system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties. In all cases, compliance with the General Design Criteria in Appendix A to 10 CFR Part 50 is essential. Modifications proposed for the facility generally should not create a need for compensatory programmatic activities, such as reliance on manual operator actions.	Conforms – There are no facility modifications being proposed to implement AST, and compliance with the GDCs are maintained. No new reliance is placed on compensatory programmatic actions (including manual operator actions) to maintain adequate defense-in-depth.
1.1.2	Proposed modifications that seek to downgrade or remove required engineered safeguards equipment should be evaluated to be sure that the modification does not invalidate assumptions made in facility PRAs and does not adversely impact the facility's severe accident management program.	Not Applicable – There are no modifications being proposed with this License Amendment Request.
1.1.3	The design basis accident source term is a fundamental assumption upon which a significant portion of the facility design is based. Additionally, many aspects of facility operation derive from the design analyses that	Conforms – See RG Section 1.3 discussions.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	incorporated the earlier accident source term. Although a complete re-assessment of all facility radiological analyses would be desirable, the NRC staff determined that recalculation of all design analyses would generally not be necessary. Regulatory Position 1.3 of this guide provides guidance on which analyses need updating as part of the AST implementation submittal and which may need updating in the future as additional modifications are performed.	
1.1.3	This approach would create two tiers of analyses, those based on the previous source term and those based on an AST. The radiological acceptance criteria would also be different with some analyses based on whole body and thyroid criteria and some based on TEDE criteria. Full implementation of the AST revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. Selective implementation of the AST also revises the plant licensing basis and may establish the TEDE dose as the new acceptance criteria. Selective implementation differs from full implementation only in the scope of the change. In either case, the facility design bases should clearly indicate that the source term assumptions and radiological criteria in these affected analyses have been superseded and that future revisions of these analyses, if any, will use the updated approved assumptions and criteria.	Conforms – This is a full scope AST implementation for the radiological dose consequences of the VEGP Design Basis Accidents.
1.1.3	Radiological analyses generally should be based on assumptions and inputs that are consistent with corresponding data used in other design basis safety analyses, radiological and nonradiological, unless these data would result in nonconservative results or otherwise conflict with the guidance in this guide.	Conforms- This License Amendment Request includes re-evaluation of the radiological consequences of the most severe DBAs. It relies on assumptions and inputs that do not create a conflict with, or render non-conservative, other design basis safety analyses.
1.1.4	Although the AST provided in this guide was based on a limited spectrum of severe accidents, the particular characteristics have been tailored specifically for DBA	Conforms – No changes are proposed in this License

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	analysis use. The AST is not representative of the wide spectrum of possible events that make up the planning basis of emergency preparedness. Therefore, the AST is insufficient by itself as a basis for requesting relief from the emergency preparedness requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50.	Amendment Request to Emergency Preparedness requirements.
1.2.1	Full implementation is a modification of the facility design basis that addresses all characteristics of the AST, that is, composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. This applies not only to the analyses performed in the application (which may only include a subset of the plant analyses), but also to all future design basis analyses. At a minimum for full implementations, the DBA LOCA must be re-analyzed using the guidance in Appendix A of this guide. Additional guidance on analysis is provided in Regulatory Position 1.3 of this guide. Since the AST and TEDE criteria would become part of the facility design basis, new applications of the AST would not require prior NRC approval unless stipulated by 10 CFR 50.59, “Changes, Tests, and Experiments,” or unless the new application involved a change to a technical specification. However, a change from an approved AST to a different AST that is not approved for use at that facility would require a license amendment under 10 CFR 50.67.	Conforms – This License Amendment Request involves recalculation of the dose consequences of the most severe DBAs. The characteristics of the AST methods are addressed in the recalculations. The DBA LOCA has been re-analyzed per Appendix A.
1.2.2	Selective implementation is a modification of the facility design basis that (1) is based on one or more of the characteristics of the AST or (2) entails re-evaluation of a limited subset of the design basis radiological analyses. The NRC staff will allow licensees flexibility in technically justified selective implementations provided a clear, logical, and consistent design basis is maintained. An example of an application of selective implementation would be one in which a licensee desires to use the release timing insights of the AST to increase the required closure time for a containment isolation valve by a small amount. Another example would be a request to remove the charcoal filter media from the spent fuel building ventilation exhaust. For the latter, the licensee may only need to re-analyze DBAs that credited the iodine removal by the charcoal media. Additional analysis guidance is provided in Regulatory	Not Applicable – This License Amendment Request is for full scope AST implementation for the radiological dose consequences of the major VEGP DBA.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	Position 1.3 of this guide. NRC approval for the AST (and the TEDE dose criterion) will be limited to the particular selective implementation proposed by the licensee. The licensee would be able to make subsequent modifications to the facility and changes to procedures based on the selected AST characteristics incorporated into the design basis under the provisions of 10 CFR 50.59. However, use of other characteristics of an AST or use of TEDE criteria that are not part of the approved design basis, and changes to previously approved AST characteristics, would require prior staff approval under 10 CFR 50.67. As an example, a licensee with an implementation involving only timing, such as relaxed closure time on isolation valves, could not use 10 CFR 50.59 as a mechanism to implement a modification involving a reanalysis of the DBA LOCA. However, this licensee could extend use of the timing characteristic to adjust the closure time on isolation valves not included in the original approval.	
1.3.1	There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of design basis accidents. These requirements include, but are not limited to, the following. • Environmental Qualification of Equipment (10 CFR 50.49) • Control Room Habitability (GDC 19 of Appendix A to 10 CFR Part 50) • Emergency Response Facility Habitability (Paragraph IV.E.8 of Appendix E to 10 CFR Part 50) • Alternative Source Term (10 CFR 50.67) • Environmental Reports (10 CFR Part 51) • Facility Siting (10 CFR 100.11)[5] There may be additional applications of the accident source term identified in the technical specification bases and in various licensee commitments. These include, but are not limited to, the following from Reference 2, NUREG-0737. • Post-Accident Access Shielding (NUREG-0737, II.B.2) • Post-Accident Sampling Capability (NUREG-0737, II.B.3) • Accident Monitoring Instrumentation (NUREG-0737, II.F.1) • Leakage Control (NUREG-0737, III.D.1.1)	Conforms- This full scope AST License Amendment Request is salient to: a) Control Room Habitability (GDC 19 and NUREG-0737 Item III.D.3.4), b) AST (10 CFR 50.67), and c) Facility Siting (10 CFR 100.11). Control Room Habitability and compliance with the Alternative Source Term requirements are the principal subjects of this submittal and are discussed in Sections 3 and 4 of this License Amendment Request. Regarding Emergency Response Facility Habitability, VEGP will continue to meet the NUREG-0654 Planning Standard for Emergency Facilities

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	• Emergency Response Facilities (NUREG-0737, III.A.1.2) • Control Room Habitability (NUREG-0737, III.D.3.4)	and Equipment as described in the VEGP Emergency Plan. Design Basis dose calculations for non-control room Emergency Response Facilities, such as the Technical Support Center, are not part of the VEGP current licensing basis. The Emergency Response Facilities continue to meet NUREG-0696 habitability requirements. As stated in Footnote 5 of this RG, the dose guidelines of 10 CFR 100.11 are superseded by 10 CFR 50.67 for licensees that have implemented an AST.
1.3.2	Any implementation of an AST, full or selective, and any associated facility modification should be supported by evaluations of all significant radiological and nonradiological impacts of the proposed actions. This evaluation should consider the impact of the proposed changes on the facility's compliance with the regulations and commitments listed above as well as any other facility-specific requirements. These impacts may be due to (1) the associated facility modifications or (2) the differences in the AST characteristics. The scope and extent of the re-evaluation will necessarily be a function of the specific proposed facility modification[6] and whether a full or selective implementation is being pursued. The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn	Conforms- The License Amendment Request for this full scope application of the AST evaluated the impact of the proposed change against the Current Licensing Basis, mitigating system design basis requirements, and Technical Specifications. No facility modifications are proposed as part of this License Amendment Request and compliance with regulations and commitments are maintained.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	on those results, are no longer valid. Generic analyses, such as those performed by owner groups or vendor topical reports, may be used provided the licensee justifies the applicability of the generic conclusions to the specific facility and implementation. Sensitivity analyses, discussed below, may also be an option. If affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed. The license amendment request should describe the licensee's re-analysis effort and provide statements regarding the acceptability of the proposed implementation, including modifications, against each of the applicable analysis requirements and commitments identified in Regulatory Position 1.3.1 of this guide.	
1.3.2	The NRC staff has performed an evaluation of the impact of the AST on three representative operating reactors (Ref. 14). This evaluation determined that radiological analysis results based on the TID-14844 source term assumptions (Ref. 1) and the whole body and thyroid methodology generally bound the results from analyses based on the AST and TEDE methodology. Licensees may use the applicable conclusions of this evaluation in addressing the impact of the AST on design basis radiological analyses. However, this does not exempt the licensee from evaluating the remaining radiological and nonradiological impacts of the AST implementation and the impacts of the associated plant modifications. For example, a selective implementation based on the timing insights of the AST may change the required isolation time for the containment purge dampers from 2.5 seconds to 5.0 seconds. This application might be acceptable without dose calculations. However, evaluations may need to be performed regarding the ability of the damper to close against increased containment pressure or the ability of ductwork downstream of the dampers to withstand increased stresses.	Conforms- There are no plant modifications that are planned to implement the AST analyses. The radiological and nonradiological impacts of full scope implementation of the AST have been considered and discussed in the License Amendment Request, as applicable.
1.3.2	For full implementation, a complete DBA LOCA analysis as described in Appendix A of this guide should be performed, as a minimum. Other design basis analyses are updated in accordance with the guidance in this section.	Conforms – The DBA LOCA analysis is provided in this License Amendment Request which is consistent with Appendix A.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
1.3.2	A selective implementation of an AST and any associated facility modification based on the AST should evaluate all the radiological and nonradiological impacts of the proposed actions as they apply to the particular implementation. Design basis analyses are updated in accordance with the guidance in this section. There is no minimum requirement that a DBA LOCA analysis be performed. The analyses performed need to address all impacts of the proposed modification, the selected characteristics of the AST, and if dose calculations are performed, the TEDE criteria. For selective implementations based on the timing characteristic of the AST, e.g., change in the closure timing of a containment isolation valve, re-analysis of radiological calculations may not be necessary if the modified elapsed time remains a fraction (e.g., 25%) of the time between accident initiation and the onset of the gap release phase. Longer time delays may be considered on an individual basis. For longer time delays, evaluation of the radiological consequences and other impacts of the delay, such as blockage by debris in sump water, may be necessary. If affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed.	Not Applicable - This License Amendment Request is a full scope AST implementation that evaluates the dose consequences of the most severe VEGP DBAs.
1.3.3	It may be possible to demonstrate by sensitivity or scoping evaluations that existing analyses have sufficient margin and need not be recalculated. As used in this guide, a sensitivity analysis is an evaluation that considers how the overall results vary as an input parameter (in this case, AST characteristics) is varied. A scoping analysis is a brief evaluation that uses conservative, simple methods to show that the results of the analysis bound those obtainable from a more complete treatment. Sensitivity analyses are particularly applicable to suites of calculations that address diverse components or plant areas but are otherwise largely based on generic assumptions and inputs. Such cases might include post accident vital area access dose calculations, shielding calculations, and equipment environmental qualification (integrated dose). It may be possible to identify a bounding case, re-analyze that case, and use the results to draw conclusions regarding the remainder of the analyses. It may also be possible to show that for some analyses the whole body and thyroid doses determined with the previous source term	Not Applicable- The VEGP AST analysis does not rely on sensitivity or scoping analyses.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	would bound the TEDE obtained using the AST. Where present, arbitrary "designer margins" may be adequate to bound any impact of the AST and TEDE criteria. If sensitivity or scoping analyses are used, the license amendment request should include a discussion of the analyses performed and the conclusions drawn. Scoping or sensitivity analyses should not constitute a significant part of the evaluations for the design basis exclusion area boundary (EAB), low population zone (LPZ), or control room dose.	
1.3.4	Full implementation of the AST replaces the previous accident source term with the approved AST and the TEDE criteria for all design basis radiological analyses. The implementation may have been supported in part by sensitivity or scoping analyses that concluded many of the design basis radiological analyses would remain bounding for the AST and the TEDE criteria and would not require updating. After the implementation is complete, there may be a subsequent need (e.g., a planned facility modification) to revise these analyses or to perform new analyses. For these recalculations, the NRC staff expects that all characteristics of the AST and the TEDE criteria incorporated into the design basis will be addressed in all affected analyses on an individual as-needed basis. Re-evaluation using the previously approved source term may not be appropriate. Since the AST and the TEDE criteria are part of the approved design basis for the facility, use of the AST and TEDE criteria in new applications at the facility do not constitute a change in analysis methodology that would require NRC approval.[7]	Not Applicable- The VEGP AST design basis radiological analyses do not rely on sensitivity or scoping analyses.
1.3.4	This guidance is also applicable to selective implementations to the extent that the affected analyses are within the scope of the approved implementation as described in the facility design basis. In these cases, the characteristics of the AST and TEDE criteria identified in the facility design basis need to be considered in updating the analyses. Use of other characteristics of the AST or TEDE criteria that are not part of the approved design basis, and changes to previously approved AST characteristics, requires prior NRC staff approval under 10 CFR 50.67.	Not Applicable – This is a full scope License Amendment Request that evaluates the dose consequences of the most severe VEGP DBAs.
1.3.5	Current environmental qualification (EQ) analyses may be impacted by a proposed plant modification associated with the AST implementation. The EQ	Conforms – The VEGP AST License Amendment Request is

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	analyses that have assumptions or inputs affected by the plant modification should be updated to address these impacts. 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. The EQ dose estimates should be calculated using the design basis survivability period.	not proposing to modify the equipment qualification design basis to adopt AST. The VEGP EQ analysis will continue to be based on TID-14844 assumptions.
1.4	The use of an AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents. The AST has no direct effect on the probability of the accident. Use of an AST alone cannot increase the core damage frequency (CDF) or the large early release frequency (LERF). However, facility modifications made possible by the AST could have an impact on risk. If the proposed implementation of the AST involves changes to the facility design that would invalidate assumptions made in the facility's PRA, the impact on the existing PRAs should be evaluated.	Not Applicable - No facility modifications are proposed or planned as implementation actions of the FHA AST analysis.
1.4	Consideration should be given to the risk impact of proposed implementations that seek to remove or downgrade the performance of previously required engineered safeguards equipment on the basis of the reduced postulated doses. The NRC staff may request risk information if there is a reason to question adequate protection of public health and safety.	Not Applicable- The VEGP AST License Amendment Request is not seeking to remove or downgrade the performance of previously required engineered safeguards equipment on the basis of the reduced postulated doses.
1.4	The licensee may elect to use risk insights in support of proposed changes to the design basis that are not addressed in currently approved NRC staff positions. For guidance, refer to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Ref. 15).	Not Applicable- The VEGP AST License Amendment Request is not utilizing risk insights as a basis for any proposed changes.
1.5	According to 10 CFR 50.90, an application for an amendment must fully describe the changes desired and should follow, as far as applicable, the form	Conforms- The License Amendment Request is formatted in

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	prescribed for original applications. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (Ref 16), provides additional guidance. The NRC staff's finding that the amendment may be approved must be based on the licensee's analyses, since it is these analyses that will become part of the design basis of the facility. The amendment request should describe the licensee's analyses of the radiological and nonradiological impacts of the proposed modification in sufficient detail to support review by the NRC staff. The staff recommends that licensees submit affected FSAR pages annotated with changes that reflect the revised analyses or submit the actual calculation documentation.	accordance with accepted NRC/industry guidance. The request describes the radiological and nonradiological impacts of the VEGP AST analysis. Consistent with previous precedent, affected FSAR pages are not included in the analyses. However, a detailed summary of the AST dose calculations are included. Approval of this License Amendment Request will result in the necessary revisions to the FSAR, with revised FSAR pages submitted pursuant to 10 CFR 50.71(e).
1.5	If the licensee has used a current approved version of an NRC-sponsored computer code, the NRC staff review can be made more efficient if the licensee identifies the code used and submits the inputs that the licensee used in the calculations made with that code. In many cases, this will reduce the need for NRC staff confirmatory analyses. This recommendation does not constitute a requirement that the licensee use NRC-sponsored computer codes.	DBA dose analyses contained in this LAR submittal utilize Serco's version of RADTRAD Version 3.10.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
1.6	Requirements for updating the facility's final safety analysis report (FSAR) are in 10 CFR 50.71, "Maintenance of Records, Making of Reports." The regulations in 10 CFR 50.71(e) require that the FSAR be updated to include all changes made in the facility or procedures described in the FSAR and all safety evaluations performed by the licensee in support of requests for license amendments or in support of conclusions that changes did not involve unreviewed safety questions. The analyses required by 10 CFR 50.67 are subject to this requirement. The affected radiological analysis descriptions in the FSAR should be updated to reflect the replacement of the design basis source term by the AST. The analysis descriptions should contain sufficient detail to identify the methodologies used, significant assumptions and inputs, and numeric results. Regulatory Guide 1.70 (Ref. 16) provides additional guidance. The descriptions of superseded analyses should be removed from the FSAR in the interest of maintaining a clear design basis.	Conforms- Approval of this License Amendment Request will result in the necessary revisions to the FSAR, with revised FSAR pages submitted pursuant to 10 CFR 50.71(e).
2.1	The AST must be based on major accidents, hypothesized for the purposes of design analyses or consideration of possible accidental events, that could result in hazards not exceeded by those from other accidents considered credible. The AST must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.	Conforms- This License Amendment Request applies the AST methods when evaluating the dose consequences of the most severe DBAs applicable to VEGP.
2.2	The AST must be expressed in terms of times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.	Conforms – For the DBAs that release to Containment (LOCA and Control Rod Ejection), the AST is expressed in terms of times and rates of release of radioactive fission products, the types and quantities of the radioactive species released, and the chemical forms of iodine released.
2.3	The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered.	Conforms- This License Amendment Request considers a number of release scenarios, as applicable, for the

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.	DBAs being revised for use of AST. The most limiting of these releases are analyzed for radiological consequences.
2.4	The AST must have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and be documented in a scrutable form that facilitates public review and discourse.	Conforms- The DBA AST dose calculations have been developed based on NUREG-1465 and this Regulatory Guide. The calculations, which utilize RADTRAD were developed in accordance with 10 CFR 50 Appendix B, Criterion III.
2.5	The AST must be peer-reviewed by appropriately qualified subject matter experts. The peer-review comments and their resolution should be part of the documentation supporting the AST.	Conforms- The VEGP AST dose calculations have been developed and independently reviewed by internal experts at SNC. SNC internal experts are qualified via SNC internal processes and procedures in the performance of dose analyses. The calculations were developed in accordance with 10 CFR 50 Appendix B program, Criterion III.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
3.1	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. 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. 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.	Conforms –The VEGP DBAs that release to the Containment are the LOCA, FHA, and Control Rod Ejection. Core Inventory has been determined using an appropriate isotope generation and depletion code, such as ORIGEN2 or ORIGEN-ARP.
3.1	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.	With the exception of DBAs where cladding damage is postulated with a gap release, the analyses of events which involve fuel damage assume that the entire core is affected with a source term based upon full power, core average conditions. The source term for DBAs where cladding damage is postulated with a gap release is derived from the core source term, the number of damaged fuel rods, and a conservative assembly peaking factor, which exceeds the maximum fuel rod peaking factor specified in the COLR.
3.1	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,	The analysis of the FHA considers radioactive decay between the time of core shutdown and the

<table>
<tr><th colspan="3">Table A: Conformance With Regulatory Guide 1.183 Section C</th></tr>
<tr><th>RG Section</th><th>RG Position</th><th>VEGP Analysis</th></tr>
<tr><td></td><td>e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.</td><td>beginning of fuel movement.</td></tr>
<tr><td>3.2</td><td>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 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

Table 2
PWR Core Inventory Fraction
Released Into Containment
<table><tr><th>Group</th><th>Gap Release Phase</th><th>Early In-vessel Phase</th><th>Total</th></tr><tr><td>Noble Gases</td><td>0.05</td><td>0.95</td><td>1.0</td></tr><tr><td>Halogens</td><td>0.05</td><td>0.35</td><td>0.4</td></tr><tr><td>Alkali Metals</td><td>0.05</td><td>0.25</td><td>0.3</td></tr><tr><td>Tellurium Metals</td><td>0.00</td><td>0.05</td><td>0.05</td></tr><tr><td>Ba, Sr</td><td>0.00</td><td>0.02</td><td>0.02</td></tr><tr><td>Noble Metals</td><td>0.00</td><td>0.0025</td><td>0.0025</td></tr><tr><td>Cerium Group</td><td>0.00</td><td>0.0005</td><td>0.0005</td></tr><tr><td>Lanthanides</td><td>0.00</td><td>0.0002</td><td>0.0002</td></tr></table></td><td>Conforms – The LOCA AST calculation models Table 2 in the release fraction and timing file.</td></tr>
<tr><td>3.2</td><td>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.[11] Non-LOCA Fraction of Fission Product Inventory in Gap

Table 3

<table><tr><th>Group</th><th>Fraction</th></tr><tr><td>I-131</td><td>0.08</td></tr><tr><td>Kr-85</td><td>0.10</td></tr><tr><td>Other Noble Gases</td><td>0.05</td></tr><tr><td>Other Halogens</td><td>0.05</td></tr><tr><td>Alkali Metals</td><td>0.12</td></tr></table></td><td>Conforms – The FHA, Control Rod Ejection, and Locked Rotor accidents result in fuel damage, so the non-LOCA gap fractions of Table 3 are used. While the SGTR and MSLB accidents conservatively assume a pre-existing 1% leaking fuel source term for everything except noble gases and Iodines, this is not the result of damage caused by the accident, and so the non-LOCA gap fractions of Table 3 are not included for these events.</td></tr>
</table>

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. 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.12 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 (PWR)** **Phase** / **Onset** / **Duration** Gap Release / 30 sec / 0.5 hr Early In-vessel / 0.5 hr / 1.3 hr	Conforms – The LOCA AST calculation models Table 4 in the release fraction and timing file.
3.3	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.	Conforms – The LOCA AST calculation models Table 4 in the release fraction and timing file.
3.4	Elements listed in Table 5 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 Lanthenides / La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am Cerium / Ce, Pu, Np	Conforms The source term in the design basis analysis represents the most dose significant isotopes from the elements listed in Table 5 of Regulatory Guide 1.183.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
3.5	Of the radioiodine 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 - The chemical composition of the iodine released from the RCS to containment in the LOCA event is 95% aerosol, 4.85% elemental, and 0.15% organic. All non iodine and non-noble gas fission products are assumed to be in particulate form. The chemical composition of iodine species in the non-LOCA events are based upon the guidance in the respective appendices of Regulatory Guide 1.183.
3.6	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.	Conforms - The amount of fuel damage in the Locked Rotor event is based upon the fraction of the core which experiences DNB as reported in the Updated Final Safety Analysis Report (FSAR). The fraction of the fuel rods assumed to melt in the Control Rod Ejection event is conservatively based upon the portion of the fuel centerline that is calculated to exceed the melting temperature as documented in the FSAR.
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	Conforms – The AST dose consequences are calculated in TEDE.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.[13]	
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 - Dose Conversion Factors for inhalation in this analysis are taken from Table 2.1 of Federal Guidance Report 11.
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5 x 10-4 cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8 x 10-4 cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3 x 10-4 cubic meters per second.	Conforms - Offsite breathing rates used in the analysis are consistent with the values specified in Section 4.1.3 of Regulatory Guide 1.183.
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 - Dose Conversion Factors for air submergence are taken from the Table III.1 of Federal Guidance Report 12.
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	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

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	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).	performing a 'sliding' sum over 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 is determined for the most limiting person at the LPZ.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms - No correction is made for deposition of the effluent plume by deposition on the ground.
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 analyses consider the applicable sources of contamination to the control room atmosphere for each event. With respect to external and containment shine sources and their impact on control room doses, the physical design of the control room envelope and the surrounding auxiliary building provide more than 18" of concrete shielding between the operators and shine sources in all directions around the control room. The Control Room Emergency Filtration System filters are located outside of and above the control room envelope. The control room ceiling is approximately 18" thick. Accordingly, shielding

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
		from the walls and the filter unit casings prevents an appreciable dose to the operators during the accident.
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 SNC AST dose calculations use the same source term, transport, and release assumptions for Control Room, EAB, and LPZ dose values.
4.2.3	The models used to transport radioactive material into and through the control room,[15] 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.	Conforms - The models used to transport radioactive material into and through the control room have been structured 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	Conforms – For the AST DBAs covered under this License Amendment Request, credit is taken for control room isolation and reconfiguring into the emergency ventilation mode upon accident initiation by a high radiation or Safety Injection signal, where appropriate.

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	effects of different radionuclide accident isotopic mixes on monitor response.	
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- No credit is 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.16 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.	Conforms – Control room occupancy and breathing rates are consistent with this regulatory position.
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). $DDE_{finite} = \frac{DDE_{\varpi} V^{0.338}}{1173}$	Conforms - Control room doses are calculated using dose conversion factors identified in Position 4.1 above.
4.3	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.	Not Applicable – This full scope AST implementation LAR is for the radiological consequences of major VEGP DBAs.
4.4	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	The EAB and LPZ acceptance criteria from Table 6 of RG 1.183 are applied. The control room acceptance of 5 rem

<table>
<tr><th colspan="3">Table A: Conformance With Regulatory Guide 1.183 Section C</th></tr>
<tr><th>RG Section</th><th>RG Position</th><th>VEGP Analysis</th></tr>
<tr><td></td><td>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.

Table 6
Accident Dose Criteria

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

</td><td>TEDE is taken from 10 CFR 50.67(b)(2)(iii).</td></tr>
</table>

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
	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.	
4.4	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 EAB and LPZ acceptance criteria from Table 6 of RG 1.183 are applied. The control room occupant acceptance criteria of 5 rem TEDE is taken from 10 CFR 50.67(b)(2)(iii).
5.1.1	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.	Conforms- The VEGP AST dose calculations were prepared, reviewed, and maintained, by SNC under a 10 CFR 50 Appendix B Quality Assurance program.
5.1.1	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.	Not Applicable- This License Amendment Request is not proposing deviations to conformance with this Regulatory Guide.
5.1.2	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	Conforms - Only safety-related Engineered Safety Features are credited in the analysis

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	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.	with an assumed single active failure that results in the greatest impact on the radiological consequences. A loss of offsite power is assumed concurrent with the start of each event as that maximizes the dose impact.
5.1.3	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 nonconservative 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 nonconservative 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.[18] 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.	Conforms - Numerical values are selected and biased for each application in a conservative direction with the objective of maximizing the dose consequences. Numerical values for parameters which are controlled by Technical Specifications are either used as direct inputs in the analysis, or more conservative values may be used to enhance safety margin.
5.1.4	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	Conforms- The VEGP DBA analysis assumptions and methods are compatible with the AST and the TEDE criteria.

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	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.	
5.2	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.	Conforms – See Tables B, C, D, E, F, and G of this Enclosure.
5.2	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.	Conforms – See Tables B, C, D, E, F, and G of this Enclosure.

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5.2	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- PRA was not used as a basis for acceptability of this AST License Amendment Request.
5.3	Atmospheric dispersion values (X/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 X/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).	Conforms – The X/Q used for the EAB and the LPZ were previously approved by the NRC.
5.3	References 22 and 28 should be used if the FSAR X/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[19] (Ref. 26) is generally acceptable to the NRC staff for use in determining control room X/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 X/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in X/Q analysis methodology should be reviewed by the NRC staff.	Conforms – The onsite X/Q values are either those already used in the FSAR or were developed based on the approved meteorological data. ARCON96 (Reference 26) was used to generate revised onsite dose-receptor pairs.
6.0	The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological	Conforms – VEGP is retaining the use of the

Table A: Conformance With Regulatory Guide 1.183 Section C		
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	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.	TID 14844 source term as the basis for Environmental Qualification.
Footnote 6	For example, a proposed modification to change the timing of a containment isolation valve from 2.5 seconds to 5.0 seconds might be acceptable without any dose calculations. However, a proposed modification that would delay containment spray actuation could involve recalculation of DBA LOCA doses, re-assessment of the containment pressure and temperature transient, recalculation of sump pH, re-assessment of the emergency diesel generator loading sequence, integrated doses to equipment in the containment, and more.	Conforms – No modifications are being proposed as part of this AST License Amendment Request.
Footnote 7	In performing screenings and evaluations pursuant to 10 CFR 50.59, it may be necessary to compare dose results expressed in terms of whole body and thyroid with new results expressed in terms of TEDE. In these cases, the previous thyroid dose should be multiplied by 0.03 and the product added to the whole body dose. The result is then compared to the TEDE result in the screenings and evaluations. This change in dose methodology is not considered a change in the method of evaluation if the licensee was previously authorized to use an AST and the TEDE criteria under 10 CFR 50.67.	Not Applicable – This activity is a License Amendment Request made pursuant to 10 CFR Part 90.
Footnote 8	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.	Conforms – A 1.02 uncertainty factor is used for those events

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		resulting in fuel damage.
Footnote 9	Note that for 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.	Conforms – A conservative core factor is applied to the principal radionuclides to account for cycle-to-cycle variations.
Footnote 10	The 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. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.	Conforms – Burnup does not exceed 62,000 MWD/MTU at VEGP.
Footnote 11	The 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 the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.	Exception – SNC has requested an exception for the LHGR limit to be 7.5 kW/ft for 40% of the rods in an assembly. Only the FHA is impacted by this request. The gap fractions used for the FHA are consistent with those contained in PNNL-18212 Rev. 1 Table 2.9.
Footnote 12	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.	Conforms –RADTRAD can model the release either in a linear ramp manner, or instantaneous release, as required.
Footnote 13	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 – Offsite inhalation doses are calculated consistent with the definition of TEDE.
Footnote 14	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.	Not Applicable – This activity is a License Amendment Request made pursuant to 10 CFR Part 90.
Footnote 15	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	Conforms – The iodine protection factor methodology of

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	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.	Reference 22 is not used in this application.
Footnote 16	This occupancy is modeled in the X/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 – The control room occupancy assumptions are incorporated in the dose calculations
Footnote 17	For PWRs with steam generator alternative repair criteria, different dose criteria may apply to steam generator tube rupture and main steam line break analyses.	Conforms – Refer to ARC line items in Tables D and E.
Footnote 18	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 – Filter efficiencies for the PPAFES, the Control Room (CR) Pressurization Intake Filters, and the Control Room Recirculation Filters are developed from Technical Specification Surveillance requirements, with margin added for filter inefficiency and bypass leakage around the filter (in accordance with the prior CLB analyses of this type: double the penetration allowed by TS 5.5.11 and further reduce efficiency by a 0.5% bypass amount). This methodology assures compliance with Technical Specification 5.5.11 requirements and US NRC RG-1.52.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	VEGP Analysis
Footnote 19	The ARCON96 computer code contains processing options that may yield X/Q values that are not sufficiently conservative for use in accident consequence assessments or may be incompatible with release point and ventilation intake configurations at particular sites. The applicability of these options and associated input parameters should be evaluated on a case-by-case basis. The assumptions made in the examples in the ARCON96 documentation are illustrative only and do not imply NRC staff acceptance of the methods or data used in the example.	Conforms – The ARCON96 processing options and input parameters were based on the release point and ventilation intake configurations at VEGP.

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	VEGP Analysis
A-1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms – See discussions in Table A.
A-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 pH of the containment sump is maintained equal to or greater than 7.0 after the onset of the spray recirculation mode. Therefore, the radioiodine composition of 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide is used. The containment sump pH has been evaluated for the impact of the alternate source term and confirms that the sump pH remains greater than 7.0. In addition, VEGP uses trisodium phosphate to create a buffered sump solution that is resistant to change in pH.
A-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 or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms – The radioactivity released from the fuel is modeled as mixing instantaneously and homogeneously in the Containment.
A-3.2	Reduction in airborne radioactivity in the containment by natural deposition within the	Conforms - An aerosol natural deposition rate of

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	VEGP Analysis
	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.	0.1 hr^{-1} is assumed based upon values presented Section VI of NUREG/CR-6189.
A-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).	Conforms – Containment Spray is credited for elemental iodine and aerosol removal.
A-3.3	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.	Conforms – Containment Spray covers less than 90% of the Containment volume, so the modeling includes both the sprayed volume and unsprayed volume. A flow rate of 21,000 cfm is used between the sprayed and unsprayed volume which correlates to two turnovers of the unsprayed region per hour.
A-3.3	The SRP sets forth a maximum decontamination factor (DF) for elemental	Conforms - Elemental and aerosol removal

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	VEGP Analysis
	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).	coefficients are calculated for the sprayed regions of the containment using the guidelines of Chapter 6.5.2 of the Standard Review Plan. The elemental iodine removal coefficients are limited to a maximum value of 13.7/hr, and are set to zero when the elemental iodine decontamination factor (DF) reaches a value of 200. The aerosol removal coefficients are reduced by a factor of 10 when the aerosol DF reaches 50.
A-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 – No credit is taken for in-containment recirculation filter systems.
A-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.	Not Applicable - VEGP is a PWR.

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	VEGP Analysis
A-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).	Conforms – No credit is taken for ice condensers or other engineering safety features to reduce airborne radioactivity in containment.
A-3.7	The primary containment (i.e., drywell for Mark I and II containment designs) 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. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications.	Conforms – The containment leak rate for the first 24 hours is the maximum value allowed by the VEGP Technical Specifications. It is reduced to 50% of that value after 24 hours. An additional 5% margin is applied to the leak rate values for conservatism.
A-3.7	For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Not Applicable. VEGP is a PWR.
A-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	Conforms - Based upon the isolation of the mini-purge flow within 30 seconds, the mini-purge system will be isolated before the onset of the gap release as defined in Table 4 of this Regulatory Guide. Therefore, only

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	VEGP Analysis
	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.	those nuclides in the RCS source term are available for release.
A-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.	Not Applicable. VEGP does not have a dual containment.
A-5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or 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 nonconservative 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 is assumed to instantaneously and homogeneously mix in the primary sump water.
A-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.	The VEGP Technical Specifications do not provide a specific limit for operational leakage from ECCS systems. However, administrative limits ensure that the operational leakage outside of containment from ECCS systems is limited to no more than 2

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	VEGP Analysis
	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.	gpm total, which is multiplied by two, consistent with this Regulatory Position. In addition, two times the assumed leak rate of 7.0 gpm past valves that isolate return flow to the Refueling Water Storage Tank (RWST) is evaluated separately. The leakage is assumed to start at the earliest time that recirculation occurs in the ECCS system and continues for the 30-day duration of the event.
A-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 is modeled as being retained in the liquid phase.
A-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, 212ºF); and h_{fg} is the heat of vaporization at 212ºF.	Conforms - It is assumed for the case when the temperature of the ECCS leakage exceeds 212^0 F that the fraction of total iodine in the liquid that becomes airborne is equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, is determined assuming a constant enthalpy, h, process, and is based on the maximum time-dependent sump water temperature.

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	VEGP Analysis
A-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 - Since the calculated flashing fraction is less than 10%, and without a basis for justifying a smaller value, 10% of the iodine in the ECCS leakage is assumed to be released.
A-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 is modeled as 97% elemental and 3% organic.
A-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.	Not Applicable. VEGP is a PWR.
A-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	Conforms – VEGP uses hydrogen recombiners for post-accident hydrogen control. As such, the containment mini-purge system is assumed to not be available for combustible gas management and this pathway is assumed to remain closed following a containment isolation signal.

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	VEGP Analysis
	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).	
Footnote A-1	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. For BWRs, the simplified model should be used only if the release from the core is not directed through the suppression pool. Iodine removal in the suppression pool affects the iodine species assumed by the model to be present initially.	Conforms - The removal rate constants selected for use in the LOCA calculation are those that will maximize the dose consequences.

Table C: Conformance With Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	RG Position	VEGP Analysis
B-1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms – See discussions in Table A.
B-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 - The FHA is a single fuel assembly dropped from within either the Containment or the Fuel Handling Building, and consistent with CLB hits another assembly damaging 50 rods in the ”target” assembly. The number of fuel rods damaged is equal to one fuel assembly plus the 50 damaged rods in the target assembly (264+50=314).
B-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.	Exception - The fission product release is equal to the gap release, with isotopic fractions as given in Table 2.9 of PNNL-18212 Rev. 1. This is conservative compared to RG-1.183. Cycle to cycle fuel load variations are accounted for with adjustments to the core source term: +10% FDM.
B-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 – The chemical forms of radioiodine released from the fuel to the spent fuel pool is assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel completely dissociates in the pool water and re-evolves as elemental iodine. The dissociation and re-evolution occurs instantaneously.
B-2	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,	Conforms – Water level is greater than 23 feet. Therefore, the pool water is

Table C: Conformance With Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	RG Position	VEGP Analysis
	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).	assumed to have an overall decontamination factor of 200 for the iodine isotopes released from the gap.
B-3	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).	Conforms – Noble gases are not scrubbed by the pool water (decontamination factor of 1). Particulate releases are assumed to be entirely scrubbed (infinite decontamination factor).
B-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 – For releases in the Fuel Handling Building, the VEGP fuel handling analysis considers a release to the environment over a 2-hour time period.
B-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[1] should be determined and accounted for in the radioactivity release analyses.	Conforms – no filtration is modeled in the FHA analysis.
B-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 for mixing or dilution is modeled.

Table C: Conformance With Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	RG Position	VEGP Analysis
B-5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable – Containment is not assumed to be isolated during fuel handling operations.
B-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.	Not Applicable –A 2 hour release from the Fuel Building bounds a FHA in containment assuming all airlocks are open.
B-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.	Not Applicable – The FHA radiological release is over a two-hour period, but the release from the Fuel Building is closer than a release from the containment and thus the dispersion factors for the Fuel Building release bound that for containment.
B-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.	Not Applicable – No credit is taken for ESF filter systems to mitigate radioactive material release from the Containment.
B-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.	Not Applicable – see response to B-5.4

Table C: Conformance With Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	RG Position	VEGP Analysis
Footnote B-1	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.	Conforms – The FHA calculation assumes a 600 s delay for concentration levels to reach the CREFS isolation setpoint, an additional 8 s delay for CREFS components to close and an additional 90 seconds for pressurization to occur. This results in a total delay time of 698 s.
Footnote B-2	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.	Not Applicable – Containment is not assumed to be isolated during fuel handling operations.
Footnote B-3	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.	Not Applicable – No credit is taken for containment isolation.

Table D: Conformance With Regulatory Guide 1.183 Appendix E (Main Steam Line Break Accident)		
RG Section	RG Position	VEGP Analysis
E-1	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 – See discussions in Table A.
E-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.	Consistent with the VEGP current licensing basis a leaking fuel term is conservatively included with the two cases of iodine spiking.
E-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 Main Steam Line Break Accident dose calculation includes a case for a preaccident iodine spike with the maximum iodine concentration permitted by the VEGP technical specifications.
E-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 Main Steam Line Break Accident dose calculation includes a case for a concurrent iodine spike causing the iodine release rate from the fuel rods to the RCS to increase to a value 500 times greater than the release rate that yields the maximum equilibrium iodine concentration specified in the technical specifications. The iodine spike duration is 8 hours.

Table D: Conformance With Regulatory Guide 1.183 Appendix E (Main Steam Line Break Accident)		
RG Section	RG Position	VEGP Analysis
E-3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms – The initial activity from the fuel is assumed to be released instantaneously and homogeneously to the reactor coolant system.
E-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 – The iodine releases from the steam generators to the environment are 97% elemental and 3% organic for the pre-accident case and the concurrent iodine spike case.
E-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 –The assumed primary-to-secondary leak rate in the three intact steam generators are 0.65 gpm (936 gallons per day). This is conservative relative to VEGP TS 3.4.13 which allows 150 gallons per day per Steam Generator.
E-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 assumed density is 62.4 lbm/ft^3.
E-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 – For the faulted steam generator, primary-to-secondary leakage continues for the duration of the event. The release from the unaffected steam generators continues until the RHR system is placed in service in 20 hours.

Table D: Conformance With Regulatory Guide 1.183 Appendix E (Main Steam Line Break Accident)		
RG Section	RG Position	VEGP Analysis
E-5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms – All noble gases are released from the steam generator water without credit for scrubbing.
E-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:	Conforms – See below.
E-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 – The leakage of the faulted steam generator is modeled as a direct vapor flow from the RCS to the environment without partitioning. For the intact steam generators, primary-to-secondary leakage mixes with the secondary water without flashing for the duration of the event.
E-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 - For conservatism, no credit is taken for scrubbing.
E-5.5.3	The leakage that does not immediately flash is assumed to mix with the bulk water.	Conforms – The leakage that does not immediately flash mixes with the bulk water.
E-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 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 - For flows out of the intact SGs, radioactivity to the environment is a function of the steaming rate, and the iodine partition factor is assumed to be 100. Moisture carryover is modeled at 0.32%.
E-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	Conforms – The steam generator with the faulted main steamline in the MSLB accident is assumed to blow completely dry, causing a

Table D: Conformance With Regulatory Guide 1.183 Appendix E (Main Steam Line Break Accident)		
RG Section	RG Position	VEGP Analysis
	(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.	direct release of radioactivity from that source to the environment.
Footnote E-1	Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, “Steam Generator Tube Integrity,” for acceptable assumptions and methodologies for performing radiological analyses.	Conforms – VEGP is licensed to ARC.
Footnote E-2	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 I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Consistent with the VEGP current licensing basis the maximum RCS concentrations allowed by TS are used.

Table E: Conformance With Regulatory Guide 1.183 Appendix F (Steam Generator Tube Rupture Accident)		
RG Section	RG Position	VEGP Analysis
F-1	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 – See discussions in Table A.
F-2	If no or minimal[2] 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.	Conforms-the maximum RCS concentrations allowed by TS are used.
F-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 – Case 1 is a pre-accident spike using the maximum Dose Equivalent Iodine permitted by the VEGP Technical Specifications.
F-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 concurrent iodine spike case assumes the RCS transient associated with the accident creates an iodine spike, causing the iodine release rate from the fuel rods to the RCS to increase to a value 335 times greater than the release rate that yields the maximum allowable equilibrium iodine concentration specified in the technical specifications. A 20-hour release duration is modeled.
F-3	The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms - Mixing in the primary coolant is assumed to be instantaneously and homogeneously.
F-4	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms – The iodine released to the environment is assumed to be 97% elemental and 3% organic.

Table E: Conformance With Regulatory Guide 1.183 Appendix F (Steam Generator Tube Rupture Accident)		
RG Section	RG Position	VEGP Analysis
F-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 assumed primary-to-secondary leak rate in the three intact steam generators are 0.7 gpm (1008 gallons per day). This is conservative relative to VEGP TS 3.4.13 which allows 150 gallons per day per Steam Generator.
F-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 assumed density is 62.4 lbm/ft^3.
F-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 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 - It is assumed that the RHR system is placed in service at 20 hours, terminating the accident.
F-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 - The SGTR assumes a concurrent LOOP to maximize the release to the environment.
F-5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms - Noble gases are modeled as going directly to the environment without reduction or mitigation.
F-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 – consistent with the CLB, flashing fraction is as calculated by the LOFTRAN code, and particulate transport is based on the maximum moisture carryover.
Footnote F-1	Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074,	Conforms – VEGP is licensed to ARC.

Table E: Conformance With Regulatory Guide 1.183 Appendix F (Steam Generator Tube Rupture Accident)		
RG Section	RG Position	VEGP Analysis
	"Steam Generator Tube Integrity" (USNRC, December 1998), for acceptable assumptions and methodologies for performing radiological analyses.	
Footnote F-2	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 I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Conforms-The initial RCS activities are based on the maximum allowable values by TS.

Table F: Conformance With Regulatory Guide 1.183 Appendix G (Locked Rotor Accident)		
RG Section	RG Position	VEGP Analysis
G-1	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 discussions in Table A.
G-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 – The transient causes fuel damage and so a radiological analysis is provided.
G-3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms - The gap activity in the damaged rods is instantaneously released to and uniformly mixed within the reactor coolant system at the onset of the accident.
G-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 – The iodine releases from the steam generators to the environment are 97% elemental and 3% organic for the pre-accident case and the concurrent iodine spike case, including damaged fuel.
G-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 – Leakage is 1 gpm, which is bounding over the Technical Specification limit of 150 gallons per day per steam generator.
G-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 assumed density is 62.4 lbm/ft^3.
G-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	Conforms – The accident terminates after 20 hours and the RHR system is placed in service.

Table F: Conformance With Regulatory Guide 1.183 Appendix G (Locked Rotor Accident)		
RG Section	RG Position	VEGP Analysis
	is less than 100°C (212°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.	
G-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 – The Locked Rotor Accident assumes a concurrent LOOP to maximize the release to the environment.
G-5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms - Noble gases are assumed to leak directly to the environment without holdup in the SG.
G-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 Position 5.5 and 5.6 of Appendix E is applied to releases from the steam generators.
Footnote G-1	Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, “Steam Generator Tube Integrity” (USNRC, December 1998), for acceptable assumptions and methodologies for performing radiological analyses.	Conforms – VEGP is licensed to ARC.

Table G: Conformance With Regulatory Guide 1.183 Appendix H (Rod Ejection Accident)		
RG Section	RG Position	VEGP Analysis
H-1	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 discussions in Table A. The fission product release is based upon Appendix H, the amount of damaged fuel, and the assumption that 10% of the core inventory of noble gases and iodine isotopes are in the fuel rod gap. Alkali metals (12%) and other halogens (5%) are also assumed to be in the fuel gap, consistent with Table 3. For releases from containment involving fuel melting, 100% of the noble gases and 25% of the iodine isotopes contained in the portion of the fuel that melts is available for release from containment. For releases to the RCS and to the environment through the secondary side, 100% of the noble gases and 50% of the iodines are assumed to be released from melted fuel.
H-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.	Not Applicable – Failed fuel is postulated for this event.
H-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 pathways are considered. In the release from containment, 100% of the activity from fuel melting, fuel cladding damage, and initial RCS inventory instantaneously reaches the containment at the onset of the accident and is available for release to the environment. In the case with the release from the secondary system, 100% of the activity from fuel melting, fuel cladding damage, and initial RCS inventory instantaneously reaches the RCS at the onset of the accident and is available for release to the secondary system and eventually to the environment.

H-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, 4.85% elemental iodine, and 0.15% organic iodide. Since containment sprays are not assumed to be activated in this event, no credit is taken for pH being controlled at values of 7 or greater.

Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Alternative Source Term,
TSTF-51, TSTF-471, and TSTF-490

Attachment 5

Loss-of-Coolant Accident Analysis

LOSS-OF-COOLANT ACCIDENT
DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number: X6CAJ.15, Version 2

Method/Computer Program Used: RADTRAD Version 3.10

Regulatory Guidance: RG-1.183, including Appendix A

Model Discussion

The calculation was performed in four parts, evaluating the contributions from four separate release paths: Containment mini-purge, Containment Leakage, ECCS Leakage Outside of Containment, and potential leakage from the Refueling Water Storage Tank (RWST). The dose contributions from each of these pathways were summed to obtain the doses to the Main Control Room (MCR), the Exclusion Area Boundary (EAB), and the Low Population Zone (LPZ). The accident duration is 30 days, per VEGP Current Licensing Basis (CLB).

Results and Acceptance Limits

Release	**EAB (rem TEDE), Max. 2 hr**	**LPZ (rem TEDE)**	**Control Room (rem TEDE)**
Containment Purge	0.001	0.001	0.001
Containment Leakage	6.6	5.1	2.2
ECCS Leakage	1.6	3.8	1.6
RWST Back-leakage	0.18	0.69	0.36
MCR External Sources	n/a	n/a	0.3
Total	**8.4**	**9.6**	**4.4**
Acceptance Limit	25	25	5

(Note that rounding is applied to all values)

Key Assumptions and Inputs

Source Term Parameters

Parameter	Value
Core Power Level:	3636 MWt (includes uncertainty)
Initial Core Source Term	Includes 10% Fuel Design Margin (FDM) added to the nominal activity

Table 1 - Core Source Term

Nuclide	Core Source Term (Ci)	Nuclide	Core Source Term (Ci)
Kr-83m	1.29E+07	Rh-103m	1.69E+08
Kr-85	1.12E+06	Rh-105	1.07E+08
Kr-85m	2.74E+07	Rh-106	6.16E+07
Kr-87	5.40E+07	Ru-103	1.69E+08
Kr-88	7.23E+07	Ru-105	1.18E+08
Xe-131m	1.42E+06	Ru-106	5.39E+07
Xe-133	2.15E+08	Tc-99	1.48E+03
Xe-133m	6.83E+06	Tc-99m	1.77E+08
Xe-135	5.10E+07	Ce-141	1.79E+08
Xe-135m	4.59E+07	Ce-143	1.65E+08
Xe-138	1.87E+08	Ce-144	1.34E+08
Br-82	3.41E+05	Np-237	3.44E+01
Br-83	1.28E+07	Np-238	4.27E+07
Br-84	2.32E+07	Np-239	2.07E+09
I-130	2.04E+06	Pu-238	3.02E+05
I-131	1.07E+08	Pu-239	3.09E+04
I-132	1.57E+08	Pu-240	4.50E+04
I-133	2.20E+08	Pu-241	1.30E+07
I-134	2.47E+08	Pu-242	2.06E+02
I-135	2.10E+08	Pu-243	4.22E+07
Cs-134	1.67E+07	Am-241	1.28E+04
Cs-134m	4.50E+06	Am-242	7.06E+06
Cs-135	4.51E+01	Am-243	2.53E+03
Cs-136	5.37E+06	Cm-242	3.69E+06
Cs-137	1.18E+07	Cm-244	3.71E+05

Nuclide	Core Source Term (Ci)	Nuclide	Core Source Term (Ci)
Cs-138	2.04E+08	Eu-154	6.80E+05
Rb-86	2.07E+05	Eu-155	2.87E+05
Rb-88	7.36E+07	Eu-156	2.46E+07
Rb-89	9.64E+07	La-140	1.96E+08
Sb-124	7.92E+04	La-141	1.76E+08
Sb-125	8.59E+05	La-142	1.70E+08
Sb-126	5.11E+04	La-143	1.63E+08
Sb-127	9.75E+06	Nb-95	1.84E+08
Sb-129	3.03E+07	Nb-95m	2.09E+06
Te-125m	1.82E+05	Nb-97	1.83E+08
Te-127	9.55E+06	Nb-97m	1.73E+08
Te-127m	1.58E+06	Nd-147	7.08E+07
Te-129	2.84E+07	Pm-147	1.77E+07
Te-129m	5.44E+06	Pm-148	1.93E+07
Te-131	9.10E+07	Pm-148m	4.01E+06
Te-131m	2.07E+07	Pm-149	6.53E+07
Te-132	1.53E+08	Pm-151	2.06E+07
Te-133	1.17E+08	Pr-143	1.61E+08
Te-133m	1.03E+08	Pr-144	1.35E+08
Te-134	1.96E+08	Pr-144m	1.88E+06
Sr-89	1.02E+08	Sm-151	4.47E+04
Sr-90	8.70E+06	Sm-153	4.93E+07
Sr-91	1.27E+08	Y-90	9.08E+06
Sr-92	1.37E+08	Y-91	1.34E+08
Ba-137m	1.12E+07	Y-91m	7.39E+07
Ba-139	1.95E+08	Y-92	1.38E+08
Ba-140	1.89E+08	Y-93	1.57E+08

Nuclide	Core Source Term (Ci)	Nuclide	Core Source Term (Ci)
Ba-141	1.75E+08	Y-95	1.72E+08
Mo-99	2.00E+08	Zr-95	1.82E+08
Pd-109	3.49E+07	Zr-97	1.82E+08

Parameter	Value
Initial RCS Source Term	Noble Gases, Halogens, and Alkali Metals
I-131 through I-135	1 µCi/g Dose Equivalent I-133
Noble Gases (except Kr-83m)	280 µCi/g Dose Equivalent Xe-133 consistent with TSTF-490 Rev. 0
Other Nuclides	Based on 1% Fuel Clad Defects; includes 10% Fuel Design Margin
RCS Mass	2.258E+08 grams
Release Fractions	Per RG-1.183
Release Timing	Per RG-1.183

Table 2 - RCS Source Term

Nuclide	RCS Source Term (µCi/g)	Nuclide	RCS Source Term (µCi/g)
Kr-83m	4.69E-01	I-131	7.61E-01
Kr-85	3.84E+00	I-132	7.79E-01
Kr-85m	7.46E-01	I-133	1.16E+00
Kr-87	4.87E-01	I-134	1.66E-01
Kr-88	1.64E+00	I-135	6.40E-01
Xe-131m	1.46E+00	Cs-134	1.99E+00
Xe-133	1.15E+02	Cs-134m	2.46E-02
Xe-133m	1.59E+00	Cs-135	0.00E+00
Xe-135	3.47E+00	Cs-136	2.72E+00
Xe-135m	2.19E-01	Cs-137	1.92E+00
Xe-138	2.77E-01	Cs-138	1.07E+00
Br-82	5.12E-03	Rb-86	2.68E-02
Br-83	9.88E-02	Rb-88	4.33E+00
Br-84	4.66E-02	Rb-89	1.01E-01
I-130	2.19E-02		

Note 1: Cs-135 activity omitted as it is considered to be a negligible contributor to doses with respect to the other included nuclides.

Containment Leakage Parameters

Parameter	Value
Containment Volume	2.930E+06 cubic feet
Sprayed Volume	2.30E+06 cubic feet
Unsprayed Volume	6.30E+05 cubic feet
Containment Leakage	0.21% of volume per day for first 24 hours
(Tech. Spec. Leakage plus 5% margin)	0.105% of volume per day for remainder
Containment Leakage Filtration	None
Containment Long Term Sump pH	pH ≥ 7.0 (no re-evolution of Iodine)
Containment spray removal λ, Elemental	13.7 hr^{-1}
Containment spray removal λ, Aerosol	5.34 hr^{-1}
Containment Spray Organic removal	None

Natural Deposition, Aerosol only	0.1 hr^{-1} after sprays are terminated
Containment Spray Start	10 seconds
Containment Spray Stop	2 Hours
Containment Spray Flow	2,500 gal/min
Iodine Chemical Form	95% Cesium Iodide, 4.85% elemental, 0.15% organic

Containment Purge Leakage

Parameter	Value
Iodine Chemical Form	95% Cesium Iodide, 4.85% elemental, 0.15% organic
Containment Purge Filtration	None
Removal by wall deposition	0%
Removal by Sprays	0%
Containment Purge Isolation	≤30 seconds
Containment Purge Flowrate	5000 CFM

ECCS Leakage

Parameter	Value
Sump Volume	114,922 cubic feet
Sump temperature	Varies, max is 260 ºF
ECCS Leakage Initiation Time	30 minutes
ECCS Leakage Iodine Flashing Factor	10%
Iodine Species ECCS Leakage Released to the Atmosphere	
Elemental	97%
Organic	3%
ECCS Leakage Rate	2.0 gal/min

RWST Leakage Parameters

Parameter	Value
ECCS Recirculation Start Time	30 minutes
Iodine Species ECCS Leakage Released to the Atmosphere from the RWST	
Elemental	100%
Organic	0%
ECCS Leakage Rate to the RWST	7 gal/min
RWST Leakage Iodine Flashing Factors	Varies with temperature and pH between 0% and 10.03%
RWST Capacity	731,000 gallons
RWST Volume at Transfer to Recirculation	95,837.7 gallons

CR Parameters:

Parameter	Value
CR Volume	149,000 ft^3
CR Isolation	Automatic at 11.3 Seconds
CR Pressurization Mode Initiation	Automatic at 99.3 Seconds
CR Ventilation System Normal Flow Rate	2,575 cfm < 11.3 seconds
CR Ventilation System Filtered Makeup Rate	1,800 cfm > 99.3 seconds
CR Ventilation System Recirculation Flow Rate	31,000 cfm > 99.3 seconds
CR Ventilation System Charcoal Filter Efficiencies (Supply and Recirculation use the same filter)	
All Iodine Species	99%
Particulates	99%
CR Unfiltered In-leakage	180 cfm > 11.3 seconds
CR Ingress/Egress Unfiltered In-leakage	10 cfm > 0 seconds
CR Breathing Rate	3.5E-4 m^3/sec
Occupancy Factors	
0-24 hours	1.0
1 - 4 days	0.6
4 -30 days	0.4

Atmospheric Dispersion Factors (sec/m^3):

All Release Pathways

Time (hr)	EAB*	LPZ	CR
0 – 2	1.8E-4	7.20E-5	2.22E-03
2 – 8	-	3.30E-5	1.55E-03
8 – 24	-	2.20E-5	6.57E-04
24 – 96	-	9.20E-6	5.80E-04
96 – 720	-	2.70E-6	4.47E-04

*Dispersion Factor applied for entire duration of transient in analysis

EAB & LPZ Breathing Rates

0-8 Hours	3.5E-04 m^3/sec
8-24 Hours	1.8E-04 m^3/sec
24 Hours – 30 Days	2.3E-04 m^3/sec

Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Alternative Source Term,
TSTF-51, TSTF-471, and TSTF-490

Attachment 6

Fuel Handling Accident Analysis

FUEL HANDLING ACCIDENT
DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number: X6CAJ.16, Version 3
Method/Computer Program Used: RADTRAD Version 3.10
Regulatory Guidance: RG-1.183, including Appendix B

Model Discussion

The calculation was performed to address a fuel handling accident (FHA) in the SFP area of the Auxiliary Building. For a FHA in containment, the X/Qs comparisons between an accident in the fuel building and for one in containment shows the FHA in the fuel building is more limiting. As a result, only an accident in the fuel building is analyzed. Doses in the CR are accumulated over a period of 24 hours. Releases from the damaged fuel are completed in 2 hours.

The activity is released to the environment through the plant vent stack at the elevation of the fuel building roof to maximize the releases to the CR. No credit is taken for filtration of the iodine isotopes is assumed. By inspection, doses from this accident bound the doses from an accident in containment.

Results and Acceptance Limits

Release	EAB (rem TEDE)	LPZ (rem TEDE)	Control Room (rem TEDE)
Spent Fuel Pool	1.0	0.4	3.9
Acceptance Limit	6.3	6.3	5

(Note that rounding is applied to all values)

Key Assumptions and Inputs

Source Term Parameters

Parameter	Value
Reactor Power Level	3636 MWt (includes uncertainty)
Radial Peaking Factor	1.7
Fuel Movement Time	70 hours post shutdown.
Number of Fuel Assemblies	193
Number of Fuel Rods/Assembly	264
Number of Rods in Core	50,952
Number of Damaged Assemblies	1 + 50 rods in target assembly
Number of Damaged Fuel Rods	314
Fuel Handling Building Volume	1 cu ft
Fuel Building Exhaust Rate	0.115 cfm (ensures all activity released in 2 hours)

Table 1 - Source Term

Nuclide	70-Hour Core Source Term (Ci)	Gap Fraction	DF	Released Activity (314 Rods) (Ci)
Kr-83m	9.73E-02	8.00E-02	1	8.23E-05
Kr-85	1.12E+06	3.80E-01	1	4.51E+03
Kr-85m	5.49E+02	3.80E-01	1	2.20E+00
Kr-87	1.48E-09	8.00E-02	1	1.25E-12
Kr-88	2.75E+00	8.00E-02	1	2.33E-03
Xe-131m	1.36E+06	8.00E-02	1	1.15E+03
Xe-133	1.73E+08	8.00E-02	1	1.47E+05
Xe-133m	3.99E+06	8.00E-02	1	3.38E+03
Xe-135	2.59E+06	8.00E-02	1	2.19E+03
Xe-135m	2.13E+04	8.00E-02	1	1.80E+01
Xe-138	0.00E+00	8.00E-02	1	0.00E+00
Br-82	8.65E+04	5.00E-02	200	2.28E-01
Br-83	2.32E-02	5.00E-02	200	6.12E-08
Br-84	0.00E+00	5.00E-02	200	0.00E+00
I-130	4.04E+04	5.00E-02	200	1.07E-01
I-131	8.59E+07	8.00E-02	200	3.63E+02
I-132	8.48E+07	9.00E-02	200	4.03E+02
I-133	2.19E+07	5.00E-02	200	5.79E+01
I-134	0.00E+00	5.00E-02	200	0.00E+00
I-135	1.30E+05	5.00E-02	200	3.44E-01

Released Activity = (Radial Peaking Factor) * [(70-Hour Core Source Term) * (Gap Fraction) * (Number of Damaged Rods)/(Number of Rods in Core)]/(Net Decontamination Factor)

Overlaying Pool Depth	23 feet
Iodine Chemical Form	0% Aerosol, 99.85% Elemental 0.15% Organic
Net Decontamination Factor	200
Fuel Design Margin	10%

While it is generally considered that Br in the Halogen group is in particulate form (RG-1.183 Section 3.5) and would be entrained in the 23 ft pool water depth, it is conservatively considered here. The conservative treatment of Br is considered to alleviate any considerations due to interpretation of PNNL-18212 Rev. 1 Table 2.9 for the “Other Halogens.”

CR Parameters:

Parameter	Value
CR Volume	149,000 ft^3
CR Isolation	Automatic at 0.169 hour
CR Pressurization Mode Initiation	Automatic at 0.194 hour
CR Ventilation System Normal Flow Rate	2,575 cfm < 0.169 hour
CR Ventilation System Filtered Makeup Rate	1,800 cfm > 0.194 hour
CR Ventilation System Recirculation Flow Rate	31,000 cfm > 0.194 hour
CR Unfiltered In-leakage	180 cfm > 0.169 hour
CR Ingress/Egress Unfiltered In-leakage	10 cfm > 0 seconds
CR Ventilation System Charcoal Filter Efficiencies (Supply and Recirculation use the same filter)	
All Iodine Species	99%
Particulates	99%
CR Breathing Rate	3.5E-4 m^3/sec
Occupancy Factors	
0-24 hours	1.0

Atmospheric Dispersion Factors

Time (hr)	X/Q (sec/m^3)		
	EAB*	LPZ	MCR
0 – 2	1.80E-04	7.20E-05	6.01E-03
2 – 8	-	3.30E-05	4.44E-03
8 – 24	-	2.2E-05	1.71E-03

* Applied for entire duration of event in analysis

EAB & LPZ Breathing Rates

0-8 Hours	3.5E-04 m^3/sec
8-24 Hours	1.8E-04 m^3/sec

Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Alternative Source Term,
TSTF-51, TSTF-471, and TSTF-490

Attachment 7

Main Steam Line Break Analysis

MAIN STEAM LINE BREAK ACCIDENT
DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number: X6CAJ.17, Version 2

Method/Computer Program Used: RADTRAD Version 3.10

Regulatory Guidance: RG-1.183, including Appendix E

Model Discussion:

The calculation was performed to address a Main Steam Line Break (MSLB). Per RG-1.183, two cases are considered for the dose-equivalent I-131 (DEI) concentrations in the Reactor Coolant System (RCS):

1. Pre-Accident Iodine Spike – a reactor transient occurs prior to the accident and raises the RCS iodine concentration to the maximum value permitted by the technical specifications.
2. Concurrent Iodine Spike – the RCS transient associated with the accident creates an iodine spike, causing the iodine release rate from the fuel rods to the RCS to increase to a value 500 times greater than the release rate that yields the equilibrium iodine concentration specified in the technical specifications.

Primary to Secondary leakage (consistent with CLB) is assumed to be 0.35 gallons per minute (gpm) to the “faulted” steam generator (SG), and 0.65 gpm (total) going to the three intact SGs. It is postulated that the MSLB causes the associated “faulted” SG to blow dry, releasing activity directly to the environment through the broken main steam line. Activity from three intact SGs released to the environment via steaming until the primary system (RCS) is placed on RHR cooling (assumed at 20 hours).

Doses for the Pre-Accident Iodine Spike Case and the Concurrent Iodine Spike Case were calculated, with results shown below.

Results and Acceptance Limits

Case	Location	Dose (Rem TEDE)	
		Calculated	Limit
Pre-Accident Iodine Spike	EAB	<0.1	25
	LPZ	<0.1	25
	Control Room	<0.1	5
Concurrent Iodine Spike	EAB	0.2	2.5
	LPZ	0.2	2.5
	Control Room	0.2	5

(Note that rounding is applied to all values)

Key Assumptions and Inputs

Source Terms

Table 1 – RCS Source Terms

	Pre-Accident Iodine Spike		Concurrent Iodine Spike	
Nuclide	RCS Source Term (μCi/g)	RCS Source Term (Ci)	RCS Source Term (μCi/g)	RCS Source Term (Ci)
Kr-83m	4.69E-01	1.059E+02	4.69E-01	1.059E+02
Kr-85	3.84E+00	8.669E+02	3.84E+00	8.669E+02
Kr-85m	7.46E-01	1.684E+02	7.46E-01	1.684E+02
Kr-87	4.87E-01	1.100E+02	4.87E-01	1.100E+02
Kr-88	1.64E+00	3.696E+02	1.64E+00	3.696E+02
Xe-131m	1.46E+00	3.292E+02	1.46E+00	3.292E+02
Xe-133	1.15E+02	2.594E+04	1.15E+02	2.594E+04
Xe-133m	1.59E+00	3.597E+02	1.59E+00	3.597E+02
Xe-135	3.47E+00	7.838E+02	3.47E+00	7.838E+02
Xe-135m	2.19E-01	4.940E+01	2.19E-01	4.940E+01
Xe-138	2.77E-01	6.245E+01	2.77E-01	6.245E+01
Br-82	5.12E-03	1.157E+00	5.12E-03	1.157E+00
Br-83	9.88E-02	2.230E+01	9.88E-02	2.230E+01
Br-84	4.66E-02	1.051E+01	4.66E-02	1.051E+01
I-130	2.19E-02	4.938E+00	2.19E-02	4.938E+00
I-131	4.56E+01	1.030E+04	7.61E-01	1.717E+02
I-132	4.68E+01	1.056E+04	7.79E-01	1.760E+02
I-133	6.95E+01	1.569E+04	1.16E+00	2.615E+02
I-134	9.95E+00	2.246E+03	1.66E-01	3.744E+01
I-135	3.84E+01	8.677E+03	6.40E-01	1.446E+02
Cs-134	1.99E+00	4.491E+02	1.99E+00	4.491E+02
Cs-134m	2.46E-02	5.549E+00	2.46E-02	5.549E+00
Cs-135	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Cs-136	2.72E+00	6.135E+02	2.72E+00	6.135E+02
Cs-137	1.92E+00	4.337E+02	1.92E+00	4.337E+02
Cs-138	1.07E+00	2.409E+02	1.07E+00	2.409E+02
Rb-86	2.68E-02	6.041E+00	2.68E-02	6.041E+00
Rb-88	4.33E+00	9.766E+02	4.33E+00	9.766E+02
Rb-89	1.01E-01	2.282E+01	1.01E-01	2.282E+01

Table 2 – Concurrent Spike RCS Iodine Appearance Rate

Isotope	Nominal Appearance Rate, Ci/hr	500* Nominal Appearance Rate, Ci/hr
I-131	1.75E+01	8.75E+03
I-132	7.05E+01	3.53E+04
I-133	3.44E+01	1.72E+04
I-134	3.33E+01	1.67E+04
I-135	2.95E+01	1.48E+04

Table 3 – Secondary Side Source Terms

	Intact Steam Generators		Faulted Steam Generator	
Nuclide	Secondary Side Source Term (µCi/g)	Secondary Side Source Term (Ci)	Secondary Side Source Term (µCi/g)	Secondary Side Source Term (Ci)
Kr-83m	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Kr-85	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Kr-85m	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Kr-87	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Kr-88	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Xe-131m	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Xe-133	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Xe-133m	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Xe-135	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Xe-135m	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Xe-138	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Br-82	5.12E-04	6.842E-02	5.12E-04	2.281E-02
Br-83	9.88E-03	1.318E+00	9.88E-03	4.395E-01
Br-84	4.66E-03	6.215E-01	4.66E-03	2.072E-01
I-130	2.19E-03	2.919E-01	2.19E-03	9.731E-02
I-131	7.61E-02	1.015E+01	7.61E-02	3.385E+00
I-132	7.79E-02	1.040E+01	7.79E-02	3.468E+00
I-133	1.16E-01	1.546E+01	1.16E-01	5.154E+00
I-134	1.66E-02	2.214E+00	1.66E-02	7.378E-01
I-135	6.40E-02	8.550E+00	6.40E-02	2.850E+00
Cs-134	1.99E-01	2.655E+01	1.99E-01	8.850E+00
Cs-134m	2.46E-03	3.281E-01	2.46E-03	1.094E-01
Cs-135	0.00E+00	0.000E+00	0.00E+00	0.000E+00
Cs-136	2.72E-01	3.627E+01	2.72E-01	1.209E+01
Cs-137	1.92E-01	2.564E+01	1.92E-01	8.547E+00
Cs-138	1.07E-01	1.424E+01	1.07E-01	4.747E+00
Rb-86	2.68E-03	3.571E-01	2.68E-03	1.190E-01

	Intact Steam Generators		Faulted Steam Generator	
Nuclide	Secondary Side Source Term (µCi/g)	Secondary Side Source Term (Ci)	Secondary Side Source Term (µCi/g)	Secondary Side Source Term (Ci)
Rb-88	4.33E-01	5.774E+01	4.33E-01	1.925E+01
Rb-89	1.01E-02	1.349E+00	1.01E-02	4.498E-01

Physical Parameters

Parameter	Value
RCS Mass	2.258E8 grams
RCS Volume	1.11E4 cubic feet
Intact SG Mass	1.335E8 grams (3 total)
Intact SG Volume	4,710 cubic feet
Faulted SG Mass	4.45E7 grams
Faulted SG Volume	1,570 cubic feet
Coolant Densities	Primary and Secondary water at 62.4 lbm/ft^3

Table 14 – MSLB Flow Rates

Flow Path	Time (hour)		Release (lbm)	Flow	Note
	From	to			
RCS to Env	0	20	-	0.134 cfm	1
RCS to Intact SGs	0	20	-	0.087 cfm	
RCS to Faulted SG	0	20	-	0.047 cfm	
Intact SGs to Env	0	2	466,400	62.3 cfm	2
	2	8	1,056,000	47.0 cfm	
	8	20	2,112,000	47.0 cfm	
Faulted SG to Env	0	0.5	-	1,000 cfm	3

Flow Rate Notes:

1. RCS Leakage of 1 gpm – Volumetric leakage (gpm) from RCS is divided by 7.48 gal/ft^3.
2. Intact SGs – Mass release from the intact SGs is multiplied by 1.1. Flow is the release (lbm) divided by 62.4 lbm/ft^3 and by the time duration (min).
3. Faulted SG – Flow is set arbitrarily high (consistent with CLB) in order to force release of secondary side inventory within 0.5 hours, consistent with a blowdown of the faulted steam generator.

CR Parameters:

Parameter	Value
CR Volume	149,000 ft^3
CR Isolation	Automatic at 11.3 seconds
CR Pressurization Mode Initiation	Automatic at 99.3 seconds
CR Ventilation System Normal Flow Rate	2,575 cfm < 11.3 seconds
CR Ventilation System Filtered Makeup Rate	1,800 cfm > 99.3 seconds

CR Ventilation System Recirculation Flow Rate	31,000 cfm > 99.3 seconds
CR Unfiltered In-leakage	180 cfm > 11.3 seconds
CR Ingress/Egress Unfiltered In-leakage	10 cfm > 0 seconds
CR Ventilation System Charcoal Filter Efficiencies (Supply and Recirculation use the same filter)	
All Iodine Species	99%
Particulates	99%
CR Breathing Rate	3.5E-4 m^3/sec
Occupancy Factors	
0-24 hours	1.0
1 - 4 days	0.6
4 -30 days	0.4

Atmospheric Dispersion Factors

Time (hr)	X/Q (sec/m^3)			
	EAB*	LPZ	MCR	Bounding Release/Receptor Location
0 – 2	1.80E-04	7.20E-05	7.64E-03	U1 North MSIV Room to U1 MCR Intake
2 – 8	-	3.30E-05	6.17E-03	U1 North MSIV Room to U1 MCR Intake
8 – 24	-	2.2E-05	2.72E-03	U2 North MSIV Room to U2 MCR Intake
24 – 96	-	9.2E-06	1.86E-03	U2 North MSIV Room to U2 MCR Intake
96 – 720	-	2.7E-06	1.52E-03	U2 North MSIV Room to U2 MCR Intake

* Applied for entire duration of event in analysis

EAB & LPZ Breathing Rates

0-8 Hours	3.5E-04 m3/sec
8-24 Hours	1.8E-04 m3/sec
24 Hours – 30 Days	2.3E-04 m3/sec

Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Alternative Source Term,
TSTF-51, TSTF-471, and TSTF-490

Attachment 8

Steam Generator Tube Rupture Accident Analysis

STEAM GENERATOR TUBE RUPTURE ACCIDENT
DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number: X6CAJ.18, Version 1
Method/Computer Program Used: RADTRAD Version 3.10
Regulatory Guidance: RG-1.183, including Appendix F

Model Discussion:

The calculation was performed to address a steam generator tube rupture (SGTR). Mass transfers from the primary to secondary are calculated using the Westinghouse proprietary software tool LOFTRAN, in accordance with the VEGP current licensing basis (CLB). One steam generator (SG) tube is assumed to fail, rupturing cleanly in two. Mass transfer from the primary to the secondary continues until the break flow is terminated. Activity is released to the environment from the faulted generator until operator action is taken to isolate it. Break flow and releases are as calculated by LOFTRAN.

Primary to secondary leakage through pin-hole leaks in the SG tubes is assumed at a rate of 1 gpm (0.3 gpm to the faulted SG, 0.7 gpm to the intact SGs per the current VEGP CLB) until the SGs are isolated or no longer used for cooling. Activity is released from the other three, intact, generators through steaming via the atmospheric relief valves (ARVs) until the primary system (RCS) is reduced to cold shutdown conditions (assumed at 20 hours). The analysis models an exposure duration of 30 days for the LPZ and occupants of the CR.

Doses for the Pre-Accident Iodine Spike Case and the Concurrent Iodine Spike Case were calculated, with results shown below.

Results and Acceptance Limits:

	Pre-Accident Spike	**Concurrent Spike**	**Acceptance Criteria (Pre-Accident/Concurrent)**
Location/Dose Point	**TEDE* (REM)**	**TEDE* (REM)**	**TEDE (REM)**
Exclusion Area Boundary (EAB)	1.6	1.4	25/2.5
Low Population Zone (LPZ)	0.9	0.8	25/2.5
Main Control Room (MCR)	0.6	0.5	5.0

Results conservatively rounded up to the nearest 0.1 Rem.

Key Assumptions and Inputs:

Transient Timing

Tube Rupture:	Time Zero (0)
Reactor Trip:	49.7 seconds
Faulted Generator Isolated	1200 seconds
Break Flow Terminated	5502 seconds
ARV Release from Faulted SG Ended	2162 seconds
RHR In Service	20 hours

Physical Parameters:

Parameter	Value
RCS Mass	2.258E08 grams
RCS Volume	1.11E+04 cubic feet
Intact SG Mass	1.33E+08 grams (total of 3)
Intact SG Volume	4.71E+03 cubic feet (total of 3)
Faulted SG Volume	1.57E+03 cubic feet
Coolant Densities	Primary and Secondary water at 62.4 lbm/ft
Blowdown Sample Line Flow per SG	0.48 lbm/s

Table 1 – SGTR-RCS Flow Rates

RCS Transfer Pathways					
Description	**Time**	**Flow lbm/hr**	**Flow (CFM)**		
RCS Breakflow to FSG*	0-1.53 h	134067.61	35.81		
RCS Leakage to FSG**,***	0-20 h	150.16	0.04		
RCS Leakage to ISG**,***	0-20 h	350.37	0.09		
Nobles (CFM)	**PF**	**Iodine (CFM)**	**PF**	**Alkali (CFM)**	**PF**
35.81	1.00	35.81	1.00	35.81	1.00
0.04	1.00	0.04	1.00	0.04	1.00
0.09	1.00	0.09	1.00	0.09	1.00

*No partitioning is modeled for breakflow to the FSG
**A conversion factor of 7.48 gal = 1ft^3 was used in the conversion of the RCS leakage flows
***RCS leakage pathway conservatively does not take credit for flashing fraction or partition factors for Iodine or Alkali Metals.

Table 2 – SGTR-FSG Flow Rates

FSG Transfer Pathways					
Description	**Time**	**Flow lbm/hr**	**Flow (CFM)**		
FSG to Env.**, ***	0-2 h	107000.00	28.58		
FSG to Env.	2-20 h	2061.11	0.55		
Nobles (CFM)	**PF**	**Iodine (CFM)**	**PF**	**Alkali (CFM)**	**PF**
28.58	1.00	4.29	0.15	0.09	312.50
0.55	1.00	0.0055	100.00	0.0018	312.50
Description	**Time**	**Flow lbm/hr**	**Flow (CFM)**		
Blowdown SL to Env.*	0-20 h	1728.00	0.46		
Nobles (CFM)	**PF**	**Iodine (CFM)**	**PF**	**Alkali (CFM)**	**PF**
0.46	1.00	0.46	1.00	0.46	1.00

*The blowdown sample lines discharge from the liquid region of the SG. As a result of this no partitioning of Iodine or Alkali metals is assumed to occur for this transfer pathway.

**Discharge to the environment is modeled to occur for 2 hours to remain consistent with the T/H input to dose analyses as output by LOFTRAN

***No additional partitioning is credited for the contribution of FSG flow to the condenser. This is conservative

Table 3 – SGTR-ISG Flow Rates

Table 7-4: ISG to Environment Transfer Pathways					
Description	**Time**	**Flow lbm/hr**	**Flow (CFM)**		
ISG to Env.	0-2 h	348250.00	93.02		
ISG to Env.	2-20 h	138644.44	37.03		
Nobles (CFM)	**PF**	**Iodine (CFM)**	**PF**	**Alkali (CFM)**	**PF**
93.02	1.00	0.93	100.00	0.30	312.50
37.03	1.00	0.37	100.00	0.12	312.50
Description	**Time**	**Flow lbm/hr**	**Flow (CFM)**		
Blowdown SL to Env.*,**	0-20 h	5184.00	1.38		
Nobles (CFM)	**PF**	**Iodine (CFM)**	**PF**	**Alkali (CFM)**	**PF**
1.38	1.00	1.38	1.00	1.38	1.00

*The blowdown sample lines discharge from the liquid region of the SG. As a result of this no partitioning of Iodine or Alkali metals is assumed to occur for this transfer pathway. **The blowdown sample line flows for the ISG is 3 times that of the FSG.

Radioactivity Considerations:

No fuel failure occurs as a result of the SGTR.

Iodine Release Species:	97% elemental, 3% organic.
Initial RCS activity	
I-131 through I-135	
Pre-Accident Iodine Spike Case	60 µCi/gm DEI-131
Concurrent Accident Spike	1.0 µCi/gm DEI-131
Noble Gases other than Xe-133m	280 µCi/gm DE Xe-133
Other Noble Gases, Halogens, and Alkali Metals	1% clad defects
Concurrent Accident Spike Appearance	335 times the Normal Rate for the first 8 hours

Table 4 - Normal RCS Iodine Concentrations

Nuclide	RCS Source Term (µCi/gm)	RCS Source Term (Ci)
I-130	2.19E-02	4.938E+00
I-131	7.61E-01	1.717E+02
I-132	7.79E-01	1.760E+02
I-133	1.16E+00	2.615E+02
I-134	1.66E-01	3.744E+01
I-135	6.40E-01	1.446E+02

Table 4 Notes:

1. Normal RCS Iodine source term concentrations multiplied by RCS mass of 2.258E+08 grams and 1E-06Ci/µCi to obtain the specific activity.

Table 5 – Normal Iodine Appearance Rates

Radionuclide	Appearance Rate (Ci/hr)
I-131	1.75E+01
I-132	7.05E+01
I-133	3.44E+01
I-134	3.33E+01
I-135	2.95E+01

Table 5 Notes:

1. The Concurrent spike case multiplies the normal values in Table 5 by 335 and injects into the RCS over an 8 hour period.

Initial Iodine Activities in RCS and Secondary Coolant

The initial iodine activities in the RCS and the secondary coolant corresponding to 60, 1.0, and 0.1 µCi/g DEI are shown in the following table. The normal RCS activity is used as the starting point for the concurrent spike case.

Table 6 – RCS Source Terms for Pre-Accident and Concurrent Iodine Spike Cases

Nuclide	RCS (Normal) (Ci)	RCS Pre-Accident Spike (Ci)	FSG (Ci)	ISG (Ci)
Kr-83m	1.06E+02	1.06E+02	0.00E+00	0.00E+00
Kr-85	8.68E+02	8.68E+02	0.00E+00	0.00E+00
Kr-85m	1.69E+02	1.69E+02	0.00E+00	0.00E+00
Kr-87	1.10E+02	1.10E+02	0.00E+00	0.00E+00
Kr-88	3.70E+02	3.70E+02	0.00E+00	0.00E+00
Xe-131m	3.30E+02	3.30E+02	0.00E+00	0.00E+00
Xe-133	2.60E+04	2.60E+04	0.00E+00	0.00E+00
Xe-133m	3.60E+02	3.60E+02	0.00E+00	0.00E+00
Xe-135	7.85E+02	7.85E+02	0.00E+00	0.00E+00
Xe-135m	4.94E+01	4.94E+01	0.00E+00	0.00E+00
Xe-138	6.25E+01	6.25E+01	0.00E+00	0.00E+00
Br-82	1.16E+00	1.16E+00	2.28E-02	6.83E-02
Br-83	2.23E+01	2.23E+01	4.39E-01	1.32E+00
Br-84	1.05E+01	1.05E+01	2.07E-01	6.21E-01
I-130	4.94E+00	4.94E+00	9.72E-02	2.92E-01
I-131	1.72E+02	1.03E+04	3.38E+00	1.01E+01
I-132	1.76E+02	1.06E+04	3.46E+00	1.04E+01
I-133	2.62E+02	1.57E+04	5.15E+00	1.54E+01
I-134	3.75E+01	2.25E+03	7.37E-01	2.21E+00
I-135	1.45E+02	8.68E+03	2.85E+00	8.54E+00
Cs-134	4.49E+02	4.49E+02	8.84E+00	2.65E+01
Cs-134m	5.55E+00	5.55E+00	1.09E-01	3.28E-01
Cs-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-136	6.14E+02	6.14E+02	1.21E+01	3.62E+01
Cs-137	4.34E+02	4.34E+02	8.54E+00	2.56E+01
Cs-138	2.41E+02	2.41E+02	4.74E+00	1.42E+01
Rb-86	6.05E+00	6.05E+00	1.19E-01	3.57E-01
Rb-88	9.77E+02	9.77E+02	1.92E+01	5.77E+01
Rb-89	2.28E+01	2.28E+01	4.49E-01	1.35E+00

Table 7 – Feedwater Source Terms

Nuclide	FSG (Ci/hr)	ISG (Ci/hr)
I-130	1.05E+00	1.65E-01
I-131	3.64E+01	5.72E+00
I-132	3.73E+01	5.86E+00
I-133	5.54E+01	8.71E+00
I-134	7.93E+00	1.25E+00
I-135	3.06E+01	4.82E+00

Table 7 Notes:

1. Activity injection from main feedwater is modeled to stop at the time of Rx trip (49.7 seconds).

CR Parameters:

Parameter	Value
CR Volume	149,000 ft^3
CR Isolation	Automatic at 121 seconds
CR Pressurization Mode Initiation	Automatic at 211 seconds
CR Ventilation System Normal Flow Rate	2,575 cfm < 121 seconds
CR Ventilation System Filtered Makeup Rate	1,800 cfm > 121 seconds
CR Ventilation System Recirculation Flow Rate	31,000 cfm > 121seconds
CR Unfiltered In-leakage	180 cfm > 121 seconds
CR Ingress/Egress Unfiltered In-leakage	10 cfm > 0 seconds
CR Ventilation System Charcoal Filter Efficiencies (Supply and Recirculation use the same filter)	
All Iodine Species	99%
Particulates	99%
CR Breathing Rate	3.5E-4 m^3/sec
Occupancy Factors	
0-24 hours	1.0
1 - 4 days	0.6
4 -30 days	0.4

Atmospheric Dispersion Factors

Time (hr)	X/Q (sec/m^3)			
	EAB*	LPZ	MCR	Bounding Release/Receptor Location
0 – 2	1.80E-04	7.20E-05	7.64E-03	U1 North MSIV Room to U1 MCR Intake
2 – 8	-	3.30E-05	6.17E-03	U1 North MSIV Room to U1 MCR Intake
8 – 24	-	2.2E-05	2.72E-03	U2 North MSIV Room to U2 MCR Intake
24 – 96	-	9.2E-06	1.86E-03	U2 North MSIV Room to U2 MCR Intake
96 – 720	-	2.7E-06	1.52E-03	U2 North MSIV Room to U2 MCR Intake

* Applied for entire duration of event in analysis

EAB & LPZ Breathing Rates

0-8 Hours	3.5E-04 m^3/sec
8-24 Hours	1.8E-04 m^3/sec
24 Hours – 30 Days	2.3E-04 m^3/sec

Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Alternative Source Term,
TSTF-51, TSTF-471, and TSTF-490

Attachment 9

Control Rod Ejection Accident Analysis

CONTROL ROD EJECTION ACCIDENT
DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number: X6CAJ.20, Version 2

Method/Computer Program Used: RADTRAD, version 3.10

Regulatory Guidance: RG-1.183 Rev. 0, including Appendix H

Model Discussion

The calculation was performed to address a Control Rod Ejection Accident (CREA). The scenario for the CREA is that the reactivity excursion due to a control rod ejection leads to localized fuel damage (including a small fraction of fuel melt). The local fuel damage results in increased radioactivity in the Reactor Coolant System (RCS). Activity in the steam generators (SG) due to primary-to-secondary leakage is released to the environment via steaming until the RCS is placed on RHR cooling.

Two release pathways are considered independently, in accordance with RG-1.183:

- Containment Leakage – Activity from fuel melting, fuel cladding damage, and initial RCS inventory instantaneously reaches the containment at the onset of the accident and is available for release to the environment.
- Secondary System Release – Activity from fuel melting, fuel cladding damage, and initial RCS inventory instantaneously mixes in the RCS at the onset of the accident and is available for release to the secondary system and eventually to the environment.

Results and Acceptance Limits

Location	Dose (Rem TEDE)		
	Containment Release	Secondary Side Release	Limit
EAB	1.5	0.5	6.3
LPZ	1.8	0.4	6.3
Control Room	0.6	1.1	5

(Note that rounding is applied to all values)

Key Assumptions and Inputs

Physical Parameters

Parameter	Value
Reactor Power Level	3636 MWt (includes uncertainty)
Containment Volume	2.93E+6 ft^3
Containment Leakage	0.21% per day for first 24 hours 0.105% per day after 24 hours
Particulate Removal	3.005E-2 per hour, credit is taken for natural deposition in Containment
RCS Mass	2.258E+8 grams
RCS Volume	1.11E+4 ft^3
SG Mass	391,908 lbm (all 4 SGs)
SG Volume	6280 ft^3 (all 4 SGs)
Coolant Densities	Primary and Secondary water at 62.4 lbm/ft^3
Partition Factors	Iodine PF = 100, Alkali Metals PF = 312 (moisture carryover = 0.32%) Noble Gases PF = 1
Primary to Secondary Leakage	1 gpm total, for the first 20 hours of the accident.
Secondary System Mass releases	
0 – 2 hours	5.5E+5 lbm
2 – 8 hours	1.365E+6 lbm
8 – 20 hours	2.73E+6 lbm

Table 1 – Secondary Side Pathway Flow Rates

Flow Path	Time (hour)		Flow	Note
	From	to		
RCS to SGs	0	20	0.134 cfm	1
RCS to Env (Noble Gases)	0	20	0.134 cfm	1, 2
SGs to Env	0	2	80.8 cfm	3
	2	8	66.8 cfm	
	8	20	66.8 cfm	

Flow Rate Notes:

1. RCS Leakage of 1 gpm – Volumetric leakage (gpm) from RCS is divided by 7.48 gal/ft^3.
2. The noble gases are released without holdup or retention, modeled as a direct release from the RCS to the environment at the primary-to-secondary leakage rate.
3. SGs – Flow is the release (lbm) divided by 62.4 lbm/ft^3 and by the time duration (min).

Radioactivity Considerations

- 0.25% of Fuel Rods experience melting.
- 100% of the noble gases and 25% of the Iodine isotopes (containment) or 50% of the iodine isotopes (secondary side release) within the melting rods are available for release.
- 10% of the fuel rods experience cladding failure. A radial power peaking factor of 1.7 is applied to the damaged rods.
- The fractions of fission product inventory contained within the fuel rod gaps are:
  - Iodine isotopes and Noble gases 0.10
  - Other Halogens 0.05
  - Alkali Metals 0.12
- Core Fission product inventories are taken from an equilibrium cycle based upon a power level of 3636 MWt. To account for potential cycle-to-cycle variations, a 10% fuel design margin is applied.
- Containment Release – 100% of the activity released from the core due to fuel melting and cladding failure and the initial RCS activity is instantaneously released to and uniformly mixed in the containment at the onset of the accident.
- Secondary Side Release – 100% of the activity released from the core due to fuel melting and cladding failure and the initial RCS and secondary side activities are instantaneously mixed within the RCS at the onset of the accident.
- Chemical form of iodine released to containment is 95% particulate, 4.85 elemental, and 0.15% organic.
- Chemical form of iodine released through the SGs is 97% elemental and 3% organic.
- RCS non-iodine activity (alkali metals and other halogens) includes an assumption of normal operations 1% fuel clad defects.
- The radioiodine concentration in the RCS is assumed to be at the Technical

Specification equilibrium limit of 1.0 µCi/gm DE I-131.

- The noble gas concentration in the RCS is based on 280 µCi/gm DE Xe-133.
- The radioiodine concentration in the secondary system is assumed to be at the Technical Specification limit of 0.1 µCi/gm DE I-131.
- The concentrations of Alkali Metals in Secondary are based upon a ratio of the concentration in the RCS: Given 0.1 µCi/gm DE I-131 in the secondary and 1.0 µCi/gm in the RCS, the concentrations of alkali metals in the secondary are assumed to be 10% of those in the RCS.

Containment and RCS Activities

Table 2 reports the total activity released to containment for the containment release case and Table 3 reports the total activity in the RCS for the secondary side release case. Table 4 reports the activity initially in the SGs prior to the event, which is available for release in the secondary side case.

Table 2 – Containment Release Activities

Nuclide	Core Source Term (Ci)	Radial Peaking Factor	Gap Release Fraction	Fraction of Fuel with Clad Damage	Melt Release Fraction	Fraction of Fuel Melted	Fuel Release Source Term (Ci)	RCS Source Term (Ci)	Total Release to Containment (Ci)
Kr-83m	1.29E+07		0.1		1.0		2.740E+05	1.059E+02	2.741E+05
Kr-85	1.12E+06		0.1		1.0		2.384E+04	8.669E+02	2.471E+04
Kr-85m	2.74E+07		0.1		1.0		5.823E+05	1.684E+02	5.824E+05
Kr-87	5.40E+07		0.1		1.0		1.148E+06	1.100E+02	1.148E+06
Kr-88	7.23E+07		0.1		1.0		1.536E+06	3.696E+02	1.536E+06
Xe-131m	1.42E+06		0.1		1.0		3.011E+04	3.292E+02	3.044E+04
Xe-133	2.15E+08		0.1		1.0		4.574E+06	2.594E+04	4.600E+06
Xe-133m	6.83E+06		0.1		1.0		1.452E+05	3.597E+02	1.455E+05
Xe-135	5.10E+07		0.1		1.0		1.084E+06	7.838E+02	1.085E+06
Xe-135m	4.59E+07		0.1		1.0		9.757E+05	4.940E+01	9.757E+05
Xe-138	1.87E+08		0.1		1.0		3.976E+06	6.245E+01	3.976E+06
Br-82	3.41E+05		0.05		0.0		2.898E+03	1.157E+00	2.899E+03
Br-83	1.28E+07		0.05		0.0		1.086E+05	2.230E+01	1.087E+05
Br-84	2.32E+07		0.05		0.0		1.974E+05	1.051E+01	1.974E+05
I-130	2.04E+06	1.70	0.1	0.1	0.25	0.0025	3.676E+04	4.938E+00	3.676E+04
I-131	1.07E+08		0.1		0.25		1.938E+06	1.718E+02	1.938E+06
I-132	1.57E+08		0.1		0.25		2.829E+06	1.759E+02	2.829E+06
I-133	2.20E+08		0.1		0.25		3.972E+06	2.619E+02	3.972E+06
I-134	2.47E+08		0.1		0.25		4.459E+06	3.748E+01	4.459E+06
I-135	2.10E+08		0.1		0.25		3.787E+06	1.445E+02	3.787E+06
Cs-134	1.67E+07		0.12		0.0		3.415E+05	4.491E+02	3.420E+05
Cs-134m	4.50E+06		0.12		0.0		9.176E+04	5.549E+00	9.176E+04
Cs-135	4.51E+01		0.12		0.0		9.200E-01	0.000E+00	9.200E-01
Cs-136	5.37E+06		0.12		0.0		1.096E+05	6.135E+02	1.102E+05
Cs-137	1.18E+07		0.12		0.0		2.399E+05	4.337E+02	2.403E+05
Cs-138	2.04E+08		0.12		0.0		4.167E+06	2.409E+02	4.167E+06
Rb-86	2.07E+05		0.12		0.0		4.230E+03	6.041E+00	4.236E+03
Rb-88	7.36E+07		0.12		0.0		1.501E+06	9.766E+02	1.502E+06
Rb-89	9.64E+07		0.12		0.0		1.966E+06	2.282E+01	1.966E+06

Table 3 – RCS Activities for Secondary Side Release

Nuclide	Core Source Term (Ci)	Radial Peaking Factor	Gap Release Fraction	Fraction of Fuel with Clad Damage	Melt Release Fraction	Fraction of Fuel Melted	Fuel Release Source Term (Ci)	RCS Source Term (Ci)	Total Release to RCS (Ci)
Kr-83m	1.29E+07		0.1		1.0		2.740E+05	1.059E+02	2.741E+05
Kr-85	1.12E+06		0.1		1.0		2.384E+04	8.669E+02	2.471E+04
Kr-85m	2.74E+07		0.1		1.0		5.823E+05	1.684E+02	5.824E+05
Kr-87	5.40E+07		0.1		1.0		1.148E+06	1.100E+02	1.148E+06
Kr-88	7.23E+07		0.1		1.0		1.536E+06	3.696E+02	1.536E+06
Xe-131m	1.42E+06		0.1		1.0		3.011E+04	3.292E+02	3.044E+04
Xe-133	2.15E+08		0.1		1.0		4.574E+06	2.594E+04	4.600E+06
Xe-133m	6.83E+06		0.1		1.0		1.452E+05	3.597E+02	1.455E+05
Xe-135	5.10E+07		0.1		1.0		1.084E+06	7.838E+02	1.085E+06
Xe-135m	4.59E+07		0.1		1.0		9.757E+05	4.940E+01	9.757E+05
Xe-138	1.87E+08		0.1		1.0		3.976E+06	6.245E+01	3.976E+06
Br-82	3.41E+05		0.05		0.0		2.898E+03	1.157E+00	2.899E+03
Br-83	1.28E+07		0.05		0.0		1.086E+05	2.230E+01	1.087E+05
Br-84	2.32E+07		0.05		0.0		1.974E+05	1.051E+01	1.974E+05
I-130	2.04E+06	1.70	0.1	0.1	0.5	0.0025	3.892E+04	4.938E+00	3.892E+04
I-131	1.07E+08		0.1		0.5		2.052E+06	1.718E+02	2.052E+06
I-132	1.57E+08		0.1		0.5		2.996E+06	1.759E+02	2.996E+06
I-133	2.20E+08		0.1		0.5		4.205E+06	2.619E+02	4.206E+06
I-134	2.47E+08		0.1		0.5		4.721E+06	3.748E+01	4.721E+06
I-135	2.10E+08		0.1		0.5		4.010E+06	1.445E+02	4.010E+06
Cs-134	1.67E+07		0.12		0.0		3.415E+05	4.491E+02	3.420E+05
Cs-134m	4.50E+06		0.12		0.0		9.176E+04	5.549E+00	9.176E+04
Cs-135	4.51E+01		0.12		0.0		9.200E-01	0.000E+00	9.200E-01
Cs-136	5.37E+06		0.12		0.0		1.096E+05	6.135E+02	1.102E+05
Cs-137	1.18E+07		0.12		0.0		2.399E+05	4.337E+02	2.403E+05
Cs-138	2.04E+08		0.12		0.0		4.167E+06	2.409E+02	4.167E+06
Rb-86	2.07E+05		0.12		0.0		4.230E+03	6.041E+00	4.236E+03
Rb-88	7.36E+07		0.12		0.0		1.501E+06	9.766E+02	1.502E+06
Rb-89	9.64E+07		0.12		0.0		1.966E+06	2.282E+01	1.966E+06

Tables 2 and 3 notes:

1. Fuel Release Source Term = Core Source Term x Radial Peaking Factor x [(Gap Release Fraction x Fraction of Fuel with Clad Damage) + (Melt Release Fraction x Fraction of Fuel Melted)]
2. RCS source term is based on 1.0 µCi/gm DE I-131 (I-131 to I-135), 280 µCi/gm DE Xe-133 for noble gases (excluding Kr-83m), and 1% defective fuel for other nuclides.
3. Total Activity – This is the sum of the fuel release source term and RCS source term.

Table 4 – Initial SG Activities

Nuclide	Secondary Side Source Term (Ci)
Br-82	9.112E-02
Br-83	1.756E+00
Br-84	8.277E-01
I-130	3.888E-01
I-131	1.352E+01
I-132	1.386E+01
I-133	2.059E+01
I-134	2.948E+00
I-135	1.139E+01
Cs-134	3.536E+01
Cs-134m	4.369E-01
Cs-135	0.000E+00
Cs-136	4.831E+01
Cs-137	3.415E+01
Cs-138	1.897E+01
Rb-86	4.757E-01
Rb-88	7.690E+01
Rb-89	1.797E+00

Table 4 note:

1. Secondary side source term is based on 10% of the equilibrium RCS source term.

Control Room Ventilation Parameters

Parameter	Value
CR Volume	149,000 ft^3
CR Isolation	Automatic at 131 Seconds
CR Pressurization Mode Initiation	Automatic at 211 Seconds
CR Ventilation System Normal Flow Rate	2,575 cfm < 131 seconds
CR Ventilation System Filtered Makeup Rate	1,800 cfm > 211 seconds
CR Ventilation System Recirculation Flow Rate	31,000 cfm > 211 seconds
CR Ventilation System Charcoal Filter Efficiencies (Supply and Recirculation use the same filter)	
All Iodine Species	99%
Particulates	99%
CR Unfiltered In-leakage	180 cfm > 131 seconds
CR Ingress/Egress Unfiltered In-leakage	10 cfm > 0 seconds
CR Breathing Rate	3.5E-4 m^3/sec
Occupancy Factors	
0-24 hours	1.0
1 - 4 days	0.6
4 -30 days	0.4

Atmospheric Dispersion Factors

Table 5 – Atmospheric Dispersion Factors

Time (hr)	X/Q (sec/m^3)				
	EAB*	LPZ	Containment Release to MCR	Secondary Side Release to MCR	Bounding Release & Receptor Location
0 – 2	1.80E-04	7.20E-05	2.22E-03	7.64E-03	U1 N MSIV to U1 MCR
2 – 8	-	3.30E-05	1.55E-03	6.17E-03	U1 N MSIV to U1 MCR
8 – 24	-	2.2E-05	6.57E-04	2.72E-03	U2 N MSIV to U2 MCR
24 – 96	-	9.2E-06	5.80E-04	1.86E-03	U2 N MSIV to U2 MCR
96 – 720	-	2.7E-06	4.47E-04	1.52E-03	U2 N MSIV to U2 MCR

* Applied for entire duration of event in analysis

EAB & LPZ Breathing Rates

0-8 Hours	3.5E-04 m^3/sec
8-24 Hours	1.8E-04 m^3/sec
24 Hours – 30 Days	2.3E-04 m^3/sec

Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Alternative Source Term,
TSTF-51, TSTF-471, and TSTF-490

Attachment 10

Locked Rotor Accident Analysis

LOCKED ROTOR ACCIDENT
DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number: X6CAJ.19, Version 2

Method/Computer Program Used: RADTRAD, Version 3.10

Regulatory Guidance: RG-1.183, including Appendix G

Model Discussion

The calculation was performed to address a Locked Rotor Accident (LRA). The scenario for the LRA is that a reactor coolant pump rotor is postulated to seize, leading to reduced coolant flow and reactor trip. The transient causes fuel damage, resulting in increased radioactivity in the Reactor Coolant System (RCS). Activity in the steam generators (SG) due to primary-to-secondary leakage is released to the environment via steaming until the RCS is placed on RHR cooling.

Results and Acceptance Limits:

Location	Dose (Rem TEDE)	
	Calculated	Limit
EAB	<0.1	2.5
LPZ	<0.1	2.5
Control Room	0.3	5

Key Assumptions and Inputs

Physical Parameters

Parameter	Value
Reactor Power Level:	3636 MWt (includes uncertainty)
RCS Mass:	2.258E8 grams
RCS Volume:	1.11E4 cubic feet
SG Mass:	1.78E8 lbm (all 4 SGs)
SG Volume:	6,281 cubic feet (all 4 SGs)
Secondary System Margin	10% increase is added to mass flows
Coolant Densities:	Primary and Secondary water at 62.4 lbm/ft^3
Partition Factors:	Iodine PF = 100 Alkali Metals PF = 312 (moisture carryover = 0.32%)
Primary to Secondary Leakage:	1 gpm total.

Table 1: Flow Rates After 10% Margin Adjustment

Pathway	Time		Release (lbm)	Flow	Note
	From	To			
RCS to SG	0	8	-	0.134 cfm	1
SG to Environment	0	2	610,500	81.5 cfm	2
	2	8	1,501,500	66.8 cfm	
	8	20	3,003,000	66.8 cfm	

Flow Rate Notes:

1. RCS – Volumetric leakage (1 gallons/minute) from the RCS is divided by 7.48 gal/ft^3.
1. SG – The mass release from the SG is increased by 10% for margin. The flow is then the mass release (lbm) divided by 62.4 lbm/ft^3 and divided by the time duration (min).

<u>Radioactivity Considerations</u>:

- 5% of the fuel rods experience cladding failure. A radial power peaking factor of 1.7 is applied to the damaged rods.
- The fractions of fission product inventory contained within the fuel rod gaps are:
  - I-131 0.08
  - Kr-85 0.10
  - Other Halogens and Noble Gases 0.05
  - Alkali Metals 0.12
- Core Fission product inventories are taken from an equilibrium cycle based upon a power level of 3636 MWt. To account for potential cycle-to-cycle variations, 10% margins are applied to the core inventory.
- 100% of the activity released from the core due to cladding failure is instantaneously mixed within the RCS at the onset of the accident.
- Chemical form of iodine released to the environment from the steam generators is 97% elemental, and 3% organic. The removal mechanism for this pathway is the same for all chemical forms of iodine.
- Radial peaking factor for rods with cladding damage is assumed to be 1.7.
- The initial RCS activity represents normal operations with a 1.0 µCi/g DEI-131 Technical Specification limit, a 280 µCi/g DEXe-133 Technical Specification limit, and 1% clad defects for all other radionuclides.
- The initial radioiodine concentration in the secondary system is assumed to be at the Technical Specification limit of 0.1 µCi/gm DEI
- The initial concentrations of Alkali Metals in Secondary are based upon a ration of the concentration in the RCS: Given 0.1 µCi/g DEI in the secondary and 1.0 µCi/g in the RCS, the concentrations of alkali metals in the secondary are assumed to be 10% of those in the RCS.
- Noble gases are not retained on the secondary side.

RCS Activities

Table 2 - RCS Activities

Nuclide	RG 1.183 Group	Total RCS Source Term (Ci)
Kr-83m	NG	5.490E+04
Kr-85	NG	1.040E+04
Kr-85m	NG	1.166E+05
Kr-87	NG	2.297E+05
Kr-88	NG	3.076E+05
Xe-131m	NG	6.351E+03
Xe-133	NG	9.408E+05
Xe-133m	NG	2.939E+04
Xe-135	NG	2.176E+05
Xe-135m	NG	1.952E+05
Xe-138	NG	7.953E+05
Br-82	H	1.450E+03
Br-83	H	5.435E+04
Br-84	H	9.870E+04
I-130	H	8.654E+03
I-131	H	7.297E+05
I-132	H	6.659E+05
I-133	H	9.348E+05
I-134	H	1.049E+06
I-135	H	8.912E+05
Cs-134	AM	1.712E+05
Cs-134m	AM	4.588E+04
Cs-135	AM	4.600E-01
Cs-136	AM	5.540E+04
Cs-137	AM	1.204E+05
Cs-138	AM	2.084E+06
Rb-86	AM	2.121E+03
Rb-88	AM	7.514E+05
Rb-89	AM	9.832E+05
Sb-124	TG	0.000E+00
Sb-125	TG	0.000E+00
Sb-126	TG	0.000E+00
Sb-127	TG	0.000E+00
Sb-129	TG	0.000E+00
Te-125m	TG	0.000E+00

Nuclide	RG 1.183 Group	Total RCS Source Term (Ci)
Te-127	TG	0.000E+00
Te-127m	TG	0.000E+00
Te-129	TG	0.000E+00
Te-129m	TG	0.000E+00
Te-131	TG	0.000E+00
Te-131m	TG	0.000E+00
Te-132	TG	0.000E+00
Te-133	TG	0.000E+00
Te-133m	TG	0.000E+00
Te-134	TG	0.000E+00
Sr-89	TG	0.000E+00
Sr-90	TG	0.000E+00
Sr-91	TG	0.000E+00
Sr-92	TG	0.000E+00
Ba-137m	TG	0.000E+00
Ba-139	TG	0.000E+00
Ba-140	TG	0.000E+00
Ba-141	TG	0.000E+00
Mo-99	NM	0.000E+00
Pd-109	NM	0.000E+00
Rh-103m	NM	0.000E+00
Rh-105	NM	0.000E+00
Rh-106	NM	0.000E+00
Ru-103	NM	0.000E+00
Ru-105	NM	0.000E+00
Ru-106	NM	0.000E+00
Tc-99	NM	0.000E+00
Tc-99m	NM	0.000E+00
Ce-141	CE	0.000E+00
Ce-143	CE	0.000E+00
Ce-144	CE	0.000E+00
Np-237	CE	0.000E+00
Np-238	CE	0.000E+00
Np-239	CE	0.000E+00
Pu-238	CE	0.000E+00
Pu-239	CE	0.000E+00
Pu-240	CE	0.000E+00
Pu-241	CE	0.000E+00
Pu-242	CE	0.000E+00
Pu-243	CE	0.000E+00

Nuclide	RG 1.183 Group	Total RCS Source Term (Ci)
Am-241	LA	0.000E+00
Am-242	LA	0.000E+00
Am-243	LA	0.000E+00
Cm-242	LA	0.000E+00
Cm-244	LA	0.000E+00
Eu-154	LA	0.000E+00
Eu-155	LA	0.000E+00
Eu-156	LA	0.000E+00
La-140	LA	0.000E+00
La-141	LA	0.000E+00
La-142	LA	0.000E+00
La-143	LA	0.000E+00
Nb-95	LA	0.000E+00
Nb-95m	LA	0.000E+00
Nb-97	LA	0.000E+00
Nb-97m	LA	0.000E+00
Nd-147	LA	0.000E+00
Pm-147	LA	0.000E+00
Pm-148	LA	0.000E+00
Pm-148m	LA	0.000E+00
Pm-149	LA	0.000E+00
Pm-151	LA	0.000E+00
Pr-143	LA	0.000E+00
Pr-144	LA	0.000E+00
Pr-144m	LA	0.000E+00
Sm-151	LA	0.000E+00
Sm-153	LA	0.000E+00
Y-90	LA	0.000E+00
Y-91	LA	0.000E+00
Y-91m	LA	0.000E+00
Y-92	LA	0.000E+00
Y-93	LA	0.000E+00
Y-95	LA	0.000E+00
Zr-95	LA	0.000E+00
Zr-97	LA	0.000E+00

Containment and RCS Activities Notes:

1. The Total RCS source term is the core gap release activity consistent with 5% fuel damage plus the RCS initial activity.

Initial Iodine Activities in the Secondary System

The initial iodine activities in the secondary coolant corresponding to 0.1 µCi/g DEI are shown in the following table.

Table 3 - Initial Iodine Activities in the Secondary System

Isotope	Secondary System Initial Activity (µCi/g)
I-131	7.61E-02
I-132	7.79E-02
I-133	1.16E-01
I-134	1.66E-02
I-135	6.40E-02

Alkali Metals in the Secondary System

The initial concentrations are assumed to be 10% of the RCS initial activities.

Table 4 - Alkali Metals in the Secondary System

Isotope	Concentration (µCi/g)	Activity (Curies)
Cs-134	1.99E-01	3.540E+01
Cs-134m	2.46E-03	4.374E-01
Cs-135	0.00E+00	0.000E+00
Cs-136	2.72E-01	4.836E+01
Cs-137	1.92E-01	3.419E+01
Cs-138	1.07E-01	1.899E+01
Rb-86	2.68E-03	4.762E-01
Rb-88	4.33E-01	7.699E+01
Rb-89	1.01E-02	1.799E+00
Notes:		1

Initial Alkali Metals in the Secondary System Notes:

1. Secondary Activities – the concentrations in Column 2 are multiplied by the mass of 1.78E8 grams and by 1.0E-06 Ci/µCi to obtain the activity in curies in Column 3.

Control Room Ventilation Parameters

CR Volume	149,000 ft^3
CR Isolation	Automatic at 608 Seconds
CR Pressurization Mode Initiation	Automatic at 698 Seconds
CR Ventilation System Normal Flow Rate	2,575 cfm < 608 seconds
CR Ventilation System Filtered Makeup Rate	1,800 cfm > 698 seconds
CR Ventilation System Recirculation Flow Rate	31,000 cfm > 698 seconds
CR Ventilation System Charcoal Filter Efficiencies (Supply and Recirculation use the same filter)	

All Iodine Species	99%
Particulates	99%
CR Unfiltered In-leakage	180 cfm > 131 seconds
CR Ingress/Egress Unfiltered In-leakage	10 cfm > 0 seconds
CR Breathing Rate	3.5E-4 m^3/sec
Occupancy Factors	
0-24 hours	1.0
1 - 4 days	0.6
4 -30 days	0.4

Atmospheric Dispersion Factors

Table 6 – Atmospheric Dispersion Factors

Time (hr)	EAB*	LPZ	MCR	Bounding Release & Receptor Location
0 – 2	1.80E-4	7.20E-5	7.64E-3	U1 N MSIV to U1 MCR
2 – 8		3.30E-5	6.17E-3	U1 N MSIV to U1 MCR
8 – 24		2.20E-5	2.72E-3	U2 N MSIV to U2 MCR
24 - 96		9.20E-6	1.86E-3	U2 N MSIV to U2 MCR
96 - 720		2.70E-6	1.52E-3	U2 N MSIV to U2 MCR

* Applied for entire duration of event in analysis

EAB & LPZ Breathing Rates

0-8 Hours	3.5E-04 m^3/sec
8-24 Hours	1.8E-04 m^3/sec
24 Hours – 30 Days	2.3E-04 m^3/sec

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Attachment 11

VEGP AST Accident Analysis Input Values Comparison Tables

VEGP AST Accident Analysis Input Values Comparison Tables

To facilitate the review and to more readily assess the impact of the adoption of the Alternative Source Term (AST) at Vogtle Nuclear Plant (VEGP), summary tables are provided in this enclosure for each accident being analyzed including a comparison between current licensing basis (CLB) input parameters and the values utilized in the new AST accident analysis, and the basis for any changes. The tables are provided within this enclosure for the following accident scenarios:

Table 2 – Loss of Coolant Accident (LOCA)
Table 3 – Fuel Handling Accident (FHA)
Table 4 – Main Steam Line Break (MSLB) Accident
Table 5 – Steam Generator Tube Rupture (SGTR) Accident
Table 6 – Control Rod Ejection Accident
Table 7 – Locked Rotor Accident

Additionally, Table 1, "Control Room Parameters," is provided to show the parameters of interest for Control Room habitability. In this table, the LOCA parameters are provided as they resulted in the most limiting dose to the Control Room occupants.

Table 1: Control Room Parameters

Input/Assumption	CLB Value	New AST Value	Reason for Change
Control Room Volume	1.72E+05 ft^3	1.49E+05 ft^3	AST reduces CR free volume to account for equipment.
Normal Operation			
Filtered Make-up Flow Rate	0 cfm	0 cfm	No change
Filtered Recirculation Flow Rate	0 cfm	0 cfm	No change
Unfiltered Make-up Flow Rate	3000 cfm	2575 cfm	AST value based on process flow diagrams biased conservatively to account for uncertainty (TS 5.5.11).
Unfiltered In-leakage	0 cfm	0 cfm	No change
Emergency Operation			
Recirculation Mode			
Filtered Make-up Flow Rate	1500 cfm	1800 cfm	AST value based on CREFS process flow diagrams biased conservatively to account for uncertainty (TS 5.5.11).
Filtered Recirculation Flow Rate	19000 cfm	31000 cfm	AST considers dual unit actuation resulting from high radiation signal.
Unfiltered Make-up Flow Rate	0 cfm	0 cfm	No change
Unfiltered In-leakage	140 cfm	190 cfm	The revised value is intended to provide operational margin to the CR measured CR in-leakage. Includes 10 cfm for CR ingress/egress.
Filter Efficiencies			
Pressurization Filters	All iodine99%	All iodine 99%	No change
Recirculation Filters	Elemental – 99% Organic - 99% Particulate – 99%	Elemental – 99% Organic - 99% Particulate – 99%	No change
Particulate	99%	99%	No change
Occupancy			
0-24 hours 1-4 days 4-30 days	100% 60% 40%	100% 60% 40%	No change

Table 1: Control Room Parameters			
Input/Assumption	**CLB Value**	**New AST Value**	**Reason for Change**
Breathing Rate	3.47E-4 m^3/sec (0-720 hr)	3.5E-4 m^3/sec (0-720 hr)	Difference complies with RG-1.183 Rev. 0

Table 2: LOCA Inputs and Assumptions			
Input/Assumption	**CLB Value For Offsite and Control Room**	**New AST Value For Offsite and Control Room**	**Reason for Change**
Containment Purge			
Iodine Chemical Form	5% particulate, 91% elemental, 4% organic	95% cesium iodide, 4.85% elemental, 0.15% organic	Adoption of RG 1.183 methodology.
Containment Volume	2,930,000 ft^3	2,930,000 ft^3	No change
Containment Purge Filtration	0%	0%	No change
Removal by Wall Deposition	None	None	No change
Removal by Sprays	None	None	No change
Containment Leakage			
Iodine Chemical Form	5% particulate, 91% elemental, 4% organic	95% cesium iodide, 4.85% elemental, 0.15% organic	Adoption of RG 1.183 methodology.
Containment Sump pH	>7.0	>7.0	No change
Containment Sprayed Volume	2,300,000 ft^3	2,300,000 ft^3	No change
Containment unsprayed Volume	630,000 ft^3	630,000 ft^3	No change
Containment Spray Start Time	0 seconds	110 seconds	Provides additional conservatism to Containment Leakage Pathway.
Containment Spray Stop Time	2 hours	2 hours	No change
Containment Spray Flow Rate	2500 gpm	2500 gpm	No change
Elemental Iodine Spray Removal Coefficient	10 hr^{-1}	13.7 hr^{-1}	Revision is consistent with RG 1.183 Appendix A RP 3.3
Aerosol Spray Removal Coefficient	4.2 hr^{-1}	5.34 hr^{-1}	Revision is consistent with RG 1.183 Appendix A RP 3.3
Organic Iodine Spray Removal	None	None	No change
Natural Deposition	Elemental, Organic- None, Aerosol – None	Elemental, Organic iodine – None, Aerosols – 0.1 hr^{-1}	Aerosol natural deposition is permitted per Appendix A of RG 1.183.
Containment Leakage Rate 0 to 24 hours 24 hours to 30 days	 0.2%/day 0.1%/day	 0.21%/day 0.105%/day	Additional 5% conservative margin applied
Containment Leakage Filtration	0%	0%	No change
ECCS Leakage to the Auxiliary Building			
Iodine Chemical Form	5% aerosol, 91% elemental, 4% organic	0% aerosol, 97% elemental, 3% organic	The revised percentages are as specified in RG 1.183.
Containment Sump Volume	115,000 ft^3	114,922 ft^3	Lower value without rounding is more conservative due to higher sump concentrations
ECCS Recirculation Start Time	30 minutes	30 minutes	No change
ECCS Leakage Flow Rate	2 gpm	2 gpm	No change
ECCS Flashing Fraction	10%	10%	No change
ECCS Leakage to the RWST (Not Explicitly Modeled in the CLB)			

Table 3: FHA Inputs and Assumptions			
Input/Assumption	**CLB Value For Offsite and Control Room**	**New AST Value For Offsite and Control Room**	**Reason for Change**
Iodine Chemical Form	0% aerosol, 99.75% elemental, 0.25% organic	0% aerosol, 99.85% elemental, 0.15% organic	Chemical composition is as described in RG 1.183 Appendix B Section 2.
Number of Fuel Assemblies Damaged	314 (1 FA + 50 rods in target assembly)	314 Rods (1 FA + 50 rods in target assembly)	No change
Percentage of Fuel Rods Damaged	100%	100%	No change
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU	0	314	100% of rods analyzed with elevated gap fractions for conservatism. Requesting exception for 40% of rods.
Water Level Above Damaged Fuel	23 ft	23 ft	No change
Pool Decontamination Factors	200-Overall	200-Overall	Decontamination Factors are as described in RG 1.183 Appendix B Section 2.
Delay Before Fuel Movement	90 hours	70 hours	AST analyzed at earlier times
Onsite X/Qs Containment 0-2 hrs 2-8 hrs 8-24 hrs Fuel Building 0-2 hrs 2-8 hrs 8-24 hrs	 1.04E-3 sec/m^3 1.04E-3 sec/m^3 1.04E-3 sec/m^3 4.99E-3 sec/m^3 4.99E-3 sec/m^3 4.99E-3 sec/m^3	 N/A N/A N/A 6.01E-3 sec/m^3 4.44E-3 sec/m^3 1.71E-3 sec/m^3	AST did not analyze containment as the release durations being equal the fuel building will be bounding as the X/Qs are nearly 6 times larger throughout the release duration. Regarding the Fuel Building, the AST X/Qs are bounding throughout the entire release duration which is conservative.

Table 4: MSLB Accident Inputs and Assumptions			
Input/Assumption	**CLB Value For Offsite and Control Room**	**New AST Value For Offsite and Control Room**	**Reason for Change**
Maximum Pre-Accident Iodine Spike Concentration	Maximum Value in TS	60 µCi/gm Dose Equivalent I-131	Specified by RG-1.183 Section E.
Concurrent Iodine Spike Appearance Rate	500 X Equilibrium	500 X Equilibrium	No change
Initial Steam Generator Iodine Source Term	0.1 µCi/gf Dose Equivalent Iodine	0.1 µCi/gf Dose Equivalent Iodine	No change
Iodine Chemical Form	100% Elemental	0% aerosol, 97% elemental, 3% organic	The AST chemical form is as provided in RG 1.183 Appendix E, Section 4.
Percentage of Fuel Rods Failed	0%	0%	No change. Note- the MSLB does not result in failed fuel. This is a leaking fuel pre-condition included for conservatism.
RCS Mass	2.53E+08 g	2.258E+08 g	RCS mass as provided by NSSS vendor
Steam Generator Secondary Liquid Mass	1.9E+08 (all 4 SGs)	1.78E+08 g (all 4 SGs)	SG mass as taken from MURPU Engineering report from NSSS vendor
Intact Steam Generator Steam Release	0 – 2 hrs: 4.24E+05 lbm 2 – 8 hrs: 9.6E+05 lbm 8 – 20 hrs: 9.6E+05 lbm	0 – 2 hrs: 4.66E+05 lbm 2 – 8 hrs: 1.06E+06 lbm 8 – 24 hrs: 2.11E+06 lbm	AST added 10% margin to steam releases for conservatism.
Primary-Secondary Leak Rate	0.65 gpm to three intact SGs 0.35 gpm to faulted SG	0.65 gpm to two intact SGs 0.35 gpm to faulted SG	No change
Density Used for Leakage Volume-to-Mass Conversion	62.4 lbm/ft^3	62.4 lbm/ft^3	No change
Duration of Intact SG Tube Uncovery After Reactor Trip	Not modeled-	Not modeled-	Tube uncovery does not occur with intact SGs
Time to Cool RCS to RHR Cut-in	20 hrs	20 hrs	No change

Table 4: MSLB Accident Inputs and Assumptions			
Input/Assumption	**CLB Value For Offsite and Control Room**	**New AST Value For Offsite and Control Room**	**Reason for Change**
Intact Steam Generator Iodine partition factor	100	100	RG 1.183 Appendix E Section 5.5.4 allows an iodine partition factor of 100 for the intact SG.
Intact Steam Generator Moisture Carryover Fraction	Not modeled-Alkali Metals not considered in CLB MSLB source term.	0.32% (Alkali Metal Partition Factor =312)	Carryover is provided for per RG 1.183 Appendix E Section 5.5.4. Moisture carryover is as specified by NSSS vendor report for MURPU.

Table 5: SGTR Accident Inputs and Assumptions			
Input/Assumption	**CLB Value For Offsite and Control Room**	**New AST Value For Offsite and Control Room**	**Reason for Change**
Maximum Pre-Accident Iodine Spike Concentration	60 µCi/gm Dose Equivalent I-131	60 µCi/gm Dose Equivalent I-131	No change
Concurrent Iodine Spike Appearance Rate	500 X Equilibrium	335 X Equilibrium	RG 1.183 Appendix F Section 2.2 allows the 335 factor.
Initial Steam Generator Iodine Source Term	0.1 µCi/gm Dose Equivalent I-131	0.1 µCi/gm Dose Equivalent I-131	No change
Iodine Chemical Form	100% Elemental	0% aerosol, 97% elemental, 3% organic	Iodine chemical form is per RG 1.183 Appendix F Section 4.
Percentage of Fuel Rods Failed	0%	0%	No change. Note- the SGTR does not result in failed fuel. This is a leaking fuel pre-condition included for conservatism.
RCS Mass	2.53E+08 g	2.258E+08 g	AST value as supplied by NSSS vendor.
Steam Generator Secondary Liquid Mass	4.2E+07 g (each)	4.44 E+07 g (each)	AST uses value as specified by NSSS vendor MURPU engineering report.
Intact Steam Generator Steam Release	0 – 2 hours –696500 lbm 2 – 20 hr – 2495600 lbm	0 – 2 hours –696500 lbm 2 – 20 hr – 2495600 lbm	No change
Ruptured Steam Generator Steam Release	0 – 2 hr – 214000 lbm 2 – 20 hr – 37100 lbm	0 – 2 hr – 214000 lbm 2 – 20 hr – 37100 lbm	No change
Time of Reactor Trip	49.7 s	49.7 s	No change
Primary-Secondary Leak Rate	1 gpm	1 gpm	No change
Density Used for Leakage Volume-to-Mass Conversion	62.4 lbm/ft^3	62.4 lbm/ft^3	No change
Ruptured Tube Break Flow	0 – 5502 s – 204900 lbm	0 – 5502 s – 204900 lbm	No change
Break Flow Flashing Fraction	Variable	0.15 Constant	Constant value for AST is conservative compared to the variable flow depicted in FSAR Figure 15.6.3-13
Duration of Intact SG Tube Uncovery After Reactor Trip	0 minutes	0 minutes	No change
Time to Cool RCS to RHR cut in	20 hours	20 hours	No change
Intact Steam Generator Iodine Partition Coefficient	100	100	No change
Intact Steam Generator Moisture Carryover Fraction	CLB source term did not model Alkali Metals	0.32% (Partition factor for Alkali Metals = 312)	Carryover is provided for per RG 1.183 Appendix F Section 5.6 and Appendix E Section 5.5.4. Moisture carryover is as specified by NSSS vendor MURPU engineering report.

Table 6: Control Rod Ejection Accident Inputs and Assumptions			
Input/Assumption	**CLB Value for Offsite and Control Room**	**New AST Value for Offsite and Control Room**	**Reason for Change**
Fuel Rod Gap Fractions	Iodine – 12% Kr_{85} - 10%	Iodine/noble gases – 0.10 Other halogens – 0.05 Alkali metals – 0.12	AST gap fractions are per RG 1.183 Appendix H (iodines and noble gases) and Table 3 (other halogens and alkali metals)
Fuel Rod Peaking Factor	Not provided	1.7	Radial peaking factor is applied per RG 1.183 Section 3.1.
Percentage of Fuel Rods Damaged	10%	10%	No change
Percentage of Fuel That Experiences Melting	0.25%	0.25%	No change
Number of rods exceeding 6.3 kw/ft above 54 GWD/MTU	0	0	No change
Initial RCS Iodine Source Term	60 µCi/gm DE I-131	1.0 µCi/gm DE I-131	RG 1.183 Appendix H does not require a preaccident spike and is silent on initial RCS source term. 1.0 µCi/gm is the maximum equilibrium concentration. Initial RCS concentration is small in relation to fuel release.
Initial RCS non-Iodine Activity	1% defective fuel	1% defective fuel (alkali metals and other halogens) 280 µCi/gm DE Xe-133 (noble gases)	Initial alkali metals based on 1% defective fuel are included in AST. Noble gas based on 280 µCi/gm DE Xe-133 per TSTF-490.
Initial Steam Generator Iodine Source Term	0.1 µCi/gm DE I-131	0.1 µCi/gm DE I-131	No change
Iodine Chemical Form – Secondary Release	Not provided	97% elemental, 3% organic	Iodine chemical form is in accordance with RG 1.183 Appendix H Section 5
Iodine Chemical Form – Containment Release	Not provided	95% aerosol, 4.85% elemental, 0.15% organic	Iodine chemical form is in accordance with RG 1.183 Appendix H Section 4
Containment Volume	2.93E6 ft^3	2.93E6 ft^3	No change
Containment Leakage Rate 0 to 24 hours 24 hours to 30 days	 0.2% 0.1%	 0.201% 0.105%	5% additional margin added to AST analysis for conservatism
Containment Leakage Filtration	0%	0%	No change
Natural Deposition in Containment	50% plateout of RCS release	Elemental iodine – None Aerosols – 3.005E-2 hr^{-1}	Natural deposition is credited per RG 1.183 Appendix H Section 6.1.
Iodine/Particulate Removal by Containment Sprays	Not provided	Not credited	No change
RCS Mass	2.3E+08 grams	2.258E+08 grams	AST value as supplied by NSSS vendor.
Steam Generator Secondary Liquid Mass	1.8E+08 grams (4 SGs)	1.778E+08 grams	AST uses value as specified by NSSS vendor MURPU engineering report.
Primary-Secondary Leak Rate	1 gpm total	1 gpm total	No change
Duration of Primary-to-Secondary Leakage	214 seconds	20 hours	Until shutdown cooling established per RG 1.183 Appendix H Section 7.1.
Density Used for Leakage Volume-to-Mass Conversion	62.4 lbm/ft^3	62.4 lbm/ft^3	No change
Secondary Steam Release	49,000 lbm (total)	0-2 hrs – 5.5E+05 lbm 2-8 hrs – 1.365E+06 lbm 8-20 hrs – 2.73E+06 lbm	Steam releases are assumed until shutdown cooling established per RG 1.183 Appendix H Section 7.1
Duration of SG Tube Uncovery Following Reactor Trip	0 minutes	0 minutes	No change
Steam Generator Iodine Partition Coefficient	100	100	No change

Table 6: Control Rod Ejection Accident Inputs and Assumptions			
Input/Assumption	**CLB Value for Offsite and Control Room**	**New AST Value for Offsite and Control Room**	**Reason for Change**
Steam Generator Moisture Carryover Fraction	Not provided	0.32% (Partition factor for Alkali Metals = 312)	Carryover is provided for per RG 1.183 Appendix H Section 7.4 and Appendix E Section 5.5.4. Moisture carryover is as specified by NSSS vendor MURPU engineering report.

Table 7: Locked Rotor Accident Inputs and Assumptions			
Input/Assumption	**CLB Value For Offsite and Control Room**	**New AST Value For Offsite and Control Room**	**Reason for Change**
Fuel Rod Gap Fractions	10% for all except: Kr-85, I-127, I-129 = 30%	I-131 - 0.08 Kr-85 - 0.10 Other Halogens and Noble Gases - 0.05 Alkali Metals - 0.12 5% Failed Fuel	AST gap fractions are per RG 1.183 Table 3
Fuel Rod Peaking Factor	Not Specified	1.7	Radial peaking factor is applied per RG 1.183 Section 3.1.
Number of rods exceeding 6.3 kw/ft above 54 GWD/MTU	0	0	Per reload requirements assemblies that could exceed 6.3 kW/ft are not loaded in locations where DNB could occur.
Initial Steam Generator Iodine Source Term	0.1 µCi/gm	0.1 µCi/gm	No change
Iodine Chemical Form	100% Elemental	95% particulate 4.85% elemental 0.15% organic	Iodine chemical form is in accordance with RG 1.183 Appendix G Section 5.6.
RCS Mass	2.3 E+08 g	2.258E+08 lbm	RCS mass as supplied by NSSS Vendor
Steam Generator Secondary Liquid Mass	1.9 E+08 g	1.78E+08 lbm	AST uses value as specified by NSSS vendor MURPU engineering report.
Primary-Secondary Leak Rate	1 gpm	1 gpm total	AST value increased for additional conservatism.
Density Used for Leakage Volume-to-Mass Conversion	Not Specified	62.4 lbm/ft^3	No change
Secondary Steam Release	0 – 2 hr 555000 lbm 2 – 8 hr 1365000 lbm 8 – 20 hr 2730000 lbm	0 – 2 hr 610500 lbm 2 – 8 hr 1501500 lbm 8 – 20 hr 3003000 lbm	AST assumes 10% adder on steam flows for conservatism.
Duration of SG Tube Uncovery Following Reactor Trip	0 minutes	0 minutes	No change
Steam Generator Iodine Partition Coefficient	100	100	RG 1.183 Appendix G Section 5.6 allows an iodine partition factor of 100 for SG releases.
Intact Steam Generator Moisture Carryover Fraction	CLB does not consider Alkali Metals in source term	0.32% (partition coefficient of 312 for Alkali Metals)	Carryover is provided for per RG 1.183 Appendix G Section 5.6 and Appendix E Section 5.5.4. Moisture Carryover is as specified in NSSS vendor report for MURPU

Vogtle Electric Generating Plant – Units 1 and 2

License Amendment Request for Alternative Source Term, TSTF-51, TSTF-471, and TSTF-490

Attachment 12

List of Regulatory Commitments

The following table identifies the regulatory commitment in this Attachment to the Enclosure.

REGULATORY COMMITMENT	TYPE		SCHEDULED COMPLETION DATE/EVENT
	One Time	Continuing Compliance	
1. The following guidelines are included in the assessment of systems removed from service during movement of irradiated fuel: • Ventilation system and radiation monitor availability should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the proposed license amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay. • A single normal or contingency method to promptly close containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure; rather the prompt methods should enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.		X	Prior to implementation of the license amendment