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

Joseph M. Farley Nuclear Plant – Units 1 and 2
Alternative Source Term
License Amendment Request

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Southern Nuclear Operating Company (SNC) hereby requests an amendment to the Technical Specifications (TS) for Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2.

SNC requests Nuclear Regulatory Commission (NRC) review and approval of proposed revisions to the licensing basis of FNP that support a full scope application of an Alternative Source Term (AST) methodology. Proposed TS changes, which are supported by the AST Design Basis Accident radiological consequence analyses, are included in this license amendment request (LAR). In addition, the proposed amendment incorporates Technical Specification Task Force (TSTF) Traveler, TSTF-448-A, "Control Room Habitability," Revision 3, and TSTF-312-A, Administrative Control of Containment Penetrations," Revision 1.

Enclosure 1 to this letter contains SNC's evaluation of the proposed changes. Enclosures 2 and 3 provide the markup changes to the Operating License, TS, and the TS Bases (for information). Enclosure 4 provides the clean, retyped Operating License and TS pages. Enclosures 5 through 12 provide additional information in support of this LAR. Enclosure 13 provides the RADTRAD Input/Output files for the Loss-of-Coolant Accident and Fuel Handling Accident in CD format, as well as the ARCON96 input files used to develop the Reactor Water Storage Tank atmospheric dispersion factors.

SNC requests approval of the proposed LAR by November 30, 2017. The proposed changes would be implemented within 120 days of issuance of the amendment.

In accordance with 10 CFR 50.91, a copy of this LAR with enclosures is being provided to the designated Alabama state officials.

This letter contains NRC commitments, as stated in Enclosure 14. If you have any questions, please contact Ken McElroy at 205.992.7369.

A001
NRR

Mr. C. R. Pierce states he is the Regulatory Affairs Director for Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

C R Pierce

C. R. Pierce
Regulatory Affairs Director

Sworn to and subscribed before me this 22 *day of* November*, 2016.*

Laura L. Craft
Notary Public

My commission expires: 10-8-2017

CRP/wrv

Enclosures:
1. Basis for Proposed Change
2. Operating License and Technical Specification Pages (Markup)
3. Bases Pages (Markup) (For information only)
4. Operating License and Technical Specification Pages (Retyped)
5. Regulatory Guide 1.183 Conformance Tables
6. Loss-of-Coolant Accident Analysis
7. Fuel Handling Accident Analysis
8. Main Steam Line Break Accident Analysis
9. Steam Generator Tube Rupture Accident Analysis
10. Control Rod Ejection Accident Analysis
11. Locked Rotor Accident Analysis
12. FNP AST Accident Analysis Input Values Comparison Tables
13. FNP AST LAR Supporting Information
14. Summary of Regulatory Commitments

cc: <u>Southern Nuclear Operating Company</u>
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Ms. C. A. Gayheart, Vice President – Farley
Mr. M. D. Meier, Vice President – Regulatory Affairs
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T. M. Miller, MD, State Health Officer

Joseph M. Farley Nuclear Plant – Units 1 and 2
Alternative Source Term
License Amendment Request

Enclosure 1

Basis for Proposed Change

Enclosure 1

Basis for Proposed Change

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1.0 Summary Description

This evaluation supports a request to revise Operating License NPF-2 and NFP-8 for Joseph M. Farley Nuclear Plant (FNP), 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 FNP that support a selected scope application of an Alternative Source Term (AST) methodology. The proposed amendment also incorporates Revision 3 of Technical Specification Task Force (TSTF) Traveler, TSTF-448-A, "Control Room Habitability," as well as TSTF-312-A, "Administrative Control of Containment Penetrations," into the FNP Technical Specifications (TS).

This application is made to improve two key parameters for which FNP currently has low margin: 1) Emergency Core Cooling System (ECCS) leakage outside containment, and 2) unfiltered in-leakage for the control room. This application will also extend the time for the critical operator action of initiating the pressurization and recirculation mode for the Control Room Emergency Filtration/Pressurization System (CREFS) in a Fuel Handling Accident (FHA).

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 FNP design basis accidents. 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 Criteria (GDC) 19.

2.2 TSTF-448

The proposed amendment would modify TS requirements related to CR envelope habitability in TS 3.7.10, "Control Room" and TS Section 5.5, "Administrative Controls—Programs and Manuals." The changes are consistent with NRC approved Industry/TSTF TS change TSTF-448 Revision 3. The availability of this TS improvement was published in the Federal Register on

January 17, 2007 as part of the consolidated line item improvement process (CLIIP) (Reference 2).

Adoption of TSTF-448 supersedes in its entirety the current licensing basis for the Control Room Integrity Program (CRIP), as established for FNP Units 1 and 2 by License Amendments 166 and 158, respectively (Reference 3). These amendments were issued on September 30, 2004, well before NRC acceptance of TSTF-448 Revision 3 as a basis for CR habitability.

2.3 TSTF-312

The proposed TS change adds a Note to the LCO for TS 3.9.3, "Containment Penetrations," allowing penetration flow path(s) that have direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control.

3.0 TECHNICAL EVALUATION

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 current licensing basis, as described in FNP Final Safety Analysis Report (FSAR) Table 2.3-12. In the AST LOCA analysis, new and revised X/Q values for the CR have been developed to address potential leakage from the Reactor Water Storage Tank (RWST) vent and from the containment mini-purge system. The values resulting at the CR intakes are calculated using the NRC-sponsored computer code ARCON96 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 Enclosure 13.

3.1.1 Meteorological Data

In the AST analyses, two sets of meteorological data were used. The first was a continuous temporally representative four year period of hourly average data from the FNP meteorological tower (January 1, 2000, through December 31, 2003). The second data set included the same 2000 – 2003 information, but also contained an additional six months of derived data from 1999, resulting in a collection span of 4 ½ years. Both sets of data were selected as they have been previously reviewed by the NRC as being of good quality (Reference 5). In Reference 3, the NRC concluded that X/Qs developed from either data set were acceptable if they produced the more limiting result. As shown in Tables 3.4 and 3.5, the 4-year data set generally provided more conservative X/Q results.

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. Usage of the current licensing basis X/Q values for the EAB and the LPZ were approved by the NRC in References 3 and 5.

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	7.6×10^{-4}
LPZ	0 – 2 hours	2.8×10^{-4}
	2 – 8 hours	1.1×10^{-4}
	8 – 24 hours	1.0×10^{-5}
	24 – 96 hours	5.4×10^{-6}
	96 – 720 hours	2.9×10^{-6}

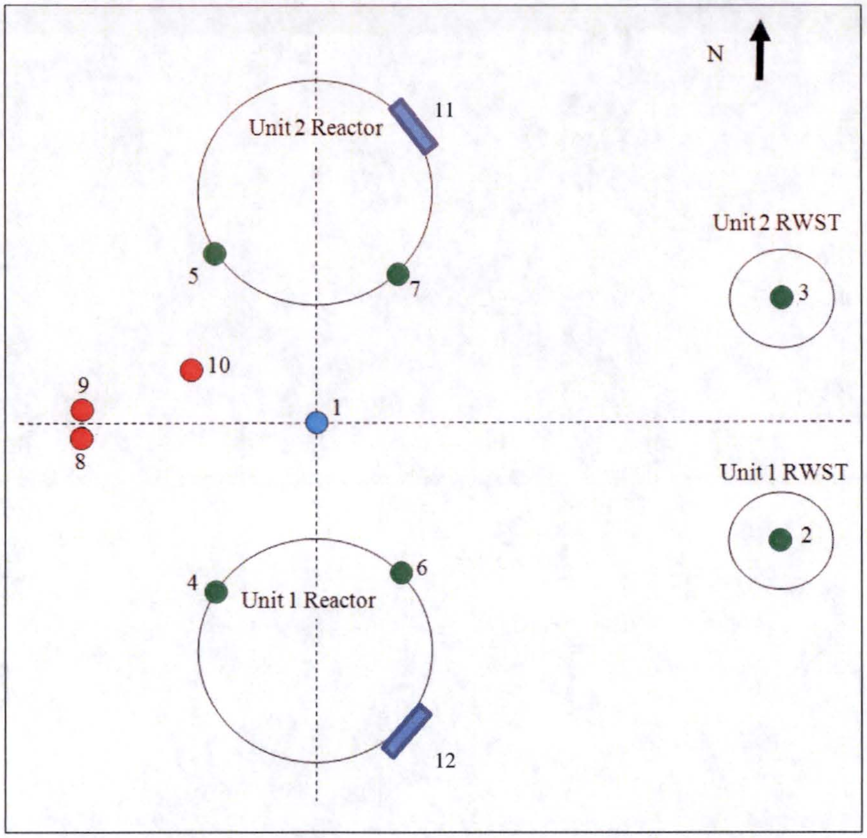
3.1.3 Control Room Atmospheric Dispersion Factors

X/Q factors for onsite release-receptor combinations were developed using the ARCON96 computer code. A number of various release-receptor combinations were considered for the onsite CR atmospheric dispersion factors. These different cases are considered to determine the limiting release-receptor combination for the events. The X/Q factors from the existing calculations of record were used for the Containment Hatch, Reactor, and Plant Vent release points, based on the 4-year and 4 ½-year meteorological data described previously. New X/Q values were developed for the Containment mini-purge and RWST release points for the Loss-of-Coolant Accident (LOCA) based on the 4-year data set.

Figure 3.1 provides a sketch of the general layout of FNP that has been annotated to highlight the onsite release and receptor point locations for the LOCA (the LOCA provided the most limiting receptor-release locations and so were used by the other DBAs for conservatism). All releases are taken as ground level releases per RG 1.194 Position 3.2.1.

Tables 3.2 and 3.3 provide 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 new RWST release point. 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



Location	Description
1	Plant/Reactor Centerline Intersection (Coordinate Origin)
2	Unit 1 RWST Release
3	Unit 2 RWST Release
4	Unit 1 Reactor Release
5	Unit 2 Reactor Release
6	Unit 1 Vent Release
7	Unit 2 Vent Release
8	Unit 1 Control Room Emergency Intake
9	Unit 2 Control Room Emergency Intake
10	Normal Control Room Intake
11	Unit 2 Hatch Door
12	Unit 1 Hatch Door

Table 3.2 - Distance and Geometry of Release and Receptor Locations

Receptor	Release Point	Horizontal Distance (ft)	Vertical Distance (ft)	Horizontal Distance (m)	Vertical Distance (m)	Direction to Source
Unit 1 Control Room Emergency Air Intake	Unit 1 RWST	372	3.5	113	1.07	101
Unit 1 Control Room Emergency Air Intake	Unit 2 RWST	373	3.5	114	1.07	78
Unit 2 Control Room Emergency Air Intake	Unit 1 RWST	373.27	3.5	114	1.07	102
Unit 2 Control Room Emergency Air Intake	Unit 2 RWST	372.38	3.5	113.5	1.07	79
Control Room Normal Air Intake	Unit 1 Vent Reactor*	154.04 111.46	121.5 0.0	46.9 33.9	37.0 0.0	135 159
Control Room Normal Air Intake	Unit 2 Vent Reactor*	121.73 61.61	121.5 0.0	37.1 18.7	37.0 0.0	64 30

* Reactor release heights are conservatively made the same as the normal and emergency Control Room intakes.

Table 3.3 - Elevations of Control Room Air Intakes and Release Points

Location	Elevation (ft)	Elevation (m)	Height above grade (ft)	Height above grade (m)
Emergency Intake	192	58.5	37.5	11.4
Normal Intake	178.5	54.4	24.0	7.3
Vent Stack Release	300	91.4	145.5	44.3
RWST Release	195.5	59.6	41	12.5

Table 3.4a provides the ARCON96 modeling outputs for releases originating at the reactor buildings, reactor vents, hatch doors, and RWST. These Control Room X/Q factors for LOCA, Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Control Rod Ejection, and Locked Rotor Accidents are based on 4 years of meteorological data from years 2000 to 2003, as this data set provided the most limiting values. Table 3.4b provides the X/Q values based on 4 ½ years of data from 1999 to 2003 which was used in the FHA. These atmospheric dispersion factors were previously reviewed by the NRC in References 3 and 5.

Table 3.4a - X/Q Values at the Control Room Air Intakes (4 years meteorological data)

Release Point	Receptor	0 - 2 hours	2 - 8 hours	8-24 hours	1- 4 days	4 - 30 days
U1 Vent	U1 XCR	1.61E-03	1.26E-03	5.65E-04	3.43E-04	2.32E-04
U1 Reactor	U1 XCR	1.51E-03	9.01E-04	3.95E-04	2.75E-04	1.91E-04
U1 Hatch Door	U1 XCR	8.39E-04	5.11E-04	2.09E-04	1.47E-04	9.35E-05
U1 RWST	U1 XCR	4.97E-04	3.69E-04	1.53E-04	1.15E-04	7.98E-05
U2 Vent	U1 XCR	1.64E-03	1.37E-03	7.17E-04	5.41E-04	3.60E-04
U2 Reactor	U1 XCR	1.66E-03	1.36E-03	6.81E-04	5.60E-04	4.21E-04
U2 Hatch Door	U1 XCR	7.95E-04	6.73E-04	3.35E-04	2.48E-04	1.87E-04
U2 RWST	U1 XCR	4.80E-04	3.82E-04	1.70E-04	1.28E-04	9.98E-05
U1 Vent	U2 XCR	1.59E-03	1.25E-03	5.54E-04	3.34E-04	2.27E-04
U1 Reactor	U2 XCR	1.51E-03	8.91E-04	3.91E-04	2.71E-04	1.87E-04
U1 Hatch Door	U2 XCR	8.04E-04	4.90E-04	2.01E-04	1.39E-04	9.04E-05
U1 RWST	U2 XCR	4.95E-04	3.66E-04	1.52E-04	1.14E-04	7.90E-05
U2 Vent	U2 XCR	1.65E-03	1.38E-03	7.20E-04	5.47E-04	3.63E-04
U2 Reactor	U2 XCR	1.65E-03	1.34E-03	6.75E-04	5.40E-04	4.03E-04
U2 Hatch Door	U2 XCR	8.33E-04	6.98E-04	3.43E-04	2.57E-04	1.91E-04
U2 RWST	U2 XCR	4.82E-04	3.82E-04	1.70E-04	1.28E-04	1.00E-04
U1 Vent	Normal CR	2.01E-03	1.46E-03	6.07E-04	3.77E-04	2.59E-04
U1 Reactor	Normal CR	1.56E-03	8.89E-04	3.62E-04	2.72E-04	1.85E-04
U2 Vent	Normal CR	2.79E-03	2.36E-03	1.23E-03	9.18E-04	6.22E-04
U2 Reactor	Normal CR	3.88E-03	3.11E-03	1.38E-03	1.29E-03	1.04E-03

Table 3.4b - X/Q Values at the Control Room Air Intakes (4 ½ years meteorological data)

Release Point	Receptor	0 - 2 hours	2 - 8 hours	8-24 hours	1- 4 days	4 - 30 days
U1 Vent	U1 XCR	1.62E-03	1.21E-03	5.37E-04	3.35E-04	2.32E-04
U1 Reactor	U1 XCR	1.54E-03	9.62E-04	4.45E-04	2.61E-04	3.09E-04
U1 Hatch Door	U1 XCR	8.79E-04	5.89E-04	2.63E-04	1.57E-04	1.93E-04
U2 Vent	U1 XCR	1.59E-03	1.35E-03	7.08E-04	5.16E-04	3.59E-04
U2 Reactor	U1 XCR	1.64E-03	1.34E-03	6.65E-04	5.45E-04	4.18E-04
U2 Hatch Door	U1 XCR	7.83E-04	6.52E-04	3.22E-04	2.43E-04	1.85E-04
U1 Vent	U2 XCR	1.60E-03	1.19E-03	5.23E-04	3.24E-04	2.28E-04
U1 Reactor	U2 XCR	1.54E-03	9.50E-04	4.43E-04	2.70E-04	3.11E-04
U1 Hatch Door	U2 XCR	8.56E-04	5.67E-04	2.50E-04	1.52E-04	1.92E-04
U2 Vent	U2 XCR	1.60E-03	1.37E-03	7.10E-04	5.21E-04	3.60E-04
U2 Reactor	U2 XCR	1.64E-03	1.32E-03	6.60E-04	5.27E-04	4.01E-04
U2 Hatch Door	U2 XCR	8.18E-04	6.77E-04	3.32E-04	2.49E-04	1.89E-04

Notes:
U1 - Unit 1
U2 - Unit 2
XCR – Emergency Control Room Intake
Normal CR – Non-emergency Control Room Intake

3.2 Analytical Models

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

RADTRAD is used to determine the CR and offsite doses for the LOCA and FHA using the source term and X/Q inputs. The code considers the release timing, filtration, hold-up, and chemical form of the nuclides released into the environment.

LocaDose is used to determine the CR and offsite doses for the MSLB, SGTR, Control Rod Ejection, and Locked Rotor Accidents using source term and X/Q inputs. This proprietary Bechtel software calculates radioactive isotope activities within regions, radioactive releases from regions, doses and dose rates within regions for humans and equipment, and inhalation and immersion doses to plant personnel.

ARCON96 (NUREG/CR-6331) was used to determine the X/Qs at the CR intakes for selected release locations from plant meteorological data.

ORIGEN2 was used to calculate plant-specific fission product inventories for use in the LOCA dose calculation.

3.3 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 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.

3.3.1 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) Emergency Core Cooling System (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.

3.3.2 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 Table 3.5a. The calculated dose results are given in Table 3.5b. 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 and LPZ for the duration of the accident, and 5 rem in the CR for the duration of the accident.

Table 3.5a - Parameters and Assumptions for the LOCA

Parameter	Value
ECCS Leakage Initiation Time	20 minutes
ECCS Leakage Iodine Flashing Factor:	10%
Iodine Species ECCS Leakage Released to the Atmosphere	
Elemental	100%
Organic	0%
ECCS Leakage Rate to the RWST	1 gal/min
RWST Leakage Iodine Flashing Factors:	0% to 13.9%
RWST Capacity	505,562 gallons
RWST Volume at Transfer to Recirculation	29,002 gallons
Atmospheric Dispersion Factors (sec/m^3)	

Containment:

Time (hr)	EAB	LPZ	Control Room
0 – 2	7.6E-4	2.80E-4	1.66E-03
2 – 8	-	1.10E-4	1.36E-03
8 – 24	-	1.00E-5	6.81E-04
24 – 96	-	5.40E-6	5.60E-04
96 – 720	-	2.90E-6	4.21E-04

Plant Vent:

Time (hr)	EAB	LPZ	Control Room
0 – 0.0167	7.6E-4	2.80E-4	2.79E-03
0.0167 – 2	7.6E-4	2.80E-4	1.65E-03
2 – 8	-	1.10E-4	1.38E-03
8 – 24	-	1.00E-5	7.20E-04
24 – 96	-	5.40E-6	5.47E-04
96 – 720	-	2.90E-6	3.63E-04

RWST:

Time (hr)	EAB	LPZ	Control Room
0 – 2	7.6E-4	2.80E-4	4.97E-04
2 – 8	-	1.10E-4	3.82E-04
8 – 24	-	1.00E-5	1.70E-04
24 – 96	-	5.40E-6	1.28E-04
96 – 720	-	2.90E-6	1.00E-04

Control Room Parameters

Parameter	Value
Volume	114,000 ft^3
Ventilation System Normal Flow Rate	2340 cfm < 60 seconds
Ventilation System Makeup Rate	375 cfm > 60 seconds
Ventilation System Recirculation Flow Rate	2700 cfm > 60 seconds
Ventilation System Charcoal Filter Efficiencies	
Pressurization Filters	98.5% all iodine species
Recirculation Filters	94.5% elemental and organic 98.5% particulate
Unfiltered In-leakage	325 cfm (includes 10 cfm for CR ingress/egress)
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

Table 3.5b - Calculated LOCA Radiological Consequences

	TEDE (rem)		
	EAB	LPZ	Control Room
Calculated results	13.2	6.0	4.7
Dose acceptance criteria	25	25	5

3.4 Fuel Handling Accident

Two cases are analyzed for the FHA: an accident in Containment and an accident in the spent fuel pool area of the Auxiliary Building.

For the FHA in Containment, the accident occurs in the Refueling Cavity. All gap activity of the damaged fuel rods is assumed to be released instantly into the overlying water. That activity that escapes the overlying water is then assumed to be uniformly distributed throughout the free volume of the Containment above the operating deck (EL 155'-0"). There are two unfiltered release paths from containment: one through the open Equipment Hatch directly to the environment and one through the open Personnel Airlock (PAL) to the Auxiliary Building and the Vent Stack. The release through the open Equipment Hatch is directly to the environment. The release through the open Personnel Airlock credits neither filtration by the Auxiliary Building Ventilation System nor the holdup and dilution in the Exhaust Plenum or Vent Stack. Radioactivity released through the PAL mixes in a portion of the Auxiliary Building on the same level as the CR. Mixing in the Auxiliary Building volume is assumed to be instantaneous. This mixing establishes a pathway for the 10 CFM unfiltered ingress/egress CR in-leakage. Upon detection of the FHA in containment, containment is evacuated and the penetrations open to containment are promptly closed.

Note that the release from containment to the Auxiliary Building through the PAL bounds the release through other containment penetrations into the Auxiliary Building. This is because the other penetrations are on elevations below the control room, and releases through those penetrations would have a tortuous path (through additional mixing volumes and up stairwells) to the area around the CR. Normal auxiliary building heating ventilating and air conditioning (HVAC) systems, which may be running at the time of the FHA, which ventilate the areas containing these other penetrations, and which might not be turned off in the course of the accident, exhaust to the normal auxiliary building plume. This plume is vented to the plant vent, and not to areas around the CR.

High radiation in the CR makeup air intake results in isolation of the CR. Consistent with the FNP current licensing basis, manual operator action is taken to initiate the CREFS pressurization mode. For conservatism, this manual action was extended from 10 minutes to 20 minutes in the FHA analysis.

Releases which pass through the PAL can contaminate the Auxiliary Building at the level of the CR. Doses to operators from the ingress/egress of the CR through the door into the contaminated area have been evaluated and are included in the dose results.

The FHA in the SFP area has also been evaluated. The results from the accident in this area are bounded by the accident in Containment.

The FHA analysis used assumptions and inputs that follow the guidance in RG 1.183. The key parameters and assumptions are listed in Table 3.6a. The calculated dose results are given in Table 3.6b. 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 and 5 rem in the CR for the duration of the accident.

Table 3.6a - Parameters and Assumptions for the FHA

Parameter	Value
Reactor power	2,831 MWt
Fraction of Fission Product Inventory in Gap	
I-131	0.08
Kr -85	0.10
Other Noble Gases	0.05
Other Halogens	0.05

Number of Damaged Fuel Assembly	1
Irradiated Fuel Decay	100 hours
Radial Peaking Factor	1.7
Iodine Chemical Form Release from Fuel to Water	
Elemental	99.85%
Organic	0.15%
Minimum Refueling Cavity and Pool Water Depths	23 feet
Overall Effective Decontamination Factor (DF) for Iodine	200
Chemical Form of Iodine Released from Pool Water	
Elemental	57%
Organic	43%
DF of Noble Gas	1
Duration of Release	2 hours
Activity Release Rate	55,000 cfm

Atmospheric Dispersion Factors (sec/m^3)

Time (hr)	EAB	LPZ	Control Room	
			Vent	Hatch
0 – 2	7.6E-4	2.80E-4	1.62E-3	8.79E-4
2 – 8	-	1.10E-4	1.37E-3	6.77E-4

Control Room Parameters

Parameter	Value
Volume	114,000 ft^3
Ventilation System Makeup Rate	375 cfm
Ventilation System Recirculation Flow Rate	2700 cfm
Ventilation System Charcoal Filter Efficiencies	
Pressurization Filters	98.5% all iodine species
Recirculation Filters	94.5% elemental and organic
Unfiltered Inleakage	600 cfm (Isolation Mode) 325 cfm* (Pressurization Mode)
Breathing Rate	3.5E-4 m^3/sec
Occupancy Factors	
0-24 hours	1.0

*For FHA in containment, 10 cfm for CR ingress/egress is modeled through the CR door from the contaminated auxiliary building. For FHA in the Spent Fuel Pool (SFP) area, 10 cfm for CR ingress/egress is included in the 325 cfm shown above.

Table 3.6b - Calculated FHA Radiological Consequences

	TEDE (rem) EAB	LPZ	Control Room
Calculated results	2.4	0.9	1.0
Dose acceptance criteria	6.3	6.3	5

3.5 Main Steam Line Break Accident

This event consists of a break in one main steam line outside of containment in which the faulted 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 30 µCi/gm. For the concurrent spike case, the initial primary iodine activity concentration is at the equilibrium TS value of 0.5 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 consideration of 1% failed (leaking) fuel, consistent with the FNP current licensing basis.

Leakage from the RCS into all of the SGs, and steam release from the intact SGs, continues until the RCS is cooled to 200 ^{0}F after 24 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. Flows out of the faulted SG are assumed to be released to the environment without partitioning.

The release locations from the faulted and intact SGs are conservatively taken as the most limiting release locations from the LOCA. 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 Table 3.7a. The calculated dose results are given in Table 3.7b. 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, and the duration is until cold shutdown is established.

Table 3.7a - Parameters and Assumptions for the MSLB Accident

Parameter	Value
Steam Releases from Intact SG to Environment	
0 - 2 hours	316,715 lbm
2 - 8 hours	703,687 lbm
8 - 24 hours	948,000 lbm

Intact SG Liquid Iodine Partition Coefficient	100
Steam mass released from faulted SG to the Environment	439,145 lbm
Faulted SG Dryout Time	322.8 seconds
Atmospheric Dispersion Factors (sec/m^3)	

Time (hr)	EAB	LPZ	Control Room
0 – 2	7.6E-4	2.80E-4	1.66E-3
2 – 8		1.10E-4	1.38E-3
8 – 24		1.00E-5	7.20E-4

Control Room Parameters

Parameter	Value
Volume	114,000 ft^3
Ventilation System Makeup Rate	375 cfm
Ventilation System Recirculation Flow Rate	2700 cfm
Ventilation System Charcoal Filter Efficiencies	
Pressurization Filters	98.5% all iodine species
Recirculation Filters	94.5% elemental and organic 98.5% particulate
Unfiltered In-leakage	310 cfm (includes 10 cfm CR ingress/egress)
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

Table 3.7b - Calculated MSLB Accident Radiological Consequences

	TEDE (rem)		
	EAB	LPZ	Control Room
Calculated results			
Pre-Incident Spike	0.94	0.37	0.23
Concurrent Iodine Spike	0.95	0.45	0.45

	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

3.6 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 occurs at a rate equal to 0.35 gpm to the ruptured and 0.65 gpm for both of the intact SGs (1.0 gpm total). Consistent with the FNP current licensing basis, all leakage flow into the ruptured SG is secured after 30 minutes. Leakage into the intact SGs continues until the RCS is cooled to cold shutdown conditions after 8 hours.

A portion of the break and leakage 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 and leakage 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 30 µCi/gm. For the concurrent spike case, the initial primary iodine activity concentration is at the equilibrium TS value of 0.5 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 consideration of 1% failed (leaking) fuel , consistent with the FNP current licensing basis.

The release locations from the faulted and intact SGs are conservatively taken as the most limiting release locations from the LOCA. 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 Table 3.8a. The calculated dose results are given in Table 3.8b. 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 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, and the duration is until cold shutdown is established.

Table 3.8a - Parameters and Assumptions for the SGTR Accident

Parameter	Value
Reactor Coolant Activity (Initial)	
Pre-Accident Iodine Spike	30 µCi/gm DE 1-131
Accident-Initiated Iodine Spike	0.5 µCi/gm DE 1-131
Noble Gas	1% failed (leaking) fuel
Concurrent Iodine Spiking Factor	335

Duration of Intact SG Flow	8 hours
Secondary Coolant Iodine Specific Activity	0.1 µCi/gm DE 1-131
Atmospheric Dispersion Factors (sec/m^3)	

Time (hr)	EAB	LPZ	Control Room
0 – 2	7.6E-4	2.80E-4	1.66E-3
2 – 8	-	1.10E-4	1.38E-3

Control Room Parameters

Parameter	Value
Volume	114,000 ft^3
Ventilation System Makeup Rate	375 cfm
Ventilation System Recirculation Flow Rate	2700 cfm
Ventilation System Charcoal Filter Efficiencies	
Pressurization Filters	98.5% all iodine species
Recirculation Filters	94.5% elemental and organic 98.5% particulate
Unfiltered In-leakage	310 cfm (includes 10 cfm CR ingress/egress)
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

Table 3.8b - Calculated SGTR Accident Radiological Consequences

	TEDE (rem) EAB	TEDE (rem) LPZ	Control Room
Calculated results			
Pre-Incident Spike	2.4	**0.92**	0.48
Concurrent Iodine Spike	0.82	0.34	0.17

	TEDE (rem) EAB	TEDE (rem) 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 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: 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 are also assumed to be located in the fuel rod gap.

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 50% 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. Activity is released from containment at the TS leak rate limit for the first 24 hours and at half that rate after that. The release of iodine initially present in the SG secondary side is also included.

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 2500 seconds. In keeping with previous evaluations of the Control Rod Ejection accident, the Secondary System mass releases to the environment last for 98 seconds. With the large amount of fission products introduced into the reactor coolant by failed fuel, the initial activity of the RCS prior to the event is not considered. However, the dose contribution from the iodine activity initially present in the SG secondary is included in the analysis.

The release locations are conservatively taken as the most limiting release locations from the LOCA. 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 Table 3.9a. The calculated dose results are given in Table 3.9b. 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 duration is 30 days for the Containment pathway, and until cold shutdown is established for the secondary pathway.

Table 3.9a - Parameters and Assumptions for the Control Rod Ejection Accident

Parameter	Value
Reactor power	2831 MWt
Post-accident damaged fuel	10%
Percentage of Melted Fuel Release	
Containment Leakage	
Iodine	50%
Noble Gases	100%
Primary-to-Secondary Leakage	
Iodine	50%
Noble Gases	100%
Iodine Chemical Form Release to Containment	
Aerosol (cesium iodide)	95%
Elemental	4.85%
Organic	0.15%
Containment Leak Rates	
0-24 hours	0.15 weight %/day
> 24 hours	0.075 weight %/day
Primary-to-Secondary Leak Duration	2500 seconds
RCS Leakage	1 gpm
SG Liquid Iodine Partition Coefficient	100
Steam Releases from Intact SG to Environment	426,000 lbm
Atmospheric Dispersion Factors (sec/m^3)	

Time (hr)	EAB	LPZ	Control Room
0 – 2	7.6E-4	2.80E-4	1.66E-3
2 – 8	-	1.10E-4	1.38E-3
8 – 24	-	1.00E-5	7.20E-4
24 – 96	-	5.40E-6	5.60E-4
96 – 720	-	2.90E-6	4.21E-4

Control Room Parameters

Parameter	Value
Volume	114,000 ft^3
Ventilation System Makeup Rate	375 cfm
Ventilation System Recirculation Flow Rate	2700 cfm
Ventilation System Charcoal Filter Efficiencies	
Pressurization Filters	98.5% all iodine species
Recirculation Filters	94.5% elemental and organic 98.5% particulate
Unfiltered In-leakage	310 cfm (includes 10 cfm CR ingress/egress)
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

Table 3.9b - Calculated Control Rod Ejection Accident Radiological Consequences

	TEDE (rem)		
	EAB	LPZ	Control Room
Calculated results	3.8	2.7	3.7
Dose acceptance criteria	6.3	6.3	5

3.8 Locked Rotor Accident

The Locked Rotor Accident dose analysis is defined by the 20% 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 8 hours, by which time the RCS temperature is cooled to cold shutdown conditions.

Since the quantity of the fission products released from the failed fuel dominates the RCS activity during the event, the initial nuclide concentration in the RCS prior to the event is not considered. However, the analysis does include 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 LOCA. The analysis assumes that the CR isolates and enters the emergency ventilation mode at the onset of the accident. For conservatism, an assessment is being performed for a delayed manual CREFS initiation. Results of this assessment are expected to be within 5 rem TEDE.

The analysis used assumptions and inputs that follow the guidance in RG 1.183. The key parameters and assumptions are listed in Table 3.10a. The calculated dose results are given in Table 3.10b. 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 is 30 days for the Containment pathway, and until cold shutdown is established for the secondary pathway.

Table 3.10a - Parameters and Assumptions for the Locked Rotor Accident

Parameter	Value
Reactor power	2831 MWt
Post-Locked Rotor Accident	20%
Secondary Coolant Iodine Specific Activity	0.1 μCi/gm DE 1-131

Fraction of Fission Product Inventory in Gap	
1-131	0.08
Kr-85	0.10
Other Halogens and Noble Gases	0.05
Alkali Metals	0.12
RCS Leakage	1 gpm
SG Liquid Iodine Partition Coefficient	100
Iodine Release from SG	
Elemental	97%
Organic	3%
Steam Releases from SG to Environment	
0-2 hours	512,325 lbm
2 - 8 hours	833,221 lbm

Atmospheric Dispersion Factors (sec/m^3)

Time (hr)	EAB	LPZ	Control Room
0 – 2	7.6E-4	2.80E-4	1.66E-3
2 – 8	-	1.10E-4	1.38E-3

Control Room Parameters

Parameter	Value
Volume	114,000 ft^3
Ventilation System Makeup Rate	375 cfm
Ventilation System Recirculation Flow Rate	2700 cfm
Ventilation System Charcoal Filter Efficiencies	
Pressurization Filters	98.5% all iodine species
Recirculation Filters	94.5% elemental and organic 98.5% particulate
Unfiltered In-leakage	310 cfm (includes 10 cfm CR ingress/egress)
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

Table 3.10b - Calculated Locked Rotor Accident Radiological Consequences

	TEDE (rem)		
	EAB	LPZ	Control Room
Calculated results	1.2	0.83	<5*
Dose acceptance criteria	2.5	2.5	5

* The actual CR dose is not reported in the FNP FSAR for this event.

3.9 Conclusions

The proposed changes provide a source term for FNP that will result in a more accurate assessment of the DBA radiological doses. The results from all of the dose analyses show that the predicted dose consequence results are within the allowable regulatory limits. The revised radiological dose to the CR occupants allows for a revised unfiltered air in-leakage assumption that provides a conservative margin over that determined by air in-leakage testing.

SNC has assessed the dose to CR occupants during CR ingress/egress, such as with shift turnover activities. Consistent with Standard Review Plan 6.4, 10 CFM of unfiltered in-leakage is included for CR ingress/egress in the dose analyses to account for this. However, SNC recognizes the Environmental Protection Agency (EPA) guidance which limits the dose to an emergency worker during nuclear incidents to 5 rem (which includes shift turnover transit time from the EAB to the CR). Consistent with the FNP current licensing basis, controls are in place during radiological events to limit dose to emergency workers. In general, for any accident where a release is in progress, radiological protection personnel are monitoring the dose rates onsite in various areas, including offsite and in the control room. The Emergency Response Organization has specific members that are monitoring the release and the direction of the plume. It is not permitted for any relief workers to enter the plant from the direction of the plume, and especially not in their own vehicles. SNC radiological protection staff will bring the relief workers to the plant in buses/vans, per the Emergency Plan. Masks and or respirators will be provided, as needed, to assure workers dose is as low as reasonably achievable. For additional conservatism, CR occupants will ingress/egress via the secondary CR door during an FHA. Given these controls and protection factors afforded by emergency equipment, the doses to CR occupants during shift turnover ingress and egress will be within the EPA dose guidance for emergency workers.

3.10 TS Discussion

3.10.1 TSTF-448

With License Amendment 166/158 for FNP Units 1/2 (Reference 3), the NRC approved a new section TS 5.5.18, "Control Room Integrity Program," to the Programs and Manuals Section of the TS. As stated in the Safety Evaluation (SE) for that amendment:

> "The CRIP represents the manner in which the licensee will demonstrate CRE [Control Room Envelope] integrity. SNC has proposed a CRIP which incorporates the FNP design and licensing-basis details. It has been formulated following numerous discussions with the NRC staff. The proposal reflects the evolution of the NRC staff's guidance on the CRIP from that which was presented in RG 1.196. This guidance is current as of the date this SE was issued."

However, the issuance date of License Amendment 166/158 (September 30, 2004) predated NRC approval of TSTF-448 Revision 3 (January 17, 2007).

Accordingly, the proposed adoption of TSTF-448 will supersede the current licensing basis for the CRIP.

3.10.1.1 Applicability of Published Safety Evaluation

SNC has reviewed the safety evaluation dated January 17, 2007 as part of the Consolidated Line Item Improvement Process. This review included a review of the NRC staff's evaluation, as well as the supporting information provided to support TSTF–448. SNC has concluded that the justifications presented in the TSTF proposal and the safety evaluation prepared by the NRC staff are applicable to FNP Units 1 and 2 and justify this amendment for the incorporation of the changes to the FNP TS.

3.10.1.2 Optional Changes and Variations

The model Safety Evaluation and model application provided optional statements and evaluations to accommodate variations in plant design and licensing basis. For the purposes of the FNP Unit 1 and 2 TSs, the optional statements and evaluations provided on Tables 3.11a and 3.11b below are applicable.

Table 3.11a - Options With the Model Safety Evaluation

Number	Model SE Location	SE Statement	FNP Option	Justification
1	Throughout SE	Model SE uses the term "[CREEVS]" defined as "[Control Room Envelope Emergency Ventilation System]" to describe the CR ventilation system.	The FNP term is "CREFS" which is defined as the "Control Room Emergency Filtration/ Pressurization System."	This is an option authorized by the SE.
2	Throughout SE	Model SE offers the term "[trains]" or "[subsystem]" when referring to CREEVS.	FNP uses "trains" when referring to CREFS.	This is an option authorized by the SE.
3	Throughout SE	Model SE describes the CRE administrative control program as "[Specification 5.5.18, "CRE Habitability Program]"	SNC confirms that bracketed designation.	This is an SE confirmation.
4	Section 1.0, first paragraph	Model SE includes the term "[Name of the Licensee]."	The facility name is "Farley Nuclear Plant, Units 1 and 2."	This information authorized by the SE.
5	Section 2.2, second paragraph	Model SE provides the options of "[5 rem whole body dose or its equivalent to any part of the body]" or "[5 rem total effective dose equivalent (TEDE)]."	With this AST LAR, FNP will be using 5 rem TEDE.	This is an option authorized by the SE.
6	Section 2.3, first paragraph	Model SE provides an option for facilities not licensed under the 10 CFR 50 General Design Criteria.	FNP was licensed under the 10 CFR 50 General Design Criteria.	This option is not being used by FNP.
7	Section 3.0	Model SE provides the following bracketed statement: "[The emergency operational mode of the [CREEVS] at [facility name][pressurizes] [isolates but does not pressurize] the CRE to minimize unfiltered air in-leakage].	The FNP CREFS uses pressurization. The entire statement should read: "The emergency operational mode of the CREFS at Farley Nuclear Plant, Units 1 and 2 pressurizes the CRE to minimize unfiltered air in-leakage.	This is an option authorized by the SE.

Number	Model SE Location	SE Statement	FNP Option	Justification
8	Section 3.1, first paragraph	Model SE states: "[Except for plant specific differences, all of] these changes are consistent with STS as revised by TSTF-448, Revision 3."	There are no plant-specific exceptions being taken from the Standard Technical Specifications text, as revised by TSTF-448.	This option is not being used by FNP.
9	Section 3.1, second paragraph	Model SE describes the TS Bases Control Program as being "TS 5.5.[11]."	The FNP TS Bases Control Program is described in TS 5.5.14, not TS 5.5.11.	This is an option authorized by the SE.
10	Section 3.2	Model SE describes an option to correct a typographical error by replacing "irradiate" with "irradiated."	FNP TS 3.7.10 Condition E does not contain this typographical error, and so is not utilizing this option.	This option is not being used by FNP.
11	Section 3.3	Model SE provides six evaluation options	SNC has selected Evaluation 1 as that which is most applicable to the FNP technical specifications.	This is an option authorized by the SE.
12	Section 3.3, paragraph below end of Evaluation 6	Model SE describes an option of not following the ASTM E741 testing methodology, and providing an alternative methodology that has been accepted by the NRC.	SNC proposes to follow the air in-leakage testing methodology described in that paragraph.	This option is not being used by FNP.
13	Section 3.4, first paragraph	Model SE provides the option radiation exposures in units of "whole body or its equivalent to any part of the body" or "total effective dose equivalent."	With this AST LAR, radiation exposures will be in units of TEDE.	This is an option authorized by the SE.
14	Section 3.4, fifth paragraph	Model SE provides an option for taking exception(s) to Sections C.1 and C.2 of RG 1.197.	SNC is taking no exceptions to Section C.1 and C.2 of RG 1.197.	This option is not being used by FNP.
15	Section 3.4, sixth paragraph	Model SE provides a bracketed frequency interval of 18 months for CRE pressure testing.	FNP is on a 24-month fuel cycle, so the frequency interval will be 24 months.	This is an option authorized by the SE.
16	Section 4	Model SE provides a bracket to describe the State applicable to the State official.	"Alabama" is the applicable State for the State official.	This is an option authorized by the SE.

Table 3.11b - Variances With the Model Safety Evaluation

Number	Model SE Location	SE Statement	FNP Variation	Justification
1	Section 1.0, last paragraph, and Section 3.1, first paragraph	Model SE describes TS 3.7.10 as "Control Room Envelope Emergency Ventilation System (CREEVS)."	The FNP title of TS 3.7.10 is "Control Room" which is being changed to "Control Room Emergency Filtration/ Pressurization System (CREFS). This is a change not specifically addressed by the TSTF.	With License Amendments 166/158, the title of TS 3.7.10 was changed from "CREFS" to "Control Room" and established CRE operability as distinct from CREFS operability. This LAR is restoring the previous title and establishing the CRE as a subsystem for CREFS operability. The title change itself is an administrative change.
2	Section 1.0, last paragraph, and Section 3.1, first paragraph	Model SE describes TS 5.5.18 as a new administrative controls program.	FNP has an existing TS 5.5.18 Control Room Integrity Program (CRIP) that is being replaced in its entirety by the TSTF-448 Control Room Envelope Habitability program. Therefore, the model SE should change the word "new" to "revised."	The existing TS 5.5.18 for CRIP was introduced with License Amendment 166 and 158 for FNP Units 1 and 2 respectively. These amendments for issued on September 30, 2004, well before NRC acceptance of TSTF-448 Revision 3 as a basis for Control Room habitability. The CRIP requirements are comparable to the CRE Habitability program, except the CRIP is more prescriptive in the testing requirements for CR in-leakage, and cites specific acceptance criteria, rather than referring to the values included in the licensing basis DBA analyses. The CRIP also more stringently defines the CRE

Number	Model SE Location	SE Statement	FNP Variation	Justification
				configuration control program. The TSTF-448 CRE Habitability Program provides a technically justified alternative to the existing CRIP and provides reasonable assurance of public health and safety.
3	Section 3.1, Proposed Changes	N/A	SNC is making several changes to TS 3.7.10 to make it comport to the Westinghouse Owners Group (WOG) STS, as revised by TSTF-448, but that are not specifically addressed by the TSTF: 1) Revising LCO 3.7.10 from "Two Control Room Emergency Filtration/Pressurization System (CREFS) trains and the Control Room Envelope (CRE) shall be OPERABLE" to "Two CREFS trains shall be OPERABLE." 2) Revise the Required Actions of B.2.1 from B.2.1 (Restore CRE to OPERABLE status) <u>OR</u> B.2.2.1 (Verify GDC 19 is met) <u>AND</u> B.2.2.2 (Restore CRE to OPERABLE status), to B.2 (Verify mitigating actions) <u>AND</u> B.3 (Restore CRE boundary to OPERABLE status).	FNP Amendment 166/158 established the CRE as a separate entity from CREFS with its own operability requirements. The proposed change to treat the CRE as a subsystem for CREFS operability, consistent with the WOG STS, has no adverse impact on how these SSCs are controlled. The changes to Required Actions B.2 effectively eliminate the option of restoring CRE to operability in 24 hours, in lieu of verifying the effectiveness of the mitigating actions. Therefore this change is more restrictive. Specific SRs to verifying CRE Δp is within limits and require CRE integrity testing is not necessary, as these provisions are in the TS 5.5.18 CRE Habitability Program.

Number	Model SE Location	SE Statement	FNP Variation	Justification
			3) Revising SR 3.7.10.4 from "Verify CRE Δp within limits in the CRIP" to the new wording of the TSTF. 4) Deleting SR 3.7.10.5 which verifies CRE integrity in accordance with the CRIP.	
4	Section 3.3, first sentence of Evaluation 1	Model SE describes Evaluation 1 as being for facilities that have adopted the [CREEVS] TS LCO Note and Action B of TSTF-287, Rev. 5.	FNP has not adopted TSTF-287, but has a Note that is similar, although Action B is dissimilar.	Of the six Evaluation options, Evaluation 1 is the most applicable to the FNP TS.
5	Section 3.3, Evaluation 1, first paragraph	The model SE states: "The licensee propose to revise the action requirements of TS [3.7.10, "CREEVS,"] to acknowledge that an inoperable CRE boundary, depending upon the location of the associated degradation, could cause just one, instead of both [CREEVS] [trains] to be inoperable."	TS 3.7.10 for FNP is organized differently than the WOG STS described in TSTF-448. The model SE should read: "The licensee propose to revise the action requirements of TS 3.7.10 to establish that one or more CREFS trains will be inoperable due to an inoperable CRE boundary."	As discussed in Item 3 above, Amendment 166/158 established separate operability requirements for the CRE and for CREFS, which resulted in a Condition B that only covered an inoperable CRE. Consistent with the WOG STS, as revised by TSTF-448, the CRE will be a subsystem required for CREFS operability. This has no adverse impact on how these SSCs are controlled.

Number	Model SE Location	SE Statement	FNP Variation	Justification
6	Section 3.3, Evaluation 1, second paragraph	The model SE states: "This change clarifies how to apply the action requirements in the event just one CREFS train is unable to ensure CRE occupant safety...because of an inoperable CRE boundary."	As stated in Item 5 above, TS 3.7.10 for FNP is organized differently than the WOG STS described in TSTF-448. The model SE should read: "This change clarifies how to apply the action requirements in the event that one or both CREFS trains are unable to ensure CRE occupant safety...because of an inoperable CRE boundary."	As discussed in Item 3 above, Amendment 166/158 established separate operability requirements for the CRE and for CREFS, which resulted in a Condition B that only covered an inoperable CRE. Consistent with the WOG STS, as revised by TSTF-448, the CRE will be a subsystem required for CREFS operability. This has no adverse impact on how these SSCs are controlled.
7	Section 3.3, Evaluation 1, first paragraph, Condition B bullet	The model SE states: "One or more [CREEVS][trains] inoperable due to inoperable CRE boundary in MODE 1, 2, [or] 3[, or 4]."	Should read: "One or more CREFS trains inoperable due to inoperable CRE boundary."	For FNP, TS 3.7.10 is applicable in MODES 1, 2, 3, and 4, during movement of irradiated fuel assemblies, and during Core Alterations." Condition B remains applicable in all those operating states.
8	Section 3.3, Evaluation 1, third paragraph	The model SE states: "The licensee proposes to replace existing Required Action B.1, 'Restore control room boundary to OPERABLE status,' which has a 24-hour Completion Time..."	TS 3.7.10 for FNP is organized differently than the WOG STS described in TSTF-448. The model SE should read: "The licensee proposes to replace existing Required Action B.1, 'Initiate mitigating actions,' which has an immediate Completion Time...	This is considered to be an editorial change.

3.10.1.3 License Condition Regarding Initial Performance of New Surveillance and Assessment Requirements

SNC proposes the following as a license condition to support implementation of the proposed TS changes:

Upon implementation of Amendment No. xxx adopting TSTF–448, Revision 3, the determination of control room envelope (CRE) unfiltered air in-leakage as required by SR 3.7.10.4, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.18.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.18.d, shall be considered met. Following implementation:

(a)The first performance of SR 3.7.10.4, in accordance with Specification 5.5.18.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003–01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

3.10.2 TSTF-312

3.10.2.1 Summary of the Approved Traveler Justification

The proposed TS change adds a Note to the Limiting Condition for Operation (LCO) for TS 3.9.3, "Containment Penetrations," allowing "Penetration flow path(s) that have direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control." The Applicability for LCO 3.9.3 is during core alterations, and during movement or irradiated fuel assemblies within containment.

The changes proposed in TSTF-312-A, Revision 1, are consistent with those in Specification 3.6.3, "Containment Isolation Valves." TS 3.6.3, Actions Note 1, allows penetration flow path(s) (except for the 24 inch purge valves) to be unisolated intermittently under administrative control, and is Applicable in MODES 1, 2, 3, and 4. Under the applicable conditions for LCO 3.6.3, the accident analyses credit the primary containment as a release barrier. The proposed change to LCO 3.9.3 would be Applicable under significantly lower energy conditions than those that apply for LCO 3.6.3, and is therefore less risk significant. Adoption of this change is proposed to provide a consistent approach to containment boundary issues that utilizes previously approved and acceptable compensatory measures.

The proposed change also includes the addition of text to the LCO discussion in Bases 3.9.3 stipulating that the administrative controls that are put in place when penetrations flow path(s) are unisolated ensure that: 1) appropriate personnel are aware of the open status of the penetration flow path during core alterations or movement of irradiated fuel assemblies within the containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of an FHA.

TSTF-312-A includes a Reviewer's Note that identifies the need for a confirmatory FHA dose calculation that has been accepted by the NRC staff, and that indicates acceptable radiological consequences. This TSTF is predicated upon NRC acceptance of this AST LAR for the FHA dose consequences.

The Reviewer's Note identifies licensee commitments to implement administrative procedures that ensure the open containment airlock can be promptly closed in the event of an FHA following personnel evacuation, and that open penetration flow path(s) can be promptly closed. The Reviewer's Note also identifies that the time to close such penetrations, or combination of penetrations, should be included in the confirmatory dose calculations.

The Farley FHA dose calculation analyzes offsite and control room doses for FHA events within containment, and evaluates scenarios where the equipment hatch and/or personnel airlocks are open. Doses are calculated for the 0-2 hour period, and essentially all of the activity that is released into containment by the FHA event is released from containment during these 2 hours. There is fundamentally no contribution to calculated doses after the first 2 hours of the event.

Due to the fact that essentially all of the activity that is released into containment during the inside containment FHA is assumed to be released during the first 2 hours of the event, calculated doses for a release through an open equipment hatch and/or open personnel airlock doors are very nearly the same. Although prompt closure of the equipment hatch and the personnel airlock can be achieved, no credit for these actions are taken in the dose calculation.

Similarly, offsite and control room doses resulting from either a simultaneous or individual release through one or more open containment penetrations, an open equipment hatch, or open personnel airlock doors will be very nearly the same as those calculated for an open equipment hatch and/or open personnel airlock doors, and will also be within acceptance limits without assuming that any leak paths are isolated. Consequently, it is not necessary to provide the time to close unisolated containment penetrations(s) in the FHA dose calculations.

Allowing penetration flow paths to be unisolated during core alterations or movement of irradiated fuel will not invalidate the conclusion that he potential dose consequences from a FHA are within 10 CFR 50.67 limits.

SNC will establish administrative controls to ensure: 1) appropriate personnel are aware of the open status of the penetration flow path(s) during core alterations or movement of irradiated fuel assemblies within the containment, and 2) specified individuals are designated and readily available to isolate any open penetration flow path(s) in the event of an FHA inside containment.

3.10.2.2 Differences Between the Proposed Change and the Approved Traveler

The Farley Section 3.9 specification numbers are different from the Improved Standard Technical Specification (ISTS) Section 3.9 specification numbers. Farley Specification 3.9.3, "Containment Penetrations," is equivalent to Specification 3.9.4 in the ISTS. This has no effect on the requested change.

Farley LCO 3.9.3.b was previously amended to allow the personnel and equipment airlocks to remain open during core alterations or movement of irradiated fuel assemblies within the containment, provided one airlock door was available and a designated individual was available to close the open airlock door(s) if needed. The scope of this previous amendment overlaps the scope of TSTF-312-A, and as a result LCO 3.9.3 and its associated Bases differ from that presented in TSTF-312-A. The Note for LCO 3.9.3 and the supplemental LCO text for Bases 3.9.3 are incorporated without change from TSTF-312-A. No additional changes to the LCO and Bases were necessary of made as a result of the existing allowance for the personnel and equipment airlock.

4.0 Regulatory Safety Analysis

4.1 Applicable Regulatory Requirements/Criteria

Title 10 Code of Federal Regulations Section 50.36. "Technical specifications"

Changes to the FNP TSs are proposed for the adoption of TSTF-448. A description of these proposed changes and their relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published in Reference 2, and TSTF-448, Revision 3.

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 licensees 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-312

Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criteria:

Criterion 56, Primary containment isolation, states:

> Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lies, are acceptable on some other defined basis:
>
> (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
>
> (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or
>
> (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
>
> (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as he automatic isolation valve outside containment.
>
> Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic

isolation valves shall be designed to take the position that provides greater safety.

The proposed change to LCO 3.9.3 will allow containment penetration flow path(s) to be open during refueling operations under administrative control. This change does not significantly change how the plant would mitigate an accident previously evaluated, and is bounded by existing FHA accident analysis.

In conclusion, based on the considerations discussed above, (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 approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

4.2 Precedent

Although a number of alternative 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, no specific precedent submittals are referenced herein.

TSTF-312, Revision 1, was approved by the NRC as documented in a letter from William Beckner (NRC) to James Davis (NEI), dated August 16, 1999 (CAN 9908250220). TSTF-312-A, Revision 1, has been adopted by many plants as part of a complete conversion to the ISTS, such as North Anna Power Station (CAN ML0212110540). An example of a plant-specific NRC approval of the changes in TSTF-312-A, Revision 1, is Arkansas Nuclear One, Unit 1, Amendment Number 245 dated August 10, 2011 (ACN ML111940085).

4.3 Significant Hazards Consideration

Southern Nuclear Operating Company (SNC) evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on three standards set forth in 10 CFR 50.92(c) as discussed below.

With regard to the proposal to adopt TSTF-448, SNC has reviewed the proposed no significant hazards consideration determination (NSHCD) published in the Federal Register as part of the Consolidated Line Item Improvement Process (CLIIP). SNC has concluded that the proposed NSHCD presented in the Federal Register notice is applicable to Farley Nuclear Plant (FNP) and is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

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.

Regarding TSTF-312-A, the proposed change would allow containment penetrations to be unisolated under administrative controls during core alterations or movement of irradiated fuel assemblies within containment. The status of containment penetration flow paths (i.e., open or closed) is not an initiator for any design basis accident or event, and therefore the proposed change does not increase the probability of any accident previously evaluated. The proposed change does not affect the design of the primary containment, or alter plant operating practices such that the probability of an accident previously evaluated would be significantly increased. The proposed change does not significantly change how the plant would mitigate an accident previously evaluated, and is bounded by the fuel handling accident (FHA) analysis.

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 AST and the associated proposed Technical Specification changes and new X/Qs have no impact to the initiation of any DBAs. These changes do not affect the design function or 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.

Regarding TSTF-312-A, allowing penetration flow paths to be open is not an initiator for any accident. The proposed change to allow open penetration flow paths will not affect plant safety functions or plant operating practices such that a new or different accident could be created. There are no design changes associated with the proposed changes, and the change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The change does not alter assumptions made in the safety analysis, and is consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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 amendment 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.

   Regarding TSTF-312-A, TS 3.9.3 provides measures to ensure that the dose consequences of a postulated FHA inside containment are minimized. The proposed change to LCO 3.9.3 will allow penetration flow path(s) to be open during refueling operations under administrative control. These administrative controls will provide assurance that prompt closure of open penetrations flow paths can and will be achieved in the event of an FHA inside containment, and will minimize dose consequences. The proposed change does not affect the safety analysis acceptance criteria for ay analyzed event, nor is there a change to any safety analysis limit. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are deterred, not is here any adverse effect on those plant systems necessary to assure the accomplishment of protective functions. The proposed change will not result in plant operation in a configuration outside the design basis.

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 Conclusions

In conclusion, based on the considerations discussed above, (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 to the health and safety of the public.

5.0 Environmental Consideration

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

Regarding the proposal to adopt TSTF-448, SNC has reviewed the environmental evaluation included in the model safety evaluation dated January 17, 2007, as part of the CLIIP. SNC has concluded that the staff's findings presented in that evaluation are applicable to FNP and the evaluation is hereby incorporated by reference for this application.

6.0 References

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

2. Notice of Availability of Technical Specification Improvement To Modify Requirements Regarding Control Room Envelope Habitability Using the Consolidated Line Item Improvement Process, 72 Federal Register 10 (January 17, 2007).

3. Letter from S Peters (NRC) to L. Stinson (SNC), dated September 30, 2004, "Joseph M. Farley Nuclear Plant, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MC4186 and MC4187)." [ADAMS Accession Number ML042780424]

4. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

5. Letter from S Peters (NRC) to L. Stinson (SNC), dated September 30, 2004, "Joseph M. Farley Nuclear Plant, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MC0625 and MC0626)." [ADAMS Accession Number ML042820368]

Joseph M. Farley Nuclear Plant – Units 1 and 2
Alternative Source Term
License Amendment Request

Enclosure 2

Operating License and Technical Specification Pages (Markup)

(5) Updated Final Safety Analysis Report Supplement

The Updated Final Safety Analysis Report supplement, as revised, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed license. Until that update is complete, Southern Nuclear may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Southern Nuclear evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements of that section.

The Southern Nuclear Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to the period of extended operation. Southern Nuclear shall complete these activities no later than June 25, 2017, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

(6) Reactor Vessel Material Surveillance Capsules

U1/U2 Operating License Condition 7

All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion.

D. Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," and was submitted on May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 186, as supplemented by a change approved by License Amendment No. 199.

E. This renewed license is subject to the following additional conditions for the protection of the environment:

to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

c. Transition License Conditions

1) Before achieving full compliance with 10 CFR 50.48(c), as specified by 2) below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2) above.

2) The licensee shall implement the modifications to its facility, as described in Attachment S, Table S-2, "Plant Modifications Committed," of SNC letter NL-14-1273,
dated August 29, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by November 6, 2017. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3) The licensee shall implement the items as listed in Attachment S, Table S-3, "Implementation Items," of SNC letter NL-14-1273, dated August 29, 2014,
within 180 days after NRC approval, except for items 30 and 32. Items 30 and 32 shall be implemented by February 6, 2018.

U1/U2 Operating License Condition 7

(7) ~~Deleted per Amendment 144~~
(8) Deleted per Amendment 144
(9) Deleted per Amendment 144
(10) Deleted per Amendment 144
(11) Deleted per Amendment 144
(12) Deleted per Amendment 144
(13) Deleted per Amendment 144
(14) Deleted per Amendment 144
(15) Deleted per Amendment 144
(16) Deleted per Amendment 144
(17) Deleted per Amendment 144
(18) Deleted per Amendment 144
(19) Deleted per Amendment 144
(20) Deleted per Amendment 144
(21) Deleted per Amendment 144

(22) Additional Conditions
The Additional conditions contained in Appendix C, as revised through Amendment No. 137, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the additional conditions.

U1/U2 Operating License Condition 7 Insert

(7) Upon implementation of Amendment No. xxx adopting TSTF–448, Revision 3, the determination of control room envelope (CRE) unfiltered air in-leakage as required by SR 3.7.10.4, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.18.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.18.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.10.4, in accordance with Specification 5.5.18.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003–01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003–01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Filtration/Pressurization System (CREFS)

LCO 3.7.10 Two ~~Control Room Emergency Filtration/Pressurization System (CREFS)~~ trains ~~and the Control Room Envelope (CRE)~~ shall be OPERABLE.

-----NOTE-----
The control room envelope (CRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies,
During CORE ALTERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREFS train inoperable for reasons other than Condition B.	A.1 Restore CREFS train to OPERABLE status.	7 days
B. One or more CREFS trains inoperable due to inoperable CRE ~~inoperable~~boundary.	B.1 Initiate action to implement mitigating actions.	Immediately
	AND	
	~~B.2.1 Restore CRE to OPERABLE status~~	~~24 hours~~
	~~OR~~	
	B.2~~.2.1~~ ~~General Design Criteria (GDC) 19 met using mitigating actions in B.1.~~	24 hours
	~~AND~~	
Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	AND	
	B.3~~2.2.2~~ Restore CRE boundary to OPERABLE status.	~~30~~90 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. <u>AND</u> C.2 ------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. --------------------------------- Be in MODE 4.	6 hours 12 hours
D. Two CREFS trains inoperable in MODE 1, 2, 3, OR 4.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours
E. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	E.1 Place OPERABLE CREFS train in emergency recirculation mode. <u>OR</u> E.2.1 Suspend CORE ALTERATIONS. <u>AND</u> E.2.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition B not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS. OR Two CREFS trains inoperable during movement of irradiated fuel assemblies or during CORE ALTERATIONS. OR One or more CREFS trains inoperable due to an inoperable CRE boundary during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	F.1 Suspend CORE ALTERATIONS. AND F.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

SURVEILLANCE REQUREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREFS Pressurization train with the heaters operating and each CREFS Recirculation and Filtration train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Perform required CREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.10.3	---------------------------NOTE------------------------------- Not required to be performed in MODES 5 and 6. --- Verify each CREFS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.10.4	Perform required CRE unfiltered air in-leakage testing in accordance with the Control Room Envelope Habitability Program~~Verify CRE Δp within limits in the Control Room Integrity Program (CRIP)~~.	In accordance with the Control Room Envelope Habitability Program~~24 months on a STAGGERED TEST BASIS~~
~~SR 3.7.10.5~~	~~Verify CRE integrity in accordance with the CRIP.~~	~~In accordance with the CRIP~~

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 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. One door in each air lock is capable of being closed; 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 an OPERABLE Containment Purge and Exhaust Isolation System.

------------------------------------- NOTE -------------------------------------
<u>Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative control.</u>

APPLICABILITY: During CORE ALTERATIONS,
During movement of 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
	<u>AND</u>	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

5.5 Programs and Manuals

~~5.5.18~~ ~~Control Room Integrity Program (CRIP)~~

Control Room Envelope Habitability Program Insert

~~A Control Room Integrity Program (CRIP) shall be established and implemented to ensure that the control room integrity is maintained such that a radiological event, hazardous chemicals, or a fire challenge (e.g., fire byproducts, halon, etc.) will not prevent the control room operators from controlling the reactor during normal or accident conditions. The program shall require testing as outlined below. Testing should be performed when changes are made to structures, systems and components which could impact Control Room Impact (CRE) integrity. These structures, systems and components may be internal or external to the CRE. Testing should also be conducted following a modification or a repair that could affect CRE inleakage. Testing should also be performed if the conditions associated with a particular challenge result in a change in operating mode, system alignment or system response that could result in a new limiting condition. Testing should be commensurate with the type and degree of modification or repair. Testing should be conducted in the alignment that results in the greatest consequence to the operators.~~

~~A CRIP shall be established to implement the following:~~

~~a. Demonstrate, using Regulatory Guide (RG) 1.197 and ASTM E741, that CRE inleakage is less than the below values. The values listed below do not include 10 cfm assumed in accident analysis for ingress / egress.~~

~~i) 43 cfm when the control room ventilation systems are aligned in the emergency recirculation mode of operation,~~

~~ii) 600 cfm when the control room ventilation systems are aligned in the isolation mode of operation, and~~

~~iii) 2,340 cfm when the control room ventilation systems are aligned in the normal mode of operation;~~

~~b. Demonstrate that the leakage characteristics of the CRE will not result in simultaneous loss of reactor control capability from the control room and the hot shutdown panels;~~

~~c. Maintain a CRE configuration control and a design and licensing bases control program and a preventative maintenance program. As a minimum, the CRE configuration control program will determine whether the i) CRE differential pressure relative to adjacent areas and ii) the control room ventilation system flow rates, as determined in accordance with ASME N510-1989 or ASTM E2029-99, are consistent with the values measured at the time the ASTM E741 test was performed. If item i or ii has changed, determine how this change has affected the inleakage characteristics of the CRE. If there has been degradation in the inleakage characteristics of the~~

~~5.5.18 Control Room Integrity Program (CRIP) (continued)~~

~~CRE since the E741 test, then a determination should be made whether the licensing basis analyses remain valid. If the licensing basis analyses remain valid, the CRE remains OPERABLE.~~

~~d. Test the CRE in accordance with the testing methods and at the frequencies specified in RG 1.197, Revision 0, May 2003.~~

~~The provisions of SR 3.0.2 are applicable to the control room inleakage testing frequencies.~~

5.5.19 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

Control Room Envelope Habitability Program Insert

5.5.18 Control Room Envelope Habitability Program

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

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

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

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

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREFS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

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

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

Joseph M. Farley Nuclear Plant – Units 1 and 2
Alternative Source Term
License Amendment Request

Enclosure 3

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 were hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there was 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 50.67, "Accident Source Term~~100, "Reactor Site Criteria"~~ (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL 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 50.67, "Accident Source Term,"~~100, "Reactor Site Criteria,"~~ 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.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

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 IWX-5000.
4. 10 CFR 50.67~~100~~.
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.

With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal, and a 1% delta k/k SHUTDOWN MARGIN provides adequate protection.

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 50.67, "Accident Source Term,"~~100, "Reactor Site Criteria,"~~ 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.

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$ and 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

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of Immediately is adequate to ensure prompt operator action to correctly align and start the required

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

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, Section 15.4.2.
3. FSAR, Section 15.2.4.
4. 10 CFR 50.67~~100~~.
5. Letter from D.E. McKinnon to L.K. Mathews, "Operating Procedure for Mode 4/5 Boron Dilution," 90 AP*-G-0041, July 6, 1990.

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, defined in this specification as the Trip Setpoints, 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 2735 psig 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 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 50.67~~100~~ limits. Different accident categories are allowed a different fraction of these limits, based on probability of

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

purge and exhaust isolation radiation monitors act as backup to the SI signal to ensure closing of the purge and exhaust valves. They are also the primary means for automatically 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 are below 10 CFR 50.67~~100~~ (Ref. 1) limits.

The containment purge and exhaust 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 Purge and Exhaust Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

   The LCO requires two channels OPERABLE. The operator can initiate Containment Purge Isolation at any time by using either of two valve hand switches in the control room (labeled CTMT PURGE DMPRS). Each switch actuates one train of purge/exhaust isolation valves. Actuation of either handswitch isolates the Containment Purge and Exhaust System.

   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 handswitch and the interconnecting wiring to the purge/exhaust isolation valves in that train.

2. Automatic Actuation Logic and Actuation Relays

   The LCO requires two trains 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 (Paragraph 1), SI, and ESFAS Function 3.a,

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.7

The CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

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

SR 3.3.6.8

This SR ensures the individual channel response times are less than or equal to the maximum values assumed in the safety analysis. The response time testing acceptance criteria are included in FSAR Table 7.3-16 (Ref. 4). This surveillance is performed in accordance with the guidance provided in the ESF RESPONSE TIME surveillance requirement in LCO 3.3.2, ESFAS.

REFERENCES

1. 10 CFR 50.67~~100.11~~.
2. Not used.
3. Not used.
4. FSAR Table 7.3-16

B 3.3 INSTRUMENTATION

B 3.3.8 Penetration Room Filtration (PRF) System Actuation Instrumentation

BASES

BACKGROUND

The PRF ensures that radioactive materials in the Spent Fuel Pool Room atmosphere following a fuel handling accident or ECCS pump rooms and penetration rooms of the auxiliary building following a loss of coolant accident (LOCA) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.12, "Penetration Room Filtration System." The system initiates filtered ventilation of the Spent Fuel Pool Room (including isolation of the normal ventilation) automatically following receipt of a high radiation signal (gaseous) or a low air flow signal from the normal Spent Fuel Pool Room ventilation system. In addition, the system initiates filtered ventilation of the ECCS pump rooms and penetration rooms following receipt of a Phase B Containment Isolation signal. Initiation may also be performed manually as needed from the main control room.

High gaseous radiation provides PRF initiation. Each PRF train is initiated by high radiation detected by a channel dedicated to that train. There are a total of two channels, one for each train. Each channel contains a gaseous monitor. High radiation detected by either monitor or a low air flow signal from the normal Spent Fuel Pool Room ventilation or a Phase B Containment Isolation signal from the Engineered Safety Features Actuation System (ESFAS) starts the PRF. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the Spent Fuel Pool Room or ECCS pump rooms and penetration rooms. Since the radiation monitors include an air sampling system, various components such as sample line valves and sample pumps are required to support monitor OPERABILITY.

APPLICABLE SAFETY ANALYSES

The PRF ensures that radioactive materials in the Spent Fuel Pool Room atmosphere following a fuel handling accident or ECCS pump rooms and penetration rooms following a LOCA are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the plant exhaust following a LOCA or fuel handling accident so that offsite doses remain within the limits specified in 10 CFR 50.67~~100~~ (Ref. 1).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.7

The CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50.67~~100.11~~.
2. FNP – 1/2 - RCP - 252.
3. Not used.

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is typically seen as a precursor to a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gpd (i.e. total leakage less than or equal to 450 gpd) is significantly less than the conditions assumed in the safety analysis (with leakage assumed to occur at room temperature in both cases).

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

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via the main steam safety valves. The majority of the activity released to the atmosphere results from the tube rupture. Therefore, the 1 gpm primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The MSLB is more limiting for primary to secondary LEAKAGE. The safety analysis for the MSLB assumes 0.35 gpm~~500 gpd~~ and 0.65 gpm~~470 gpd~~ primary to secondary LEAKAGE in the faulted and both intact steam generators respectively as an initial condition (1 gpm total). The offsite dose consequences resulting from the MSLB accident are bounded by a small fraction (i.e., 10%) of the limits defined in 10 CFR 50.67~~100~~. The RCS specific activity assumed was 0.5 μCi/gm DOSE EQUIVALENT I-131 at a conservatively high letdown flow of 145 gpm, with either a pre-existing or an accident initiated iodine spike. These values bound the Technical Specifications values.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose ~~to the whole body and the thyroid~~ that an individual at the exclusion area~~site~~ boundary can receive for 2 hours during an accident, or for the duration of the accident at the Low Population Zone, is specified in 10 CFR 50.67~~100~~ (Ref. 1). The limits on specific activity ensure that the doses are held to an appropriate fraction of the 10 CFR 50.67~~100~~ limits (i.e., a small fraction of or well within the 10 CFR 50.67~~100~~ limits depending on the specific accident analysis) 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 offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) or main steam line break (MSLB) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the exclusion area~~site~~ boundary, or at the low population zone outer boundary for the radiological release duration, to an appropriate fraction of the 10 CFR 50.67~~100~~ dose ~~guideline~~ limits. ~~The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.~~

~~The parametric evaluations showed the potential offsite dose levels for a SGTR or MSLB accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.~~

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting doses will not exceed an appropriate fraction of the 10 CFR 50.67~~100~~ dose ~~guideline~~ limits following a SGTR or MSLB accident. The SGTR and MSLB safety analyses~~is~~ (Ref. 2 and 3) assume~~s~~ the specific activity of the reactor coolant at 0.5 μCi/gm, a conservatively high letdown flow of 145 gpm, and a bounding reactor coolant steam generator (SG) tube leakage of 1 gpm total for three

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

SGs. ~~The MSLB analysis assumes a steam generator tube leakage of 500 gpd in the faulted loop and 470 gpd in each of the intact loops for a total leakage of 1440 gpd.~~ These~~is~~ analyse~~is~~ resulted in offsite doses bounded by a small fraction (i.e., 10%) of the 10 CFR 50.67~~100~~ guidelines using FGR No. 11 and 12~~ICRP 30~~ Dose Conversion Factors (DCFs). The initial RCS specific activity assumed was 0.5 μCi/gm DOSE EQUIVALENT I-131 at a conservatively high letdown flow of 145 gpm with an iodine spike. These values bound the Technical Specifications values. The safety analysis assumes for both the SGTR and MSLB the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.16, "Secondary Specific Activity."

The analysis for the SGTR and MSLB accidents establishes~~es~~ the acceptance limits for RCS specific activity. Reference to these~~is~~ analyse~~is~~ are~~is~~ used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

~~The SGTR analysis assumes an RCS coolant activity of 0.5 μCi/gm DOSE EQUIVALENT I-131 at a conservatively high letdown flow of 145 gpm.~~ The SGTR and MSLB analyse~~is~~ consider~~s~~ two cases of reactor coolant specific activity. One case assumes specific activity at 0.5 μCi/gm DOSE EQUIVALENT I-131 at a conservatively high letdown flow of 145 gpm with an accident initiated iodine spike that increases the I-131 activity release rate into the reactor coolant by a factor of 500 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 30 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. These values bound the Technical Specifications values. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/\bar{E}$ μCi/gm for gross specific activity.

The SGTR analysis also assumes a loss of offsite power coincident with a 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 coincident 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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. The MSLB analysis assumes a double-ended guillotine break of a main steamline outside of containment. The affected steam generator will rapidly depressurize and release both the radionuclides initially contained in the secondary coolant, and the primary coolant activity transferred via SG tube leakage, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact SGs and noble gas activity due to SG tube leakage is released to the atmosphere through either the SG atmospheric relief valves (ARVs) or the SG safety relief valves.

The safety analysis assumes an accident initiated iodine spike and shows the radiological consequences of a MSLB accident are within a small fraction of the Reference 1 dose ~~guideline~~ limits.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the pre-accident activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The MSLB safety analysis has pre-accident iodine spiking levels up to 30 μCi/gm DOSE EQUIVALENT I-131.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a MSLB accident occurring during the established 48 hour time limit. The occurrence of a MSLB accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 50.67~~100~~ dose guideline limits.

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

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

LCO

The specific iodine activity is limited to 0.5 μCi/gm DOSE EQUIVALENT I-131 at a conservatively high letdown flow of 145 gpm for the SGTR analysis and for the MSLB analysis, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the ~~thyroid~~ dose to an individual during the Design Basis Accident (DBA) will be

(continued)

BASES

LCO (continued)

an appropriate fraction of the allowed ~~thyroid~~ dose. The limit on gross specific activity ensures the 2 hour ~~whole body~~ dose to an individual at the site boundary during the DBA will be a small fraction of the allowed ~~whole body~~ dose. The SGTR (Ref. 2) and MSLB accident analyses show that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR or MSLB, lead to site boundary doses that exceed the dose ~~guideline~~ limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR or MSLB to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

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 limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note permits the use of the provisions of LCO 3.0.4c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 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 fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR has been modified by a Note that indicates sampling is required to be performed within 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 at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 50.67~~100.11, 1973~~.
2. FSAR, Section 15.4.3.

(continued)

BASES

BACKGROUND (continued)

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

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 1 gpm~~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 released via the main steam safety valves. The majority of the activity released to the atmosphere results from the tube rupture.

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 gpm as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be 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 50.67~~100~~ (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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging 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 and Reference 6 provide 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~~Not used.
3. 10 CFR 50.67~~100~~.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI TR-107569, "Pressurized Water Reactor Steam Generator Examination Guidelines."

BASES

APPLICABLE SAFETY ANALYSES (continued)

be increased if the long term pH of the recirculation solution is not adjusted to 7.5 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 50.67~~100 and 10 CFR 50, Appendix A, General Design Criterion 19 respectively~~.

The Recirculation Fluid pH Control System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of the Recirculation Fluid pH Control System ensures sufficient TSP is maintained in the three TSP storage baskets to increase the long term recirculation fluid pH to between 7.5 and 10.5 following a LOCA. A pH range of 7.5 to 10.5 is sufficient to prevent significant amounts of iodine released from fuel failure and dissolved in the recirculation fluid, from converting to a volatile form and evolving from solution into the containment atmosphere during the ECCS recirculation phase. In addition, an alkaline pH in this range will minimize chloride induced stress corrosion cracking of austenitic stainless steel components, and minimize the hydrogen produced by the corrosion of galvanized surfaces and zinc-based paints.

In order to achieve the desired pH range of 7.5 to 10.5 in the post-LOCA recirculation solution a total of between 10,000 pounds (185 ft^3) and 12,900 pounds (215 ft^3) of TSP (or appropriate weights/volumes for equivalent compounds) is required. The required amount of TSP is determined considering the volume of water involved, the target pH range, and the density of different vendor types of TSP that are available. Although the amount of TSP required is based on mass, a required volume is verified since it is not feasible to weigh the entire amount of TSP in containment.

APPLICABILITY

In MODES 1, 2, 3, and 4 a DBA could cause the release of radioactive material in containment requiring the operation of the ECCS Recirculation Fluid pH Control System. The ECCS Recirculation Fluid pH Control System assists in reducing the amount of radioactive material available for release to the outside atmosphere after a DBA.

(continued)

BASES

LCO (continued)

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents such that~~that could result in~~ offsite exposures are less than~~comparable to~~ the 10 CFR 50.67~~100~~ (Ref. 4) limits.

APPLICABILITY

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

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

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each steam line. The Completion Time(s) of the inoperable MSIV Systems will be tracked separately for each steam line starting from the time the Condition was entered for that steam line.

A.1

With one MSIV inoperable in one or more steam lines in MODE 1, action must be taken to restore the inoperable MSIV to OPERABLE status within 72 hours. Some repairs to the MSIV can be made with the unit at power. The 72 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time that would require the MSIVs to close and the remaining OPERABLE MSIV in the steam line. This Completion Time is also consistent with the Completion Times provided for a single inoperable train in other ESF systems that contain redundant trains of equipment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

accident and containment analyses. This Surveillance is normally performed while 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. This surveillance may be performed in lower modes but must be performed prior to entry into MODE 2.

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 6.2.
3. FSAR, Section 15.4.2.
4. 10 CFR 50.67~~100.11~~.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

APPLICABLE SAFETY ANALYSES (continued)

the unit to RHR entry conditions. The limiting design basis accident for the ARVs is established by the Steam Generator Tube Rupture (SGTR) event (Ref. 2). The SGTR event is analyzed for two cases to determine that the offsite doses meet the NRC acceptance criteria. That is, for the case of an accident initiated Iodine spike, the doses from the accident are a small fraction of the limits defined in 10 CFR 50.67~~100~~ and for the case of a pre-accident Iodine spike, the doses from the accident are within the limits defined in 10 CFR 50.67~~100~~. The SGTR event assumes recovery with and without offsite power. The loss of offsite power assumption results in the ARVs being relied upon to reduce RCS temperature to recover from an SGTR and also to reduce RCS temperature and pressure to RHR entry conditions. The accident analysis does not assume a specific method of valve operation to mitigate the accident. The analysis assumes the SG tube break flow is terminated within 30 minutes of the initiation of the accident.

The recovery from the SGTR event requires a rapid cooldown to establish adequate subcooling as a necessary step to allow depressurization of the RCS to terminate the primary to secondary break flow in the ruptured steam generator. The time required to terminate the primary to secondary break flow in the SGTR event is more critical than the time required to cool the RCS down to RHR entry conditions for this event and other accident analyses. Thus, the SGTR is the limiting event for the ARVs.

Each ARV is equipped with two manual isolation valves in the event an ARV spuriously fails to 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. At least one manual isolation valve must be OPERABLE to isolate a failed open ARV line. A closed manual isolation valve does not render it or its ARV line inoperable. The accident analysis does not model a specific method of valve operation and allows 30 minutes to terminate the SG tube break flow. Sufficient time is available to unisolate and manually operate the ARV.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.10 ~~Control Room~~CREFS

BASES

BACKGROUND

The control room provides a protected environment from which ~~operators~~ occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke ~~toxic gas~~. This environment is protected by the integrity of the Control Room Envelope (CRE) and the operation of the Control Room Emergency Filtration/Pressurization System (CREFS). The Unit 1 and 2 control room is a common room served by a shared CREFS.

~~The control room boundary is the combination of walls, floor, roof, ducting, valves or dampers, ESF HVAC equipment housings, doors, penetrations and equipment that physically form the CRE. The CRE is the area within the confines of the control room boundary that contains the spaces that control room operators inhabit to control the plant. This space is protected for normal operation, natural events, and accident conditions.~~

Maintaining the integrity of the CRE minimizes the infiltration of unfiltered air from areas adjacent to the CRE, thereby minimizing the possibility that the effects of a radiological challenge would result in a radiological dose which exceeds General Design Criteria (GDC) 19. It also minimizes the possibility that a fire challenge would result in a condition where the operator would be disabled or impaired such that the reactor could not be controlled from the control room or the hot shutdown panels. In addition, the CRE minimizes the possibility that a hazardous chemical challenge would result in a condition where the operator would be disabled or impaired such that the reactor could not be controlled from the control room. While the CRE provides a boundary for the CREFS to operate in, the CRE is independent from the CREFS and its OPERABILITY requirements are separate from the CREFS.

The CREFS consists of two independent, redundant trains that recirculate and filter the air in the CRE ~~control room air~~ in conjunction with the CRACS, ~~and~~ two independent, redundant trains that pressurize the control room with filtered outside air, and a CRE boundary that limits the inleakage of unfiltered air. Each ~~filter unit~~ CREFS train consists of a prefilter, a high efficiency particulate air (HEPA) filter, and an activated charcoal adsorber section for removal of gaseous activity (principally iodine). Each pressurization filter also contains a heater. Each train contains filter units, fans, and instrumentation which form the system.

(continued)

BASES

BACKGROUND (continued)

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CREFS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the CRE ~~control room~~ is isolated, and the stream of ventilation air is recirculated through the system filter trains. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers. Operation of each pressurization train for at least 15 minutes per month, with the heaters energized, justifies their OPERABILITY. During operation, the heaters reduce moisture buildup on the HEPA filters and adsorbers. The heater is important to the effectiveness of the charcoal adsorbers.

Actuation of the CREFS places the system in the emergency recirculation mode of operation. Actuation of the system to the emergency recirculation mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the ~~control room~~ air within the CRE through the redundant trains of HEPA and the charcoal filters. The emergency recirculation mode of operation also initiates pressurization and filtered ventilation of the air supply to the CRE ~~control room~~.

The normal outside air supply is filtered, diluted with building air from the computer rooms, and added to the control room. The air entering the CRE ~~control room~~ is continuously monitored by radiation detectors. One detector output above the setpoint will cause the control room ventilation to be isolated. The CREFS is then started manually.

(continued)

BASES

BACKGROUND (continued)

A single CREFS train provides makeup air flow and radiological dose cleanup for the control room. The CREFS operation in maintaining the control room habitable is discussed in the FSAR, Section 6.4 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREFS is designed in accordance with Seismic Category I requirements.

The CREFS is designed to maintain a habitable environment in the CRE ~~the control room environment~~ for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem total effective dose equivalent (TEDE).

~~An inoperable CRE does not render the CREFS inoperable or vice versa. The OPERABILITY of the CREFS and the CRE are determined separately and both are required to be OPERABLE.~~

APPLICABLE SAFETY ANALYSES

The CREFS components are arranged in redundant, safety related ventilation trains. The location of components within the CRE and ducting of the CRE ensure an adequate supply of filtered air to all areas requiring access. The CREFS provides airborne radiological protection for the CRE occupants ~~control room operators~~, as demonstrated by the CRE ~~control room accident dose~~ occupant dose analyses for the most limiting design basis ~~loss of coolant~~ accident~~,~~ fission product release presented in the FSAR, Chapter 15 (Ref. 2).

Maintaining the integrity of the CRE limits the quantity of contaminants allowed into the CRE so that the radiological dose criteria of GDC 19 are met. The analysis of toxic gas releases demonstrates that the toxicity limits are not exceeded in the control room following a toxic chemical release. The evaluation of a smoke challenge demonstrates that it will not result in the inability of ~~Maintaining the integrity of the CRE helps to ensure that~~ the CRE occupants ~~control room operators may~~ to maintain reactor control either from the control room ~~and maintain separation between the control room and~~ or from the hot shutdown panels.

The worst case single active failure of a component of the CREFS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

Two independent and redundant CREFS trains are required to be OPERABLE to ensure that at least one is available ~~assuming~~ if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem TEDE to the CRE occupants ~~control room operator~~ in the event of a large radioactive release.

~~The~~Each CREFS train is considered OPERABLE when the individual components necessary to limit CRE occupant ~~operator~~ exposure are OPERABLE ~~in both trains~~. A CREFS train is OPERABLE when the associated:

a. Fans are OPERABLE; (recirculation, filtration, Pressurization, and CRACS Fans)

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

c. Heater is OPERABLE and air circulation can be maintained.

~~In addition, the CRE must be maintained OPERABLE, including the integrity of the walls, floors, ceilings, ductwork, valves and dampers, ESF HVAC equipment housings, and access doors. Inleakage must also be minimized such that operator exposure limits are not exceeded.~~

In order for the CREFS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

~~An inoperable CRE does not render the CREFS inoperable or vice versa. The OPERABILITY of the CREFS and the CRE are determined separately and both are required to be OPERABLE.~~

The LCO is modified by a Note allowing the CRE to be opened intermittently under administrative controls without requiring entry into

(continued)

BASES

LCO (continued)

Condition B for an inoperable CRE. This Note only applies to opening in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For maintenance access openings, such as hatches and test ports, the administrative control of the opening is performed by the attendant person(s) performing the maintenance. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE~~control room~~. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE~~control room~~ integrity is indicated.

APPLICABILITY

With either unit in MODES 1, 2, 3, or 4 or during movement of irradiated fuel assemblies or during CORE ALTERATIONS, the CREFS ~~and the CRE~~ must be OPERABLE to ensure that the CRE will remain habitable ~~control operator exposure~~ during and following a DBA.

During movement of irradiated fuel assemblies and CORE ALTERATIONS, the CREFS and the CRE must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1

With one CREFS train inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore it to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREFS train is adequate to perform the CRE occupant ~~control room~~ protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREFS train could result in loss of CREFS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

(continued)

BASES

ACTIONS (continued)

B.1, B.2~~.1, B.2.2.1~~, and B.3~~2.2.2~~

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

~~If the CRE is inoperable, the operator protection analyses assumption of inleakage may be exceeded. During the period that the CRE is inoperable, mitigating actions must be initiated to protect control room operators from potential hazards. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE) should be preplanned for initiation upon entry into the condition.~~

~~Within 24 hours of entry into Condition B, Actions must be taken to restore the CRE to OPERABLE status or to verify that the requirements of GDC 19 are met for the facility. GDC 19 is verified to be met by limiting dose from radioactive gas and particulates, and exposure to toxic gas and smoke, to levels that support control room~~

(continued)

BASES

ACTIONS (continued)

~~habitability, crediting, as necessary, the mitigating actions required by Required Action B.1. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, the use of mitigating actions, and the time necessary to perform an assessment.~~

~~If it is determined that the requirements of GDC 19 are met crediting, as necessary, the mitigating actions required by Required Action B.1, 30 days are provided to return the CRE to OPERABLE status. The 30 day Completion Time is a reasonable time to diagnose, plan, and repair most problems with the CRE.~~

C.1 and C.2

In MODE 1, 2, 3, or 4, if an inoperable CREFS train or CRE cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. Remaining within the applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 5). In MODE 4 the Steam Generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 5, the steam turbine driven Auxiliary Feedwater Pump must be available to remain in MODE 4. Should Steam Generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If two CREFS trains are inoperable in MODE 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

E.1, E.2.1, and E.2.2

During movement of irradiated fuel assemblies or during CORE ALTERATIONS, if an inoperable CREFS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREFS train in the emergency recirculation mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE~~control room~~. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

F.1 and F.2

During movement of irradiated fuel assemblies or during CORE ALTERATIONS, ~~if an inoperable CRE cannot be restored to OPERABLE status within the required Completion Time or~~ with two CREFS trains inoperable or with one or more CREFS trains inoperable due to an inoperable CRE boundary, action must be taken to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE~~control room~~. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train (CREFS and Pressurization) once every month provides an adequate check of this system. The CREFS trains are initiated from the control room with flow through the HEPA and charcoal filters. Systems must be operated for ≥ 15 minutes to demonstrate the function of the system (Ref. 3). Systems with heaters must be operated with the heaters energized. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.2

This SR verifies that the required CREFS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREFS filter tests are in accordance with ASME N510-1989 (Ref. 4). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

This SR verifies that each CREFS train starts and operates on an actual or simulated Safety Injection (SI) actuation signal. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. This SR is modified by a note which provides an exception to the requirement to meet this SR in MODES 5 and 6. This is acceptable since the automatic SI actuation function is not required in these MODES.

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.4 (continued)

assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 6) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 7). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 8). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

~~This SR verifies that the CRE Δp can be maintained within limits defined in the Control Room Integrity Program (CRIP) with one CREFS train in operation. If the requirements of this SR cannot be met, a determination must be made as to the cause of the failure. Once identified, the appropriate Condition (for either the CREFS or the CRE) must be entered. For example, if the failure is due to a breach in the integrity of the CRE, the Condition for an inoperable CRE would be entered but the Condition for an inoperable CREFS would not be entered. An inoperable CRE does not render the CREFS inoperable or vice versa. The frequency of 24 months on a STAGGERED TEST BASIS is adequate and has been shown to be acceptable by operating experience.~~

~~Any change in the components being tested by this SR will require reevaluation of STI Evaluation Number 558904 in accordance with the Surveillance Frequency Control Program.~~

~~SR 3.7.10.5~~

~~This SR verifies the integrity of the CRE by requiring testing for control room inleakage. The details of the inleakage testing are contained in the CRIP.~~

(continued)

BASES

REFERENCES

1. FSAR, Section 6.4.

2. FSAR, Chapter 15.

3. Regulatory Guide 1.52, Rev. 3.

4. ASME N510-1989.

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

6. Regulatory Guide 1.196

7. NEI 99-03, "Control Room Habitability Assessment," June 2001

8. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040300694)

BASES

APPLICABLE SAFETY ANALYSES

The PRF System design basis is established by the consequences of the limiting Design Basis Accidents (DBAs), which are a fuel handling accident and a large break loss of coolant accident (LOCA). The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the PRF System. The PRF System also functions following a small break LOCA with a Phase B signal or manual operator actuation in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve steam packing. The DBA analysis of the fuel handling accident and LOCA assumes that only one train of the PRF System is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the spent fuel pool room is determined for a fuel handling accident and ECCS leakage for a LOCA. The analysis of the effects and consequences of a fuel handling accident and a LOCA are presented in Reference 3. The assumptions and the analysis for the fuel handling accident follow the guidance provided in Regulatory Guide 1.1.183~~25~~ (Ref. 4).

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

LCO

Two independent and redundant trains of the PRF System are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. During movement of irradiated fuel in the spent fuel pool room both trains of PRF are required to be aligned to the spent fuel pool room. Total system failure could result in the atmospheric release from the spent fuel pool room or ECCS pump rooms exceeding 25% of the 10 CFR 50.67~~100~~ (Ref. 5) limits in the event of a fuel handling accident or LOCA respectively.

The PRF System is considered OPERABLE when the individual components necessary to control exposure in the spent fuel pool room, ECCS pump rooms, and penetration area are OPERABLE in both trains. A PRF train is considered OPERABLE when its associated:

a. Recirculation and exhaust fans are OPERABLE;

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.5 (continued)

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

SR 3.7.12.6

During the fuel handling mode of operation, the PRF is designed to maintain a slightly negative pressure in the spent fuel pool room with respect to atmospheric pressure and surrounding areas at a flow rate of ≤ 5,500 cfm, to prevent unfiltered leakage. The slightly negative pressure is verified by using a non-rigorous method that yields some observable identification of the slightly negative pressure. Examples of non-rigorous methods are smoke sticks, hand held differential pressure indicators, or other measurement devices that do not provide for an absolute measurement. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 6.2.3.
2. FSAR, Section 9.4.2.
3. FSAR, Sections 15.4.1 and 15.4.5.
4. Regulatory Guide 1.183~~25~~.
5. 10 CFR 50.67~~100~~.
6. ASME N510-1989.

B 3.7 PLANT SYSTEMS

B 3.7.13 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, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section 15.4.5 (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 1.183~~25~~ (Ref. 4). The resultant 2 hour ~~thyroid~~ dose per person at the exclusion area~~site~~ boundary is well within the 10 CFR 50.67~~100~~ (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of 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 between the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

The fuel storage pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1 (continued)

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 in accordance with SR 3.9.6.1 (refueling cavity water level verification).

REFERENCES

1. FSAR, Section 9.1.2.
2. FSAR, Section 9.1.3.
3. FSAR, Section 15.4.5.
4. Regulatory Guide 1.183~~25~~, Rev. 0.
5. 10 CFR 50.67~~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 450 gallons per day tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 0.5 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).

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the site boundary would be within the limits of 10 CFR 20.1001- 20.2402 if the main steam safety valves (MSSVs) and Atmospheric Relief Valves (ARVs) are open for 2 hours following a trip from full power.

Operating at the allowable limits results in a 2 hour exclusion area~~site~~ boundary exposure well within the 10 CFR 50.67~~100~~ (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 µCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the exclusion are~~site~~ boundary dose limits (Ref. 1)~~ for whole body and thyroid dose rates~~.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric relief valves (ARVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ARVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant in the steam generators ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

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

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies that the secondary specific activity in the steam generators is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

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

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be limited to maintain dose consequences within regulatory limits 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 "refueling integrity" rather than "containment OPERABILITY." Refueling integrity 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 of 10 CFR 50.67~~100~~. 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 unit operation 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 required. During periods of unit shutdown

(continued)

BASES

BACKGROUND (continued)

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 irradiated fuel assemblies within containment (Ref. 1).

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). The fuel handling accident analyzed includes dropping a single irradiated fuel assembly. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are less than~~well within~~ the dose limits~~guideline values~~ specified in 10 CFR 50.67~~100~~, and the more restrictive offsite exposure criteria of~~.~~ Standard Review Plan, Section 15.0.1~~7.4, Rev. 1~~ (Ref. 3)~~, defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values.~~

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations, the equipment hatch and the personnel air locks. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. For the equipment hatch and personnel air locks, closure capability is provided by a designated trained closure crew and the necessary equipment. The OPERABILITY requirements for LCO 3.3.6, "Containment Purge and Exhaust Isolation Instrumentation," ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and, therefore, meet the assumptions used in the safety achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves

(continued)

BASES

LCO (continued)

are terminated, such that radiological doses are within the acceptance limit.

The equipment hatch and personnel air locks are considered isolable when the following criteria are satisfied:

1. the necessary equipment required to close the hatch and personnel air locks is available,

2. at least 23 feet of water is maintained over the top of the reactor vessel flange in accordance with Specification 3.9.6,

3. a designated trained closure crew is available.

The equipment hatch and personnel air locks door openings must be capable of being cleared of any obstruction so that closure can be achieved as soon as possible.

The containment personnel air lock and emergency personnel air lock doors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided that one door in each air lock is capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, one door in each personnel air lock will be closed following an evacuation of containment.

The closure of the equipment hatch and the personnel air locks will be completed promptly following a fuel handling accident within containment.

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 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and 2) special individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.2 (continued)

isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

SR 3.9.3.3

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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, Section 15.4.5.
3. NUREG-0800, Section 15.0.1~~7.4~~, Rev. 0~~1~~, July 2000~~1981~~.
4. Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003.

B 3.9 REFUELING OPERATIONS

B 3.9.6 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 (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to less than~~< 25% of~~ 10 CFR 50.67~~100~~ limits (Ref. 4), as well as the more restrictive~~provided by the~~ guidance of Reference 3.

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 1.183~~25~~ (Ref. 1). A minimum water level of 23 ft ~~(Regulatory Position C.1.c of Ref. 1)~~ allows a decontamination factor of 2~~1~~00 ~~(Regulatory Position C.1.g of Ref. 1)~~ to be used in the accident analysis for iodine. ~~This relates to the assumption that 99% 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 pellet to cladding gap is assumed to contain 10% (except I-131 is 12%) of the total fuel rod iodine inventory (Refs. 1 and 6).~~

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 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 within allowable limits (Refs. 3 and 4~~ and 5~~).

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

BASES

SURVEILLANCE REQUIREMENTS	SR 3.9.6.1 (continued) The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. Regulatory Guide 1.183~~25, March 23, 1972~~, July 2000. 2. FSAR, Section 15.4.5. 3. NUREG-0800, Section 15.0.1~~7.4~~. 4. 10 CFR 50.67~~100.10~~. ~~5. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-828, Radiological Consequences of a Fuel Handling Accident, December 1971.~~ ~~6. NUREG/CR 5009.~~

Joseph M. Farley Nuclear Plant – Units 1 and 2
Alternative Source Term
<u>License Amendment Request</u>

Enclosure 4

Operating License and Technical Specification Pages (Retyped)

(5) Updated Final Safety Analysis Report Supplement

The Updated Final Safety Analysis Report supplement, as revised, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed license. Until that update is complete, Southern Nuclear may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Southern Nuclear evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements of that section.

The Southern Nuclear Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to the period of extended operation. Southern Nuclear shall complete these activities no later than June 25, 2017, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

(6) Reactor Vessel Material Surveillance Capsules

All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion.

(7) Upon implementation of Amendment No. xxx adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.10.4, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.18.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.18.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.10.4, in accordance with Specification 5.5.18.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

D. Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," and was submitted on May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 186, as supplemented by a change approved by License Amendment No. 199.

E. This renewed license is subject to the following additional conditions for the protection of the environment:

(1) Southern Nuclear shall operate the facility within applicable Federal and State air and water quality standards and the Environmental Protection Plan (Appendix B).

(2) Before engaging in an operational activity not evaluated by the Commission, Southern Nuclear will prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than evaluated in the Final Environmental Statement, Southern Nuclear shall provide a written evaluation of such activities and obtain prior approval of the Director, Office of Nuclear Reactor Regulation, for the activities.

F. Alabama Power Company shall meet the following antitrust conditions:

(1) Alabama Power Company shall recognize and accord to Alabama Electric Cooperative (AEC) the status of a competing electric utility in central and southern Alabama.

(2) Alabama Power Company shall offer to sell to AEC an undivided ownership interest in Units 1 and 2 of the Farley Nuclear Plant. The percentage of ownership interest to be so offered shall be an amount based on the relative sizes of the respective peak loads of AEC and the Alabama Power Company (excluding from the Alabama Power Company's peak load that amount imposed by members of AEC upon the electric system of Alabama Power Company) occurring in 1976. The price to be paid by AEC for its proportionate share of Units 1 and 2, determined in accordance with the foregoing formula, will be established by the parties through good faith negotiations. The price shall be sufficient to fairly reimburse Alabama Power Company for the proportionate share of its total costs related to the Units 1 and 2 including, but not limited to, all costs of construction, installation, ownership and licensing, as of a date, to be agreed to by the two parties, which fairly accommodates both their respective interests. The offer by Alabama Power Company to sell an undivided ownership interest in Units 1 and 2 may be conditioned, at Alabama Power Company's option, on the agreement by AEC to waive any right of partition of the Farley Plant and to avoid interference in the day-to-day operation of the plant.

(3) Alabama Power Company will provide, under contractual arrangements between Alabama Power Company and AEC, transmission services via its electric system (a) from AEC's electric system to AEC's off-system members; and (b) to AEC's electric system from electric systems other than Alabama Power Company's and from AEC's electric system to electric systems other than Alabama Power Company's. The contractual arrangements covering such transmission services shall embrace rates and charges reflecting conventional accounting and ratemaking concepts followed by the Federal Energy Regulatory Commission (or its successor in function) in testing the reasonableness of rates and charges for transmission services. Such contractual arrangements shall contain provisions protecting Alabama Power Company against economic detriment resulting from transmission line or transmission losses associated therewith.

(4) Alabama Power Company shall furnish such other bulk power supply services as are reasonably available from its system.

(5) Alabama Power Company shall enter into appropriate contractual arrangements amending the 1972 Interconnection Agreement as last amended to provide for a reserve sharing arrangement between Alabama Power Company and AEC under which Alabama Power Company will provide reserve generating capacity in accordance with practices applicable to its responsibility to the operating companies of the Southern Company System. AEC shall maintain a minimum level expressed as a percentage of coincident peak one-hour kilowatt load equal to the percent reserve level similarly expressed for Alabama Power Company as determined by the Southern Company System under its minimum reserve criterion then in effect. Alabama Power Company shall provide to AEC such data as needed from time to time to demonstrate the basis for the need for such minimum reserve level.

(6) Alabama Power Company shall refrain from taking any steps, including but not limited to, the adoption of restrictive provisions in rate filings or negotiated contracts for the sale of wholesale power, that serve to prevent any entity or group of entities engaged in the retail sale of firm electric power from fulfilling all or part of their bulk power requirements through self-generation or through purchases from some other source other than Alabama Power Company. Alabama Power Company shall further, upon request and subject to reasonable terms and conditions, sell partial requirements power to any such entity. Nothing in this paragraph shall be construed as preventing an applicant from taking reasonable steps, in accord with general practice in the industry, to ensure that the reliability of its system is not endangered by any action called for herein.

(7) Alabama Power Company shall engage in wheeling for and at the request of any municipally-owned distribution system:

a. of electric energy from delivery points of Alabama Power Company to said distribution system(s); and

b. of power generated by or available to a distribution system as a result of its ownership or entitlement[2] in generating facilities, to delivery points of Alabama Power Company designated by the distribution system.

Such wheeling services shall be available with respect to any unused capacity on the transmission lines of Alabama Power Company, the use of which will not jeopardize Alabama Power Company's system. The contractual arrangements covering such wheeling services shall be determined in accordance with the principles set forth in Condition (3) herein.

[2] "Entitlement" includes, but is not limited to, power made available to an entity pursuant to an exchange agreement.

Alabama Power Company shall make reasonable provisions for disclosed transmission requirements of any distribution system(s) in planning future transmission. "Disclosed" means the giving of reasonable advance notification of future requirements by said distribution system(s) utilizing wheeling services to be made available by Alabama Power Company.

(8) The foregoing conditions shall be implemented in a manner consistent with the provisions of the Federal Power Act and the Alabama Public Utility laws and regulations thereunder and all rates, charges, services or practices in connection therewith are to be subject to the approval of regulatory agencies having jurisdiction over them.

Southern Nuclear shall not market or broker power or energy from Joseph M. Farley Nuclear Plant, Units 1 and 2. Alabama Power Company shall continue to be responsible for compliance with the obligations imposed on it by the antitrust conditions contained in this paragraph 2.F. of the renewed license. Alabama Power Company shall be responsible and accountable for the actions of its agent, Southern Nuclear, to the extent said agent's actions may, in any way, contravene the antitrust conditions of this paragraph 2.F.

G. <u>Mitigation Strategy License Condition</u>

The licensee shall develop and maintain strategies for addressing large fires and explosions that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

H. In accordance with the requirement imposed by the October 8, 1976 order of the United States Court of Appeals for the District of Columbia Circuit in Natural Resources Defense Council vs. Nuclear Regulatory Commission, No. 74-1385 and 74-1586, that the Nuclear Regulatory Commission "shall make any licenses granted between July 21, 1976 and such time when the mandate is issued subject to the outcome of such proceeding herein," this renewed license shall be subject to the outcome of such proceedings.

I. This renewed operating license is effective as of the date of issuance and shall expire at midnight on June 25, 2037.

FOR THE NUCLEAR REGULATORY COMMISSION

J. E. Dyer

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A - Technical Specifications
2. Preoperational Tests, Startup Tests and Other Items Which Must Be Completed Prior to Proceeding to Succeeding Operational Modes
3. Appendix B - Environmental Protection Plan
4. Appendix C - Additional conditions

Date of Issuance: May 12, 2005

to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

c. Transition License Conditions

1) Before achieving full compliance with 10 CFR 50.48(c), as specified by 2) below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2) above.

2) The licensee shall implement the modifications to its facility, as described in Attachment S, Table S-2, "Plant Modifications Committed," of SNC letter NL-14-1273, dated August 29, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by November 6, 2017. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3) The licensee shall implement the items as listed in Attachment S, Table S-3, "Implementation Items," of SNC letter NL-14-1273, dated August 29, 2014, within 180 days after NRC approval, except for items 30 and 32. Items 30 and 32 shall be implemented by February 6, 2018.

(7) Upon implementation of Amendment No. xxx adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.10.4, in accordance with TS 5.5.18.c.(i), the assessment of CRE habitability as required by Specification 5.5.18.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.18.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.10.4, in accordance with Specification 5.5.18.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

(8) Deleted per Amendment 144
(9) Deleted per Amendment 144
(10) Deleted per Amendment 144
(11) Deleted per Amendment 144
(12) Deleted per Amendment 144
(13) Deleted per Amendment 144
(14) Deleted per Amendment 144
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(16) Deleted per Amendment 144
(17) Deleted per Amendment 144
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(19) Deleted per Amendment 144
(20) Deleted per Amendment 144
(21) Deleted per Amendment 144

(22) <u>Additional Conditions</u>
The Additional conditions contained in Appendix C, as revised through Amendment No. 137, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the additional conditions.

(23) <u>Updated Final Safety Analysis Report</u>

The Updated Final Safety Analysis Report supplement shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed license. Until that update is complete, Southern Nuclear may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Southern Nuclear evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements of that section.

The Southern Nuclear Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to the period of extended operation. Southern Nuclear shall complete these activities no later than June 25, 2017, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

(24) Reactor Vessel Material Surveillance Capsules

All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion.

D. Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," and was submitted on May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 181, as supplemented by a change approved by License Amendment No. 195.

E. Deleted per Amendment 144

F. Alabama Power Company shall meet the following antitrust conditions:

(1) Alabama Power Company shall recognize and accord to Alabama Electric Cooperative (AEC) the status of a competing electric utility in central and southern Alabama.

(2) Alabama Power Company shall offer to sell to AEC an undivided ownership interest in Units 1 and 2 of the Farley Nuclear Plant. The percentage of ownership interest to be so offered shall be an amount based on the relative sizes of the respective peak loads of AEC and Alabama Power Company (excluding from the Alabama Power Company's peak load that amount imposed by members of AEC upon the electric system of Alabama Power Company) occurring in 1976.

The price to be paid by AEC for its proportionate share of Units 1 and 2, determined in accordance with the foregoing formula, will be established by the parties through good faith negotiations. The price shall be sufficient to fairly reimburse Alabama Power Company for the proportionate share of its total costs related to the Units 1 and 2 including, but not limited to, all costs of construction, installation, ownership and licensing, as of a date, to be agreed to by the two parties, which fairly accommodates both their respective interests. The offer by Alabama Power Company to sell an undivided ownership interest in Units 1 and 2 may be conditioned, at Alabama Power Company's option, on the agreement by AEC to waive any right of partition of the Farley Plant and to avoid interference in the day-to-day operation of the plant.

(3) Alabama Power Company will provide, under contractual arrangements between Alabama Power Company and AEC, transmission services via its electric system (a) from AEC's electric system to AEC's off-system members; and (b) to AEC's electric system from electric systems other than Alabama Power Company's, and from AEC's electric system to electric systems other than Alabama Power Company's. The contractual arrangements covering such transmission services shall embrace rates and charges reflecting conventional accounting and ratemaking concepts followed by the Federal Energy Regulatory Commission (or its successor in function) in testing the reasonableness of rates and charges for transmission services. Such contractual arrangements shall contain provisions protecting Alabama Power Company against economic detriment resulting from transmission line or transmission losses associated therewith.

(4) Alabama Power Company shall furnish such other bulk power supply services as are reasonably available from its system.

(5) Alabama Power Company shall enter into appropriate contractual arrangements amending the 1972 Interconnection Agreement as last amended to provide for a reserve sharing arrangement between Alabama Power Company and AEC under which Alabama Power Company will provide reserve generating capacity in accordance with practices applicable to its responsibility to the operating companies of the Southern Company System. AEC shall maintain a minimum level expressed as a percentage of coincident peak one-hour kilowatt load equal to the percent reserve level similarly expressed for Alabama Power Company as determined by the Southern Company System under its minimum reserve criterion then in effect. Alabama Power Company shall provide to AEC such data as needed from time to time to demonstrate the basis for the need for such minimum reserve level.

(6) Alabama Power Company shall refrain from taking any steps, including but not limited, to the adoption of restrictive provisions in rate filings or negotiated contracts for the sale of wholesale power, that serve to prevent any entity or group of entities engaged in the retail sale of firm electric power from fulfilling all or part of their bulk power requirements through self-generation or through purchases from some other source other than Alabama Power Company.

Alabama Power Company shall further, upon request and subject to reasonable terms and conditions, sell partial requirements power to any such entity. Nothing in this paragraph shall be construed as preventing an applicant from taking reasonable steps, in accord with general practice in the industry, to ensure that the reliability of its system is not endangered by any action called for herein.

(7) Alabama Power Company shall engage in wheeling for and at the request of any municipally-owned distribution system:

a. of electric energy from delivery points of Alabama Power Company to said distribution system(s); and

b. of power generated by or available to a distribution system as a result of its ownership or entitlement[2] in generating facilities, to delivery points of Alabama Power Company designated by the distribution system.

Such wheeling services shall be available with respect to any unused capacity on the transmission lines of Alabama Power Company, the use of which will not jeopardize Alabama Power Company's system. The contractual arrangements covering such wheeling services shall be determined in accordance with the principles set forth in Condition (3) herein.

Alabama Power Company shall make reasonable provisions for disclosed transmission requirements of any distribution system(s) in planning future transmission. "Disclosed" means the giving of reasonable advance notification of future requirements by said distribution system(s) utilizing wheeling services to be made available by Alabama Power Company.

(8) The foregoing conditions shall be implemented in a manner consistent with the provisions of the Federal Power Act and the Alabama Public Utility laws and regulations thereunder and all rates, charges, services or practices in connection therewith are to be subject to the approval of regulatory agencies having jurisdiction over them.

Southern Nuclear shall not market or broker power or energy from Joseph M. Farley Nuclear Plant, Units 1 and 2. Alabama Power Company shall continue to be responsible for compliance with the obligations imposed on it by the antitrust conditions contained in this paragraph 2.F. of the

[2] "Entitlement" includes, but is not limited to, power made available to an entity pursuant to an exchange agreement.

renewed license. Alabama Power Company shall be responsible and accountable for the actions of its agent, Southern Nuclear, to the extent said agent's actions may, in any way, contravene the antitrust conditions of this paragraph 2.F.

G. The facility requires relief from certain requirements of 10 CFR 50.55a(g) and exemptions from Appendices G, H and J to 10 CFR Part 50. The relief and exemptions are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 5. They are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, the relief and exemptions are hereby granted. With the granting of these relief and exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

H. Southern Nuclear shall immediately notify the NRC of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

I. <u>Mitigation Strategy License Condition</u>

The licensee shall develop and maintain strategies for addressing large fires and explosions that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

J. Alabama Power Company shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

K. This renewed operating license is effective as of the date of issuance and shall expire at midnight on March 31, 2041.

FOR THE NUCLEAR REGULATORY COMMISSION

J. E. Dyer

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachment:

1. Appendix A - Technical Specifications (NUREG-0697, as revised)
2. Appendix B - Environmental Protection Plan
3. Appendix C - Additional conditions

Date of Issuance: May 12, 2005

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Filtration/Pressurization System (CREFS)

LCO 3.7.10 Two CREFS trains shall be OPERABLE.

--- NOTE ---
The control room envelope (CRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies,
During CORE ALTERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREFS train inoperable for reasons other than Condition B.	A.1	Restore CREFS train to OPERABLE status.	7 days
B. CRE or more CREFS trains inoperable due to inoperable CRE boundary.	B.1	Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>		
	B.2.	Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	<u>AND</u>		
	B.3	Restore CRE boundary to OPERABLE status.	90 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 ------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. --------------------------------- Be in MODE 4.	6 hours 12 hours
D. Two CREFS trains inoperable in MODE 1, 2, 3, OR 4.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	E.1 Place OPERABLE CREFS train in emergency recirculation mode. OR E.2.1 Suspend CORE ALTERATIONS. AND E.2.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition B not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS. <u>OR</u> Two CREFS trains inoperable during movement of irradiated fuel assemblies or during CORE ALTERATIONS. <u>OR</u> One or more CREFS trains inoperable due to an inoperable CRE boundary during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	F.1 Suspend CORE ALTERATIONS. <u>AND</u> F.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREFS Pressurization train with the heaters operating and each CREFS Recirculation and Filtration train for $\geq$ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Perform required CREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.3	-----NOTE----- Not required to be performed in MODES 5 and 6. ----- Verify each CREFS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 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. One door in each air lock is capable of being closed; 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 an OPERABLE Containment Purge and Exhaust Isolation System.

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

APPLICABILITY: During CORE ALTERATIONS,
During movement of 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
	<u>AND</u>	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

5.5 Programs and Manuals

5.5.18 Control Room Envelope Habitability Program

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

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

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

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

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREFS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

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

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

(continued)

5.5 Programs and Manuals

5.5.19 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

Joseph M. Farley Nuclear Plant – Units 1 and 2
Alternative Source Term
License Amendment Request

Enclosure 5

Regulatory Guide 1.183 Conformance Tables

REGULATORY GUIDE 1.183 COMFORMANCE TABLES

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP 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.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
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 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.	Conforms – See RG Section 1.3 discussions.
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 FNP 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.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
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 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.	Conforms – No changes are proposed in this License 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	Not Applicable – This License Amendment Request is for full scope AST implementation for the radiological dose consequences of the major FNP DBA.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
	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 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.	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, FNP will continue

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
	• 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) • Emergency Response Facilities (NUREG-0737, III.A.1.2) • Control Room Habitability (NUREG-0737, III.D.3.4)	to meet the NUREG-0654 Planning Standard for Emergency Facilities and Equipment as described in the FNP Emergency Plan. Design Basis dose calculations for non-control room Emergency Response Facilities, such as the Technical Support Center, are not part of the FNP 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	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	FNP Analysis
	such that the results, or the conclusions drawn 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	FNP 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 FNP 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	Not Applicable- The FNP AST analysis does not rely on sensitivity or scoping analyses.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
	the previous source term 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 FNP 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 FNP DBAs.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
1.3.5	Current environmental qualification (EQ) analyses may be impacted by a proposed plant modification associated with the AST implementation. The EQ 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.	Conforms – The FNP AST License Amendment Request is not proposing to modify the equipment qualification design basis to adopt AST. The FNP 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 FNP 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 FNP AST License Amendment Request is not utilizing risk insights as a basis for any proposed changes.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
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 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.	Conforms- The License Amendment Request is formatted in accordance with accepted NRC/industry guidance. The request describes the radiological and nonradiological impacts of the FNP 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.	The LOCA dose calculation was performed using RADTRAD 3.10. The FHA dose calculations uses RADTRAD Version 3.03. The MSLB, Control Rod Ejection, and Locked Rotor dose calculations were performed using the Bechtel standard computer program LocaDose, Version 7.1. The SGTR dose calculation was performed using LocaDose, Version 7.11. LocaDose is designed to calculate radioactive isotope activities within

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
		regions, radioactive releases from regions, doses and dose rates within regions for humans and equipment, and inhalation and immersion doses and dose rates at offsite locations to plant personnel and the general public.
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 FNP.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
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, FHA, 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. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.	Conforms- This License Amendment Request considers a number of release scenarios, as applicable, for the 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 utilizes RADTRAD and Bechtel LocaDose 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 FNP AST dose calculations have been developed by industry experts and reviewed and accepted by SNC Engineering. The calculation was 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	FNP 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.8 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.9 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 FNP 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, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	The analysis of the FHA considers radioactive decay between the time of core shutdown and the beginning of fuel movement.

<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>FNP Analysis</th></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>
</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 the RCS, 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	FNP Analysis
	Other Noble Gases 0.05 Other Halogens 0.05 Alkali Metals 0.12	
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	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	FNP Analysis
	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	
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	Conforms – The AST dose consequences are calculated in TEDE.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
	components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.[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	Conforms - The TEDE was determined for the most limiting person at the EAB. The maximum two-hour TEDE was determined by calculating the postulated dose for a series of small time increments and performing a

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
	two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).	'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 24" thick. Accordingly, shielding from the walls and the filter unit casings prevents an appreciable

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
		dose to the operators during the accident. The control room is surrounded by the Auxiliary Building (and so does not abut the containment), and is shielded from containment by more than 2 feet of concrete in all directions. The containment walls are 3'9" thick as well. Accordingly, the control room is adequately shielded from containment shine, as well as shine from containment leakage sources. With respect to shine from the release plume, the exterior Auxiliary Building surrounds the control room and the exterior concrete walls are approximately 21" thick. The floors, walls, and ceilings of the control room add to the concrete shielding from the plume. Therefore, shine from the release plume to the control room occupants will not be significant. For the Fuel Handling Accident scenario where the Personnel Airlock is open, the Auxiliary Building area around the control room could become contaminated. A small section of the control room envelope wall is only 1 foot thick inside the Auxiliary Building

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
		(between the control room and an interior hallway). Doses to the control room operators due to shine from the contaminated area through the 1 foot thick wall are included in the Fuel Handling Accident evaluation of control room doses and were found to be not significant.
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 EPZ 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. Shielding models used in the FHA have been structured to provide suitably conservative estimates of the exposure to CR 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	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.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
	advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	
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. Equation 1 from Regulatory Guide 1.183 is used for finite cloud correction when calculating the DDE immersion doses due to airborne activity inside the control room in the Fuel Handling Accident.
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	Not Applicable – This full scope AST implementation LAR is for the radiological consequences of major FNP DBAs.

<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>FNP Analysis</th></tr>
<tr><td></td><td>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.</td><td></td></tr>
<tr><td>4.4</td><td>The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.

Table 6[17]
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>
</table>
</td><td>The EAB and LPZ acceptance criteria from Table 6 of RG 1.183 are applied. The control room acceptance of 5 rem 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	FNP Analysis
	Coincident Iodine Spike | 2.5 rem TEDE | PWR Locked Rotor Accident | 2.5 rem TEDE | Until cold shutdown is established PWR Rod Ejection Accident | 6.3 rem TEDE | 30 days for containment pathway; until cold shutdown is established for secondary pathway Fuel Handling Accident | 6.3 rem TEDE | 2 hours The column labeled "Analysis Release Duration" is a summary of the assumed radioactivity release durations identified in the individual appendices to this guide. Refer to these appendices for complete descriptions of the release pathways and durations.	
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 FNP AST dose calculations were prepared and accepted 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	Not Applicable- This License Amendment Request is not proposing deviations to conformance with this

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
	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.	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 actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms - Only safety-related Engineered Safety Features are credited in the analysis 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	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.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
	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.	
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 calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Conforms- The FNP DBA analysis assumptions and methods are compatible with the AST 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.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
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.
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 in License Amendments 165/157 and 166/158.
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	Not Applicable – The X/Q values used are those described in the FNP FSAR.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
	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.	
6.0	The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 11), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective. The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs TID14844) on EQ doses pending the outcome of the evaluation of the generic issue.	Conforms – FNP is retaining the use of the 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	Conforms – No modifications are being proposed as part of this AST License Amendment Request.

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
	loading sequence, integrated doses to equipment in the containment, and more.	
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 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 FNP.
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.	Conforms – Burnup does not exceed 54 GWD/MTU at FNP.
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 – Both RADTRAD and LOCADOSE can model the release either in a linear ramp manner, or

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
		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 analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 23) and RADTRAD (Ref. 24) incorporate suitable methodologies.	Conforms – The iodine protection factor methodology of 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 Penetration Room Filters (PRF), 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 inefficiency allowed by TS 5.5.11 (εTS)

Table A: Conformance With Regulatory Guide 1.183 Section C		
RG Section	RG Position	FNP Analysis
		and further reduce efficiency by a 0.5% bypass amount). Typically, the efficiency (εf) is calculated as follows: $\varepsilon f = (1 - 2 * (1 - \varepsilon TS) - 0.005)$ Example: PRF Charcoal Efficiency = (1-2*(1-0.95)-0.005) = .895 = 89.5% This methodology assures compliance with Technical Specification 5.5.11 requirements and US NRC RG-1.52.
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 FNP.

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	FNP 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 analysis was previously reviewed by the NRC in FNP License Amendment 166/158.
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 containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Conforms - An aerosol natural deposition rate of 0.1 h^{r-1} is assumed based upon values presented Section VI of NUREG/CR-6189.

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	FNP Analysis
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 and particulate iodine 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 12,045 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 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).	Conforms - Elemental and aerosol removal 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.

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	FNP Analysis
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 - FNP is a PWR.
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 FNP Technical Specifications. It is reduced to 50% of that value after 24 hours.
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. FNP is a PWR.

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	FNP Analysis
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 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.	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 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. FNP 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 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. 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	The FNP Technical Specifications do not provide a specific limit for operational leakage from ECCS systems. However, administrative limits ensure that operational leakage is adequately controlled. In the analysis, an assumed leakage from ECCS systems is taken as 20,000 cc/hr for leakage of sump water outside

Table B: Conformance With Regulatory Guide 1.183 Appendix A (Loss of Coolant Accident)		
RG Section	RG Position	FNP Analysis
	refueling water storage tank.	of containment into the Auxiliary Building, which is multiplied by two, consistent with this Regulatory Position. In addition, two times the assumed leak rate of 1.0 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 systems 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, 212°F); and h_{fg} is the heat of vaporization at 212°F.	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	FNP 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. FNP 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 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).	Conforms – FNP 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	FNP Analysis
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	FNP 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, the Fuel Handling Building, or the Auxiliary Building without interaction with any other fuel assemblies. The number of fuel rods damaged is equal to one fuel assembly.
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.	Conforms - The fission product release is equal to the gap release, with isotopic fractions as given in Table 3 of RG 1.183 (8% for I-131, 10% for Kr-85, and 5% for the other Halogens and Noble Gases). Cycle to cycle fuel load variations are accounted for with adjustments to the core source term: +15% for Kr-85, +5% for Xe-133, and +3% for the other isotopes.
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.

Table C: Conformance With Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	RG Position	FNP Analysis
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, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).	Conforms – Water level is greater than 23 feet for each case. Therefore, the pool water is assumed to have a decontamination factor of 500 for iodine isotopes in an organic form. This assumption leads to an overall effective 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 containment and the Fuel Handling Building, the FNP 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 - A reduction in the amount of radioactive material released from the spent fuel pool area of the Auxiliary Building is credited by use of the Penetration Room Filter (PRF) system. This system meets the requirements of Regulatory Guide 1.52 and is required to be in service prior to the movement of irradiated fuel in the building. It is analyzed as actuating from the source term seen from the FHA.

Table C: Conformance With Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	RG Position	FNP 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 - There is no credit taken for mixing or dilution in the spent fuel pool area of the Auxiliary Building.
B-5.1	If the containment is isolated[2] 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,[1] no radiological consequences need to be analyzed.	Not Applicable –The containment equipment hatch and personnel airlock are modeled as being open during an FHA and no credit it taken in the analysis for closing them.
B-5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Conforms – The FHA radiological release is over a two-hour period.
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.[1]	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	The free volume of the FNP containment is 2.0E6 cubic feet. The free volume used in the FHA dose

Table C: Conformance With Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)		
RG Section	RG Position	FNP Analysis
	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.	calculation was 1.0E6 cubic feet.
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 demonstrates that a sufficient concentration of radioactivity occurs at the Control Room Ventilation System sensor to result in control room isolation within the assumed 60 second delay time.
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.	Conforms – FNP TS 3.9.3 establishes the requirements for containment penetrations during refueling operations. The FHA dose calculation takes no credit for manual isolation of containment after the event.

Table D: Conformance With Regulatory Guide 1.183 Appendix E (Main Steam Line Break Accident)		
RG Section	RG Position	FNP 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 FNP 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 FNP 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 equilibrium iodine concentration specified in the technical specifications. The iodine spike duration is 8 hours. For conservatism, the concurrent iodine spike is assumed even with initial RCS activity from 1% leaking fuel.

Table D: Conformance With Regulatory Guide 1.183 Appendix E (Main Steam Line Break Accident)		
RG Section	RG Position	FNP 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 leaking 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, including failed (leaking) fuel.
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 – FNP is licensed to ARC. The assumed primary-to-secondary leak rate in the two intact steam generators are 0.65 gpm (936 gallons per day). This is conservative relative to FNP 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 Reactor Coolant System is reduced to cold shutdown conditions in 8 hours.

Table D: Conformance With Regulatory Guide 1.183 Appendix E (Main Steam Line Break Accident)		
RG Section	RG Position	FNP 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 0.1%.
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	Conforms – The steam generator with the faulted main steamline in the MSLB

Table D: Conformance With Regulatory Guide 1.183 Appendix E (Main Steam Line Break Accident)		
RG Section	RG Position	FNP Analysis
	reactor trip (Ref. E-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	accident is assumed to blow completely dry, causing a 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 – FNP 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 FNP current licensing basis a failed (leaking) fuel term is conservatively included with the two cases of iodine spiking.

Table E: Conformance With Regulatory Guide 1.183 Appendix F (Steam Generator Tube Rupture Accident)		
RG Section	RG Position	FNP 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.	Consistent with the FNP current licensing basis a failed (leaking) fuel term is conservatively included with the two cases of iodine spiking.
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 FNP 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 equilibrium iodine concentration specified in the technical specifications. Initial RCS activity conservatively includes a 1% leaking fuel source term, consistent with the FNP current licensing basis for this event. An 8-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	FNP 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 two intact steam generators are 0.65 gpm (936 gallons per day). This is conservative relative to FNP 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 cold shutdown is established at 8 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. However, continued feedwater flow is modeled with its secondary side iodine contribution for conservatism.
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 - The transport model described in Position 5.5 and 5.6 of Appendix E is applied to releases from the steam generators.

Table E: Conformance With Regulatory Guide 1.183 Appendix F (Steam Generator Tube Rupture Accident)		
RG Section	RG Position	FNP Analysis
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, "Steam Generator Tube Integrity" (USNRC, December 1998), for acceptable assumptions and methodologies for performing radiological analyses.	Conforms – FNP is licensed to ARC.
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.	Consistent with the FNP current licensing basis, the initial RCS activity conservatively considers 1% leaking fuel with the two cases of iodine spiking.

Table F: Conformance With Regulatory Guide 1.183 Appendix G (Locked Rotor Accident)		
RG Section	RG Position	FNP 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 is less than 100°C (212°F). The release of	Conforms – The accident terminates after 8 hours and cold shutdown conditions have been achieved.

Table F: Conformance With Regulatory Guide 1.183 Appendix G (Locked Rotor Accident)		
RG Section	RG Position	FNP Analysis
	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. However, continued feedwater flow is modeled with its secondary side iodine contribution for conservatism.
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 – FNP is licensed to ARC.

Table G: Conformance With Regulatory Guide 1.183 Appendix H (Rod Ejection Accident)		
RG Section	RG Position	FNP 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. For releases from containment involve fuel melting, 100% of the noble gases and 50 % of the iodine isotopes contained in the portion of the fuel which melts is available for release from containment and to the RCS for the secondary release pathway.
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 and fuel cladding damage 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 and fuel cladding damage instantaneously reaches the RCS at the onset of the accident and is

Table G: Conformance With Regulatory Guide 1.183 Appendix H (Rod Ejection Accident)		
RG Section	RG Position	FNP Analysis
		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 will not necessarily be activated in this event, no credit is taken for pH being controlled at values of 7 or greater.
H-5	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	The containment distribution was used for the secondary system pathway in the CREA model. This distribution, although different from RG 1.183, Appendix H, Section 5, is acceptable because the removal mechanism for all chemical forms of iodine is the same for this pathway.
H-6.1	A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.	Conforms - Radioactive material removal from the containment atmosphere by sprays and other engineered safety features is not credited. Natural deposition of elemental iodine is credited.
H-6.2	The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric	Conforms - The containment is assumed to leak to the environment at the technical specification limit of 0.15%/day for the first 24 hours of the accident and half this rate thereafter.

Table G: Conformance With Regulatory Guide 1.183 Appendix H (Rod Ejection Accident)		
RG Section	RG Position	FNP Analysis
	containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.	
H-7.1	A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms - The total leakage from the primary system to the secondary system is assumed to be 1 gpm, conservatively bounding the technical specification limit of 150 gpd per generator. This leakage lasts for the first 2500 sec of the accident and is conservatively modeled as being direct to the environment.
H-7.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms - The water density of both the primary and secondary coolants is assumed to be 62.4 lbm/ft^3.
H-7.3	All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Conforms - It is assumed that noble gases are not retained in the secondary water.
H-7.4	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms - The transport model described in Position 5.5 and 5.6 of Appendix E is applied to releases from the steam generators.
Footnote H-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 – FNP is licensed to ARC.

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Enclosure 6

Loss-of-Coolant Accident Analysis

LOSS-OF-COOLANT ACCIDENT DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number: SM-1080538201-003, Version 3

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 FNP Current Licensing Basis (CLB).

Results and Acceptance Limits

Release	EAB (rem TEDE)	LPZ (rem TEDE)	Control Room (rem TEDE)
Containment Purge	0.001	0.0004	0.002
Containment Leakage	12.9	5.6	3.6
ECCS Leakage	0.25	0.23	0.81
RWST Back-leakage	0.13	0.13	0.26
Total	**13.2**	**6.0**	**4.7**
Acceptance Limit	25	25	5

(Note that rounding is applied to all values)

Key Assumptions and Inputs

Source Term Parameters

Parameter	Value
Reactor Power Level:	2775 MWt (+2% uncertainty = 2831 MWt)

Table 1 - Core Cycle-to-Cycle Augments

Isotope	Factor
Kr-85	1.15
Xe-133	1.05
Cs-134	1.35
Cs-136	1.25
Cs-137	1.20
Halogens, Other Noble Gases and Particulates	1.03

Table 2 - Core Source Term

Nuclide	Activity (Ci)	Core Factor	Adjusted Activity (Ci)	Nuclide	Activity (Ci)	Core Factor	Adjusted Activity (Ci)
Co-58	0.00E+00	1.03	0.00E+00	Te-134	1.30E+08	1.03	1.34E+08
Co-60	0.00E+00	1.03	0.00E+00	I-130	2.50E+06	1.03	2.58E+06
Br-82	3.80E+05	1.03	3.91E+05	I-131	7.50E+07	1.03	7.73E+07
Br-83	9.70E+06	1.03	9.99E+06	I-132	1.10E+08	1.03	1.13E+08
Br-84	1.70E+07	1.03	1.75E+07	I-133	1.60E+08	1.03	1.65E+08
Kr-83m	9.70E+06	1.03	9.99E+06	I-134	1.70E+08	1.03	1.75E+08
Kr-85	7.20E+05	1.15	8.28E+05	I-135	1.50E+08	1.03	1.55E+08
Kr-85m	2.10E+07	1.03	2.16E+07	Xe-131m	8.40E+05	1.03	8.65E+05
Kr-87	4.00E+07	1.03	4.12E+07	Xe-133	1.50E+08	1.05	1.58E+08
Kr-88	5.70E+07	1.03	5.87E+07	Xe-133m	4.80E+06	1.03	4.94E+06
Rb-86	1.40E+05	1.03	1.44E+05	Xe-135	3.50E+07	1.03	3.61E+07
Rb-89	7.40E+07	1.03	7.62E+07	Xe-135m	3.00E+07	1.03	3.09E+07
Sr-89	7.70E+07	1.03	7.93E+07	Xe-138	1.30E+08	1.03	1.34E+08
Sr-90	5.70E+06	1.03	5.87E+06	Cs-134	1.10E+07	1.35	1.49E+07
Sr-91	9.50E+07	1.03	9.79E+07	Cs-134m	3.60E+06	1.03	3.71E+06
Sr-92	1.00E+08	1.03	1.03E+08	Cs-136	3.30E+06	1.25	4.13E+06
Y-90	5.90E+06	1.03	6.08E+06	Cs-137	7.60E+06	1.20	9.12E+06
Y-91	9.90E+07	1.03	1.02E+08	Cs-138	1.40E+08	1.03	1.44E+08
Y-91m	5.50E+07	1.03	5.67E+07	Ba-139	1.40E+08	1.03	1.44E+08
Y-92	1.00E+08	1.03	1.03E+08	Ba-140	1.30E+08	1.03	1.34E+08
Y-93	1.20E+08	1.03	1.24E+08	Ba-141	1.30E+08	1.03	1.34E+08
Y-95	1.30E+08	1.03	1.34E+08	La-140	1.40E+08	1.03	1.44E+08

Nuclide	Activity (Ci)	Core Factor	Adjusted Activity (Ci)	Nuclide	Activity (Ci)	Core Factor	Adjusted Activity (Ci)
Zr-95	1.30E+08	1.03	1.34E+08	La-141	1.30E+08	1.03	1.34E+08
Zr-97	1.30E+08	1.03	1.34E+08	La-143	1.20E+08	1.03	1.24E+08
Nb-95	1.30E+08	1.03	1.34E+08	La-142	1.20E+08	1.03	1.24E+08
Nb-95m	9.40E+05	1.03	9.68E+05	Ce-141	1.30E+08	1.03	1.34E+08
Nb-97	1.30E+08	1.03	1.34E+08	Ce-143	1.20E+08	1.03	1.24E+08
Mo-99	1.40E+08	1.03	1.44E+08	Ce-144	9.40E+07	1.03	9.68E+07
Tc-99m	1.20E+08	1.03	1.24E+08	Pr-143	1.20E+08	1.03	1.24E+08
Ru-103	1.10E+08	1.03	1.13E+08	Nd-147	5.10E+07	1.03	5.25E+07
Ru-105	7.60E+07	1.03	7.83E+07	Pm-147	9.70E+06	1.03	9.99E+06
Ru-106	3.40E+07	1.03	3.50E+07	Pm-148	2.10E+07	1.03	2.16E+07
Rh-103m	1.00E+08	1.03	1.03E+08	Pm-148m	2.30E+06	1.03	2.37E+06
Rh-105	6.90E+07	1.03	7.11E+07	Pm-149	4.60E+07	1.03	4.74E+07
Pd-109	2.20E+07	1.03	2.27E+07	Pm-151	1.50E+07	1.03	1.55E+07
Sb-124	8.80E+04	1.03	9.06E+04	Sm-153	3.30E+07	1.03	3.40E+07
Sb-125	9.40E+05	1.03	9.68E+05	Eu-154	7.10E+05	1.03	7.31E+05
Sb-126	7.90E+04	1.03	8.14E+04	Eu-155	4.60E+05	1.03	4.74E+05
Sb-127	7.90E+06	1.03	8.14E+06	Eu-156	1.20E+07	1.03	1.24E+07
Sb-129	2.40E+07	1.03	2.47E+07	Np-238	2.30E+07	1.03	2.37E+07
Te-125m	2.00E+05	1.03	2.06E+05	Np-239	1.40E+09	1.03	1.44E+09
Te-127	7.80E+06	1.03	8.03E+06	Pu-238	1.60E+05	1.03	1.65E+05
Te-127m	1.00E+06	1.03	1.03E+06	Pu-239	2.20E+04	1.03	2.27E+04
Te-129	2.40E+07	1.03	2.47E+07	Pu-240	3.10E+04	1.03	3.19E+04
Te-129m	3.50E+06	1.03	3.61E+06	Pu-241	8.30E+06	1.03	8.55E+06
Te-131	6.70E+07	1.03	6.90E+07	Pu-243	1.80E+07	1.03	1.85E+07
Te-131m	1.10E+07	1.03	1.13E+07	Am-241	8.20E+03	1.03	8.45E+03
Te-132	1.10E+08	1.03	1.13E+08	Am-242	4.50E+06	1.03	4.64E+06
Te-133	9.10E+07	1.03	9.37E+07	Cm-242	2.20E+06	1.03	2.27E+06
Te-133m	5.80E+07	1.03	5.97E+07	Cm-244	1.30E+05	1.03	1.34E+05

Parameter	Value
Initial RCS Source Term	Accounts for 1% Failed (Leaking) Fuel (limitation from previous operating experience, not accident related).
Initial RCS Source Term	The Iodine concentration is set at the 0.5 µCi/gm
RCS Mass	440,900 lbm
Release Fractions	Per RG-1.183
Release Timing	Per RG-1.183

Table 3 - RCS Source Term

Nuclide	Activity (µCi/gm)
Kr-85	7.70E+00
Kr-85m	1.80E+00
Kr-87	1.20E+00
Kr-88	3.50E+00
I-131	3.528E-01
I-132	5.796E-01
I-133	6.804E-01
I-134	1.588E-01
I-135	4.788E-01
Xe-133	2.40E+02
Xe-135	7.90E+00
Kr-83m	4.50E-01
Br-83	8.80E-02
Br-84	5.00E-02
I-130	2.20E-02
Xe-131m	2.90E+00
Xe-133m	4.60E+00
Xe-135m	4.50E-01
Xe-138	7.20E-01

Containment Leakage Parameters

Parameter	Value
Containment Volume	2.03E6 cubic Feet
Sprayed Volume	1,668,660 cubic feet
Unsprayed Volume	361,240 cubic feet
Containment Leakage	0.15% of volume per day for first 24 Hours 0.075% 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	5.45 hr^{-1} during injection mode

	5.03 hr^{-1} during recirculation mode
Containment Spray Organic removal	None
Natural Deposition, Aerosol only	0.1 hr^{-1} after sprays are terminated
Containment Spray Start	90 seconds
Containment Spray Stop	8 Hours
Containment Spray Flow	2,480 gal/min in injection phase 2,290 gal/min in recirculation phase
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	2850 CFM

ECCS Leakage

Parameter	Value
Sump Volume	49,200 cubic feet
Sump temperature	Varies, max is 265 °F
ECCS Leakage Initiation Time	20 minutes
ECCS Leakage Iodine Flashing Factor	10%
Iodine Species ECCS Leakage Released to the Atmosphere	
Elemental	97%
Organic	3%
ECCS Leakage Rate	40,000 cc/hr

RWST Leakage Parameters

Parameter	Value
ECCS Recirculation Start Time	20 minutes
Iodine Species ECCS Leakage Released to the Atmosphere from the RWST	
Elemental	100%
Organic	0%
ECCS Leakage Rate to the RWST	2 gal/min
RWST Leakage Iodine Flashing Factors	Varies with temperature and pH
RWST Capacity	505,562 gallons
RWST Volume at Transfer to Recirculation	29,002 gallons

CR Parameters:

Parameter	Value
CR Volume	114,000 ft^3
CR Pressurization Mode Initiation	Automatic at 60 Seconds
CR Ventilation System Normal Flow Rate	2340 cfm < 60 seconds
CR Ventilation System Makeup Rate	375 cfm > 60 seconds
CR Ventilation System Recirculation Flow Rate	2700 cfm > 60 seconds
CR Ventilation System Charcoal Filter Efficiencies	
Pressurization Filters	98.5% all iodine species
Recirculation Filters	94.5% elemental and organic 98.5% particulate
CR Unfiltered In-leakage	315 cfm
CR Ingress/Egress Unfiltered In-leakage	10 cfm
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):

Containment Releases

Time (hr)	EAB	LPZ	CR
0 – 2	7.6E-4	2.80E-4	1.66E-03
2 – 8	-	1.10E-4	1.36E-03
8 – 24	-	1.00E-5	6.81E-04
24 – 96	-	5.40E-6	5.60E-04
96 – 720	-	2.90E-6	4.21E-04

Plant Vent Releases

Time (hr)	EAB	LPZ	CR
0 – 0.0167	7.6E-4	2.80E-4	2.79E-03
0.0167 - 2	7.6E-4	2.80E-4	1.65E-03
2 – 8	-	1.10E-4	1.38E-03
8 – 24	-	1.00E-5	7.20E-04
24 – 96	-	5.40E-6	5.47E-04
96 – 720	-	2.90E-6	3.63E-04

RWST Releases

Time (hr)	EAB	LPZ	CR
0 – 2	7.6E-4	2.80E-4	4.97E-04
2 – 8	-	1.10E-4	3.82E-04
8 – 24	-	1.00E-5	1.70E-04
24 – 96	-	5.40E-6	1.28E-04
96 – 720	-	2.90E-6	1.00E-04

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Enclosure 7

Fuel Handling Accident Analysis

FUEL HANDLING ACCIDENT DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number:	SM-1080538201-001, Version 2
Method/Computer Program Used:	RADTRAD Version 3.03
Regulatory Guidance:	RG-1.183, including Appendix B

Model Discussion

The calculation was performed to address a fuel handling accident (FHA) in the containment and in the SFP area of the Auxiliary Building. For the containment accident, the containment equipment hatch and the personnel airlock are presumed to be open and no credit is taken to close them. The open containment airlock could allow areas around the Control Room (CR) to become contaminated, so the calculation accounts for dose impacts of ingress/egress of the CR through the CR doors. Also, a small amount of CR envelope wall is only 1 foot thick, so the shine from the contaminated area through the wall is added to the CR operator dose. Doses in the CR are accumulated over a period of 8 hours. Releases from the damaged fuel are completed in 2 hours.

For the accident in the SFP area of the Auxiliary Building, the accident releases also are completed in 2 hours. The activity is released to the environment through the plant vent stack, and credit is taken for filtration of the iodine isotopes through the Penetration Room Filtration System. Doses from this accident are bounded by the doses from an accident in containment.

Results and Acceptance Limits

Release	EAB (rem TEDE)	LPZ (rem TEDE)	Control Room (rem TEDE)
Containment	2.4	0.9	1.0
Spent Fuel Pool	0.5	0.2	0.2
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	2775 MWt (+2% uncertainty = 2831 MWt)
Reactor Peaking Factor	1.7
Fuel Movement Time	100 hours post shutdown.
Number of Fuel Assemblies	157
Number of Damaged Assemblies	1
Number of Damaged Fuel Rods	264

Table 1 - Core Cycle-to-Cycle Augments

Isotope	Factor
Kr-85	1.15
Xe-133	1.05
Other Noble Gases	1.03
Other Iodine isotopes	1.03

Table 2 - Core Source Term

Isotope	Core Activity at 100 hours post Shutdown (curies)
Kr-85	7.2E+05
Xe-131m	8.1E+05
Xe-133	1.0E+08
Xe-133m	2.0E+06
Xe-135	2.0E+05
I-131	5.4E+07
I-132	4.6E+07
I-133	5.7E+06
I-135	4.1E+03

Fraction of Fission Product Inventory in Gap	
I-131	0.08
Kr -85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Overlaying Pool Depth	23 feet
Pool Decontamination Factor	Elemental: 500 Organic: 1
Iodine Chemical Form	0% Aerosol, 99.85% Elemental 0.15% Organic
Net Decontamination Factor	200

The released activity is obtained by making the product of the100-hour post-shutdown core activity for the isotope, the design margin, the gap fraction, and the radial peaking factor. The product is then divided by the DF and by 157 (the number of assemblies) to achieve the released activity shown in the last column of the table, below.

Table 3 - Net Scrubbed Release Activities

Group	Isotope	100-hr Core Inventory (Ci)	Design Margin	Gap Fraction	Radial Peaking Factor	DF	Released Activity (Curies)
Noble Gases	Kr-85	7.20E+05	1.15	0.1	1.70E+00	1	8.97E+02
	Xe-131m	8.10E+05	1.03	0.05	1.70E+00	1	4.52E+02
	Xe-133	1.00E+08	1.05	0.05	1.70E+00	1	5.68E+04
	Xe-133m	2.00E+06	1.03	0.05	1.70E+00	1	1.12E+03
	Xe-135	2.00E+05	1.03	0.05	1.70E+00	1	1.12E+02
Halogens	I-131	5.40E+07	1.03	0.08	1.70E+00	200	2.41E+02
	I-132	4.60E+07	1.03	0.05	1.70E+00	200	1.28E+02
	I-133	5.70E+06	1.03	0.05	1.70E+00	200	1.59E+01
	I-135	4.10E+03	1.03	0.05	1.70E+00	200	1.14E-02

Containment Release

Parameter	Value
Containment Volume	2.03E6 Cubic Feet
Mixing Volume in Containment	1.0E6 Cubic Feet
Release Duration	2 hours
Containment Hatch Flow Rate	55,000 cfm
Containment Release Filtration	0%
Personnel Airlock Flow Rate	1515 cfm
Auxiliary Building Mixing Volume	37,875 cubic feet
Personnel Airlock Release Filtration	0%
Auxiliary Building Ventilation	1505 cfm to plant vent, 10 cfm to CR for ingress/egress

Spent Fuel Pool Area Release

Parameter	Value
Fuel Handling Volume	72,150 cubic feet
Overlaying Pool Depth	23 feet
Fuel Handling Area Release Rate	5,000 cfm
PRF Filtration	89.5% for iodine isotopes

CR Parameters

Parameter	Value
CR Volume	114,000 ft^3
CR Isolation Mode Initiation	Automatic at 60 Seconds
CR Pressurization Mode Initiation	Manually at 21 minutes (20 minutes after isolation)
CR Ventilation System Normal Flow Rate	2340 cfm < 60 seconds
CR Ventilation Isolation Mode Flow Rate	600 cfm (1 minute to 21 minutes)
CR Ventilation Pressurization Makeup Rate	375 cfm > 21 minutes

CR Ventilation System Recirculation Flow Rate	2700 cfm > 21 minutes
CR Ventilation System Charcoal Filter Efficiencies	
Pressurization Filters	98.5% all iodine species
Recirculation Filters	94.5% all iodine species
CR Pressurization Mode Unfiltered In-leakage	325 cfm*
CR Ingress/Egress Unfiltered In-leakage	10 cfm throughout (location changes)
CR Breathing Rate	3.5E-4 m^3/sec
Occupancy Factors	
0-8 hours	1.0

CR Ventilation Summary

Table 4 – Control Room Ventilation Summary

Time	Filtered Flow (CFM)	Unfiltered Flow (CFM)
0 to 1 minute	0	2340
1 minute to 21 minutes	0	600
21 minutes to 8 hours	375	325

* For the FHA in Containment, the 10 CFM for ingress and egress to the CR goes from the Auxiliary Building to the CR through the CR door. For the FHA in the ea, the 10 cfm for ingress and egress is conservatively added to CR through the ventilation system and is unfiltered. This unfiltered inleakage starts at time 0 and continues through the entire accident (8 hours).

Atmospheric Dispersion Factors (sec/m^3)

Containment Releases:

Time (hr)	EAB	LPZ	CR
0 – 2	7.6E-4	2.80E-4	8.79E-04
2 – 8	-	1.10E-4	6.77E-04

Plant Vent Releases:

Time (hr)	EAB	LPZ	CR
0 – 2	7.6E-4	2.80E-4	1.62E-03
2 – 8	-	1.10E-4	1.37E-03

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Enclosure 8

Main Steam Line Break Accident Analysis

MAIN STEAM LINE BREAK ACCIDENT DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number: SM-1080538201-002, Version 1

Method/Computer Program Used: LocaDose Version 7.1

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 is assumed to be 0.35 gallons per minute (gpm) to the "faulted" steam generator (SG), and 0.65 gpm (total) going to the two 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 two intact SGs released to the environment via steaming until the primary system (RCS) is reduced to cold shutdown conditions (assumed at 24 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	0.94	25
	LPZ	0.37	25
	Control Room	0.23	5
Concurrent Iodine Spike	EAB	0.95	2.5
	LPZ	0.45	2.5
	Control Room	0.45	5

The maximum 2-hour EAB dose occurs between 0 and 2 hours.
(Note that rounding is applied to all values)

Key Assumptions and Inputs

Physical Parameters

Parameter	Value
RCS Mass	440,900 lbm (2.00E8 grams)
RCS Volume	1.02E4 cubic feet
Intact SG Mass	1.68E5 lbm (each, assumed full)
Intact SG Volume	2690 cubic feet (each)
Faulted SG Volume	2690 cubic feet
Coolant Densities	Primary and Secondary water at 62.4 lbm/ft^3

Table 1 – MSLB Flow Rates

Flow Path	Time (hour)		Release (lbm)	Flow	Note
	From	to			
RCS to Env	0	24	-	4.68E-02 cfm	1
RCS to Intact SGs	0	24	-	8.69E-02 cfm	
Feedwater to Intact SGs	0	2	4.81E+05	1.09E+08 g/hr	2
	2	8	7.83E+05	5.92E+07 g/hr	
	8	24	1.04E+06	2.96E+07 g/hr	
Intact SGs to Env	0	2	3.48E+05	4.65E+01 cfm	3
	2	8	7.74E+05	3.45E+01 cfm	
	8	24	1.04E+06	1.74E+01 cfm	
	0	24	2.17E+06	-	
Faulted SG to Env	0	24	4.83E+05	5.00E+02 cfm	4

Flow Rate Notes:
1. RCS Leakage of 1 gpm – Volumetric leakage (gpm) from RCS is divided by 7.48 gal/ft^3.
2. Feedwater – Mass release from feedwater to intact SGs is multiplied by 1.1. Flow is the release (lbm) multiplied by 453.6 g/lbm and divided by time duration (hr).
3. 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).
4. Faulted SG – Mass release from the faulted SG is multiplied by 1.1. Flow is conservatively high.

Radioactivity Considerations:

No fuel failure occurs as a result of the MSLB.

Iodine Release Species: 97% elemental, 3% organic.

RCS activity includes an assumption of normal operations 1% failed (leaking) fuel in accordance with past CLB (Affects alkali metals)

Initial RCS Activity:	
Pre-Accident Spike Case	30 µCi/gm DEI
Concurrent Accident Spike	0.5 µCi/gm DEI
Normal Iodine RCS Appearance Rate	
Letdown Flow Rate	145 gal/min
Identified Leakage	10 gal/min
Unidentified Leakage	1 gal/min
Concurrent Accident Spike Appearance	500 times the Normal Rate for the first 8 hours
Concentrations of Iodine in Secondary	Technical Specification limit of 0.1 µCi/gm DEI

Concentrations of Alkali Metals in Secondary - Given 0.1 µCi/gm Dei in the secondary and 0.5 µCi/gm in the RCS, the concentrations of alkali metals are assumed to be 20% of those in the RCS corresponding to 1% failed (leaking) fuel.

Table 2 - Normal RCS Iodine Concentrations

Isotope	1% FF RCS Concen (µCi/g)	Inh TEDE DCF (rem/Ci)	Concen x DCF	Concen (µCi/g) 30 µCi/g DEI	0.5 µCi/g DEI	0.1 µCi/g DEI
I-131	2.5	3.29E+04	8.23E+04	2.27E+01	3.78E-01	7.56E-02
I-132	0.9	3.81E+02	3.43E+02	8.17E+00	1.36E-01	2.72E-02
I-133	4.0	5.85E+03	2.34E+04	3.63E+01	6.05E-01	1.21E-01
I-134	0.6	1.31E+02	7.86E+01	5.44E+00	9.07E-02	1.81E-02
I-135	2.2	1.23E+03	2.71E+03	2.00E+01	3.33E-01	6.65E-02
Total			1.09E+05	9.26E+01	1.54E+00	3.09E-01
Note	2	3	4	5	6	7

Normal RCS Concentration Notes:

1. The note numbers correspond to column numbers.
2. RCS at 1% failed (leaking) fuel – These are the RCS concentrations corresponding to 1% failed (leaking) fuel (FF).
3. DCF – Inhalation TEDE DCFs are from Federal Guidance Report 11, multiplied by 3.7E12 to convert from Sv/Bq to rem/Ci.
4. DCF Weighted Concentration – The concentrations in Column 2 are multiplied by the DCFs in Column 3. The total for this column represents the relative dose corresponding to 1% FF.

5. Concentrations at 30 µCi/g DEI – The relative dose corresponding to 30 µCi/g of I-131 is (30 µCi/g)(3.29E4 rem/Ci) = 9.87E5. In Column 5, the concentrations in Column 2 are multiplied by 9.87E5/1.09E5 to obtain the distribution corresponding to 30 µCi/g DEI.
6. Concentrations at 0.5 µCi/g DEI – The concentrations corresponding to 30 µCi/g in Column 5 are multiplied by 0.5/30.
7. Concentrations at 0.1 µCi/g DEI – The concentrations corresponding to 30 µCi/g in Column 5 are multiplied by 0.1/30.

Initial Iodine Activities in RCS and Secondary Coolant

The initial iodine activities in the RCS and the secondary coolant corresponding to 30, 0.5, and 0.1 µCi/g DEI are shown in the following table. These are entered in LocaDose as initial activities within nodes.

Table 3 - Initial Iodine Activities in the RCS and Secondary Coolant

Isotope	Initial Activity (Ci)			
	RCS Pre Spike (30 µCi/g DEI	RCS Con Spike (0.5 µCi/g DEI)	Intact SGs (0.1 µCi/g DEI)	Faulted SG (0.1 µCi/g DEI)
I-131	4.5E+03	7.6E+01	7.4E+01	1.7E+01
I-132	1.6E+03	2.7E+01	2.7E+01	6.0E+00
I-133	7.3E+03	1.2E+02	1.2E+02	2.7E+01
I-134	1.1E+03	1.8E+01	1.8E+01	4.0E+00
I-135	4.0E+03	6.7E+01	6.5E+01	1.5E+01
Total	1.9E+04	3.1E+02	3.0E+02	6.8E+01
Note	2	3	4	5

Initial Iodine Activities Notes:

1. The note numbers correspond to column numbers.
2. RCS Pre-Accident Spike – The concentrations (µCi/g) in the table above, Column 5 are multiplied by the RCS mass of 2.00E8 g and by 1.0E-6 Ci/µCi.
3. RCS Concurrent Spike – The concentrations (µCi/g) in the table above, Column 6 are multiplied by the RCS mass of 2.00E8 g and by 1.0E-6 Ci/µCi.
4. Intact SGs – The mass of fluid released from the intact SGs is 2.17E6 lbm [Table 1], which is multiplied by 453.6 g/lbm to yield 9.82E8 g. The concentrations (µCi/g) in Table 2, Column 7 are multiplied by this mass and by 1.0E-6 Ci/µCi.
5. Faulted SG – The mass of fluid released from the faulted SG is 4.83E5 lbm [Table 1], which is multiplied by 453.6 g/lbm to yield 2.19E8 g. The concentrations (µCi/g) in Table 2, Column 7 are multiplied by this mass and by 1.0E-6 Ci/µCi.

Table 4 - Noble Gases and Alkali Metals in RCS (1% failed (leaking) fuel)

Isotope	Concen (µCi/g)	Activity (Ci)
Kr-85m	2.2E+00	4.4E+02
Kr-85	5.5E+00	1.1E+03
Kr-87	1.3E+00	2.6E+02
Kr-88	3.8E+00	7.6E+02
Xe-133m	3.2E+00	6.4E+02
Xe-133	2.9E+02	5.8E+04
Xe-135m	2.0E-01	4.0E+01
Xe-135	6.1E+00	1.2E+03
Xe-138	7.0E-01	1.4E+02
Rb-88	3.8E+00	7.6E+02
Rb-89	1.0E-01	2.0E+01
Cs-134	2.6E-01	5.2E+01
Cs-136	1.5E-01	3.0E+01
Cs-137	1.3E+00	2.6E+02
Cs-138	9.6E-01	1.9E+02
Total	3.2E+02	6.4E+04
Note	2	3

Noble Gas and Alkali Metals in RCS Notes:

1. The note numbers correspond to the column numbers.
2. RCS Concentrations – These correspond to a 1% FF Assumption.
3. RCS Activities – The concentrations (µCi/g) in Column 2 are multiplied by the RCS mass of 2.00E8 g and by 1.0E-6 Ci/µCi.

Alkali Metals in the Secondary System

The initial concentrations are assumed to be 20% of the RCS initial activities.

Table 5 - Initial Alkali Metal Concentrations in the Secondary System

Isotope	Concentration (μCi/g)	Activity (Curies)	
		Intact SGs	Faulted SG
Rb-88	7.6E-01	7.5E+02	1.7E+02
Rb-89	2.0E-02	2.0E-01	4.4E+00
Cs-134	5.2E-02	5.1E+01	1.1E+01
Cs-136	3.0E-02	2.9E+01	6.6E+00
Cs-137	2.6E-01	2.6E+02	5.7E+01
Cs-138	1.9E-01	1.9E+02	4.2E+01
Total	1.3E+00	1.3E+03	2.9E+02
Note	2	3	4

Initial Alkali Metals in the Secondary System Notes:

1. The note numbers correspond to the columns of the table.
2. Concentrations – the initial RCS concentrations shown in Table 4 are multiplied by 0.2 to obtain secondary coolant concentrations.
3. Intact SGs Activities – the mass of the fluid released from the intact SGs is 2.17E6 lbm, which is multiplied by 453.6 gm/lbm to yield 9.82E8 grams. The concentrations (in μCi/gm) in Column 2 of this table are multiplied by this mass and then by 1.0E-6 Ci/μCi.
4. Faulted SG Activities – the mass of the fluid released from the faulted SG is 4.83E5 lbm [Table 1], which is multiplied 453.6 gm/lbm to yield 2.19E8 grams. The concentrations (in μCi/gm) in Column 2 of this table are multiplied by this mass and then by 1.0E-6 Ci/μCi.

Radioiodine Appearance Rates

Table 6 - RCS Iodine Appearance Rates

Isotope	Decay Rate (sec^{-1})	Total Removal Rate (hr^{-1})	Appearance Rate (Ci/hr)	
			Normal	Con Spike
I-131	9.98E-07	1.27E-01	9.6E+00	4.8E+03
I-132	8.43E-05	4.27E-01	1.2E+01	5.8E+03
I-133	9.21E-06	1.56E-01	1.9E+01	9.5E+03
I-134	2.20E-04	9.15E-01	1.7E+01	8.3E+03
I-135	2.91E-05	2.28E-01	1.5E+01	7.6E+03
Note	2	3	4	5

RCS Iodine Appearance Rates Notes:

1. The note numbers correspond to the column of the table.
2. Decay rates are from the LocaDose manual
3. Total Removal Rate – the decay rate is multiplied by 3600 sec/hr and added to the clean-up rate of 1.14E-1 hr^{-1} and the leakage rate of 9.42E-3 hr^{-1}.
4. Normal Appearance Rate – the initial iodine activity (Ci) from Table 3 above (column 3) is multiplied by the total removal rate (hr^{-1}) in column 3 of this table.
5. Concurrent Spike Appearance Rate – the normal appearance rate in column 4 of this table is multiplied by 500.

Iodine Appearance Rates in Intact SGs from Feedwater

The flow of feedwater into the intact SGs is modeled as an activity production term in LocaDose. The concentrations corresponding to 0.1 µCi/g from Table 2, Column 7 are multiplied by 1.0E-6 Ci/µCi and by time-dependent flow rates (g/hr) from Table 1, yielding the following appearance rates. These are entered in LocaDose as production terms.

Table 7 - Iodine Appearance Rates in Intact SGs

Isotope	Feedwater Appearance Rate (Ci/hr)		
	0 – 2 Hour	2 – 8 hour	8 – 24 hour
I-131	8.3E+00	4.5E+00	2.2E+00
I-132	3.0E+00	1.6E+00	8.0E-01
I-133	1.3E+01	7.2E+00	3.6E+00
I-134	2.0E+00	1.1E+00	5.4E-01
I-135	7.3E+00	3.9E+00	2.0E+00

Control Room Ventilation Parameters:

Parameter	Value
Pressurization Mode starts	Initiated at start of accident.
CR Make-up Flow Rate	375 cfm (throughout accident)
Pressurization Unfiltered In-leakage	300 cfm (throughout accident)
CR Ingress/egress	10 cfm (throughout accident through CR Vent)
CR Volume	114,000 cubic feet
CR Pressurization Filters:	
HEPA	98.5% for all particulates
Charcoal	98.5% for all iodine species
CR Recirculation Flow	2700 cfm (throughout accident)
Iodine Filter Efficiency	94.5% for organic and elemental 98.5% for particulates
CR Breathing Rates	3.5E-04 m^3/sec for 8 hours
EAB & LPZ Breathing Rates	3.5E-04 m^3/sec for 8 hours
Atmospheric Dispersion Factors:	

Table 8 – Atmospheric Dispersion Factors

Time (hr)	X/Q (sec/m^3) EAB	LPZ	Control Room
0 – 2	**7.60E-4**	**2.80E-4**	**1.66E-3**
2 – 8	-	**1.10E-4**	**1.38E-3**
8 – 24	-	**1.0E-05**	**7.20E-04**

Note: The calculation of record for the MSLB has a typographical error in the assumption section stating that the X/Q for the 8 – 24 hour period at the LPZ is 1.1E-05 sec/m^3. 1.0E-05 sec/m^3 was used in the LocaDose input files.

Joseph M. Farley Nuclear Plant – Units 1 and 2
Alternative Source Term
License Amendment Request

Enclosure 9

Steam Generator Tube Rupture Accident Analysis

STEAM GENERATOR TUBE RUPTURE ACCIDENT DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number:	SM-1080538201-005, Version 2
Method/Computer Program Used:	LocaDose Version 7.11
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 conservative hand-calculation method, in accordance with the FNP 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 low and release Isolation is assumed to occur in 30 minutes for the faulted SG.

Primary to secondary leakage through pin-hole leaks in the SG tubes is assumed at a rate of 1 gpm (0.35 gpm to the faulted SG, 0.65 gpm to the intact SGs) until the SGs are isolated or no longer used for cooling. Activity is released from the other two, intact, generators through steaming via the atmospheric relief valves (ARVs) until the primary system (RCS) is reduced to cold shutdown conditions (assumed at 8 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:

		Dose (rem TEDE)	
Case	Location	Calculated	Limit
Pre-Accident Iodine Spike	EAB	2.4	25
	LPZ	0.92	25
	Control Room	0.48	5
Concurrent Iodine Spike	EAB	0.82	2.5
	LPZ	0.34	2.5
	Control Room	0.17	5

(Note that rounding is applied to all values)

Key Assumptions and Inputs:

Transient Timing

Tube Rupture:	Time Zero (0)
Reactor Trip:	324 seconds
Faulted Generator Isolated	30 min
Break Flow Terminated	30 min
ARV Release from Faulted SG Ended	30 min
RCS cooled to Cold Shutdown	8 hours
Accident Ends	8 hours

Physical Parameters:

Parameter	Value
RCS Mass	440,900 lbm (2.00E8 grams)
RCS Volume	1.02E4 cubic feet
Intact SG Mass	1.05E5 lbm (each, assumed full)
Intact SG Volume	1685 cubic feet (each)
Faulted SG Volume	1685 cubic feet
Coolant Densities	Primary and Secondary water at 62.4 lbm/ft^3

Table 1 – SGTR Flow Rates

Flow Path	Time (hr) From	Time (hr) To	Release (lbm)	Flow	Iodine PF	Note
RCS to Env	0	8	-	4.68E-02 cfm	1.00E+00	1
RCS to Intact SGs	0	8	-	8.69E-02 cfm	1.00E+00	
RCS to Env (Rupture Flow)	0	0.090	3.12E+04	9.27E+01 cfm	4.76E+00	2
	0.090	0.5	1.28E+05	8.32E+01 cfm	6.67E+00	
RCS to Faulted SG (Rupture Flow)	0	0.090	3.12E+04	9.27E+01 cfm	1.27E+00	3
	0.090	0.5	1.28E+05	8.32E+01 cfm	1.18E+00	
Faulted SG to Env	0	0.090	3.67E+05	1.09E+03 cfm	1.00E+02	4
	0.090	0.5	7.90E+04	5.15E+01 cfm	1.00E+02	
	0	0.5	4.46E+05	-	-	
FW to Intact SGs	0.090	2	3.27E+05	7.77E+07 g/hr	1.00E+00	5
	2	8	9.81E+05	7.42E+07 g/hr	1.00E+00	
Intact SGs to Env	0	0.090	7.34E+05	2.18E+03 cfm	1.00E+02	6
	0.090	2	4.22E+05	5.90E+01 cfm	1.00E+02	
	2	8	9.34E+05	4.16E+01 cfm	1.00E+02	
	0	8	2.09E+06	-	-	

SGTR Flow Rates Notes:

1. RCS Leakage of 1 gpm – Volumetric leakage (gpm) from RCS is divided by 7.48 gal/ft^3.

2. RCS Flow Through Rupture to Environment – Mass release from RCS to the faulted SG is adjusted for reactor trip time:

0 to 0.090 hr: (21600 lbm)(324 sec)/(224 sec) = 3.12E4 lbm
0.090 to 0.5 hr: (136400 lbm)(1800-324 sec)/(1800-224 sec) = 1.28E5 lbm.

Flow is the release (lbm) divided by 62.4 lbm/ft^3 and by the time duration (min). The PFs correspond to flashing fractions.

3. RCS Flow Through Rupture to Faulted SG – Flows are the same as to the environment but the PFs correspond to the non-flashing fractions.
4. Faulted SG to Env – Time-dependent releases are converted into flows. The release from 0 to 0.090 hr is calculated as follows: (1133 lbm/sec)(324 sec) = 3.67E5 lbm. Flow is the release (lbm) divided by 62.4 lbm/ft^3 and by the time duration (min). PF is 100 for iodine and 1000 for alkali metals.
5. Feedwater to Intact SGs – Mass release (lbm) is multiplied by 453.6 g/lbm and divided by the time duration (hr).
6. Intact SGs to Env – Time-dependent releases are converted into flows. The release from 0 to 0.090 hr is as follows: (2)(1133 lbm/sec)(324 sec) = 7.34E5 lbm. Flow is the release (lbm) divided by 62.4 lbm/ft^3 and by the time duration (min). PF is 100 for iodine and 1000 for alkali metals.

<u>Radioactivity Considerations:</u>

No fuel failure occurs as a result of the SGTR.

Iodine Release Species:	97% elemental, 3% organic.

Initial RCS activity includes an assumption of 1% failed (leaking) fuel in accordance with past CLB (Affects alkali metals)

Initial RCS Activity:	
Pre-Accident Spike Case	30 μCi/gm
Concurrent Accident Spike	0.5 μCi/gm
Normal Iodine RCS Appearance Rate	
Letdown Flow Rate	145 gal/min
Identified Leakage	10 gal/min
Unidentified Leakage	1 gal/min
Concurrent Accident Spike Appearance	335 times the Normal Rate for the first 8 hours

<u>Initial Iodine Concentrations in RCS and Secondary Coolant</u>

DEI concentrations of 30, 0.5, and 0.1 μCi/g are converted to their dose-equivalent values of I-131 to I-135 as shown in the following table. The concentrations in the last three columns are used to calculate initial activities within nodes in Table 3.

Table 2 – Iodine Concentrations in RCS and Secondary Coolant

Isotope	1% FF RCS Concen (μCi/g)	Inh TEDE DCF (rem/Ci)	Concen x DCF	Concen (μCi/g) 30 μCi/g DEI	0.5 μCi/g DEI	0.1 μCi/g DEI
I-131	2.5	3.29E+04	8.23E+04	2.27E+01	3.78E-01	7.56E-02
I-132	0.9	3.81E+02	3.43E+02	8.17E+00	1.36E-01	2.72E-02
I-133	4.0	5.85E+03	2.34E+04	3.63E+01	6.05E-01	1.21E-01
I-134	0.6	1.31E+02	7.86E+01	5.44E+00	9.07E-02	1.81E-02
I-135	2.2	1.23E+03	2.71E+03	2.00E+01	3.33E-01	6.65E-02
Total			1.09E+05	9.26E+01	1.54E+00	3.09E-01
Note	2	3	4	5	6	7

Iodine Concentrations in RCS and Secondary Coolant Notes:

1. The note numbers correspond to column numbers.
2. Initial RCS activity considers 1% Failed (Leaking) Fuel.
3. DCF – Inhalation TEDE DCFs are from Federal Guidance Report 11, multiplied by 3.7E12 to convert from Sv/Bq to rem/Ci.
4. DCF Weighted Concentration – The concentrations in Column 2 are multiplied by the DCFs in Column 3. The total for this column represents the relative dose corresponding to 1% FF.
5. Concentrations at 30 μCi/g DEI – The relative dose corresponding to 30 μCi/g of I-131 is (30 μCi/g)(3.29E4 rem/Ci) = 9.87E5. In Column 5, the concentrations in Column 2 are multiplied by 9.87E5/1.09E5 to obtain the distribution corresponding to 30 μCi/g DEI.
6. Concentrations at 0.5 μCi/g DEI – The concentrations corresponding to 30 μCi/g in Column 5 are multiplied by 0.5/30.
7. Concentrations at 0.1 μCi/g DEI – The concentrations corresponding to 30 μCi/g in Column 5 are multiplied by 0.1/30.

Initial Iodine Activities in RCS and Secondary Coolant

The initial iodine activities in the RCS and the secondary coolant corresponding to 30, 0.5, and 0.1 μCi/g DEI are shown in the following table. These are entered in LocaDose as initial activities within nodes.

Table 3 – Initial Iodine Activities

Isotope	Initial Activity (Ci) RCS Pre Spike (30 μCi/g DEI)	RCS Con Spike (0.5 μCi/g DEI)	Intact SGs (0.1 μCi/g DEI)	Faulted SG (0.1 μCi/g DEI)
I-131	4.5E+03	7.6E+01	7.2E+01	1.5E+01
I-132	1.6E+03	2.7E+01	2.6E+01	5.5E+00
I-133	7.3E+03	1.2E+02	1.1E+02	2.4E+01
I-134	1.1E+03	1.8E+01	1.7E+01	3.7E+00
I-135	4.0E+03	6.7E+01	6.3E+01	1.3E+01
Total	1.9E+04	3.1E+02	2.9E+02	6.2E+01
Note	2	3	4	5

Initial Iodine Activities Notes:

1. The note numbers correspond to column numbers.
2. RCS Pre-Accident Spike – The concentrations (µCi/g) in Table 2, Column 5 are multiplied by the RCS mass of 2.00E8 g and by 1.0E-6 Ci/µCi.
3. RCS Concurrent Spike – The concentrations (µCi/g) in Table 2, Column 6 are multiplied by the RCS mass of 2.00E8 g and by 1.0E-6 Ci/µCi.
4. Intact SGs – The mass of fluid released from the intact SGs is 2.09E6 lbm [Table 1], which is multiplied by 453.6 g/lbm to yield 9.48E8 g. The concentrations (µCi/g) in Table 2, Column 7 are multiplied by this mass and by 1.0E-6 Ci/µCi.
5. Faulted SG – The mass of fluid released from the faulted SG is 4.46E5 lbm [Table 1], which is multiplied by 453.6 g/lbm to yield 2.02E8 g. The concentrations (µCi/g) in Table 2, **Error! Reference source not found.**Column 7 are multiplied by this mass and by 1.0E-6 Ci/µCi.

<u>Noble Gases and Alkali Metals in RCS</u>

The initial RCS activities of noble gases and alkali metals corresponding to 1% failed (leaking) fuel are shown in the following table. The values in the last column are entered in LocaDose as initial activities.

Table 4 - Noble Gases and Alkali Metals in RCS

Isotope	Concen (µCi/g)	Activity (Ci)
Kr-85m	2.2E+00	4.4E+02
Kr-85	5.5E+00	1.1E+03
Kr-87	1.3E+00	2.6E+02
Kr-88	3.8E+00	7.6E+02
Xe-133m	3.2E+00	6.4E+02
Xe-133	2.9E+02	5.8E+04
Xe-135m	2.0E-01	4.0E+01
Xe-135	6.1E+00	1.2E+03
Xe-138	7.0E-01	1.4E+02
Rb-88	3.8E+00	7.6E+02
Rb-89	1.0E-01	2.0E+01
Cs-134	2.6E-01	5.2E+01
Cs-136	1.5E-01	3.0E+01
Cs-137	1.3E+00	2.6E+02
Cs-138	9.6E-01	1.9E+02
Total	3.2E+02	6.4E+04
Note	2	3

Noble Gases and Alkali Metals in RCS Notes:

1. The note numbers correspond to the column numbers.
2. Initial RCS Concentrations – These correspond to a 1% Failed (Leaking) Fuel Assumption.

3. RCS Activities – The concentrations (µCi/g) in Column 2 are multiplied by the RCS mass of 2.00E8 g and by 1.0E-6 Ci/µCi.

Alkali Metals in Secondary Coolant

The initial activities of alkali metals in the secondary coolant are shown in the following table, corresponding to 20% of the RCS values at 1% failed (leaking) fuel. These are entered in LocaDose as initial activities within nodes.

Table 5 - Alkali Metals in the Secondary System:

Isotope	Concen (µCi/g)	Activity (Ci)	
		Intact SGs	Faulted SG
Rb-88	7.6E-01	7.2E+02	1.5E+02
Rb-89	2.0E-02	1.9E+01	4.0E+00
Cs-134	5.2E-02	4.9E+01	1.1E+01
Cs-136	3.0E-02	2.8E+01	6.1E+00
Cs-137	2.6E-01	2.5E+02	5.3E+01
Cs-138	1.9E-01	1.8E+02	3.9E+01
Total	1.3E+00	1.2E+03	2.7E+02
Note	2	3	4

Initial Alkali Metals in the Secondary System Notes:

1. The note numbers correspond to the columns of the table.
2. Concentrations – the initial RCS concentrations shown in Table 4 are multiplied by 0.2.
3. Intact SGs Activities – the mass of the fluid released from the intact SGs is 2.09E6 lbm [Table 1], which is multiplied by 453.6 gm/lbm to yield 9.48E8 grams. The concentrations (in µCi/gm) in Column 2 of this table are multiplied by this mass and then by 1.0E-6 Ci/µCi.
4. Faulted SG Activities – the mass of the fluid released from the faulted SG is 4.46E5 lbm [Table 1], which is multiplied 453.6 gm/lbm to yield 2.02E8 grams. The concentrations (in µCi/gm) in Column 2 of this table are multiplied by this mass and then by 1.0E-6 Ci/µCi.

Iodine Appearance Rates in RCS

In the following table, the total removal rate for each isotope is determined based on the three removal terms. For equilibrium conditions, the production rate is equal to the removal rate. The iodine concentrations in the RCS are multiplied by the removal rates, thereby yielding the production rates. The values in the last column are entered in LocaDose as production terms for the concurrent spike case.

Table 6 - RCS Appearance Rates

Isotope	Decay Rate (sec^{-1})	Total Removal Rate (hr^{-1})	Appearance Rate (Ci/hr)	
			Normal	Con Spike
I-131	9.98E-07	1.27E-01	9.6E+00	3.2E+03
I-132	8.43E-05	4.27E-01	1.2E+01	3.9E+03
I-133	9.21E-06	1.56E-01	1.9E+01	6.3E+03
I-134	2.20E-04	9.15E-01	1.7E+01	5.6E+03
I-135	2.91E-05	2.28E-01	1.5E+01	5.1E+03
Note	2	3	4	5

RCS Iodine Appearance Rates Notes:

1. The note numbers correspond to the column of the table.
2. Decay rates are from the LocaDose manual
3. Total Removal Rate – the decay rate is multiplies by 3600 sec/hr and added to the clean-up rate of 1.14 hr-1 and the leakage rate of 9.42E-3 hr-1.
4. Normal Appearance Rate – the initial iodine activity (ci) from Table 3 (column 3) is multiplied by the total removal rate (hr-1) in column 3 of this table.
5. Concurrent Spike Appearance Rate – the normal appearance rate in column 4 of this table is multiplied by 335.

Iodine Appearance Rates in the Intact SGs From Feedwater

The Feedwater system flows are modelled as a source of radioiodine for this analysis. The Technical Specification limit is 0.1 µCi/g. The secondary system iodine concentrations are shown in Table 2, Column 7. These concentrations are multiplied by 1.0E-6 Ci/µCi and by time-dependent flow rates (g/hr) shown in Table 1 to obtain the following:

Table 7 - Iodine Appearance Rates in the Intact SGs

Isotope	Feedwater Appearance Rate (Ci/hr)	
	0.09-2 hr	2-8 hr
I-131	5.9E+00	5.6E+00
I-132	2.1E+00	2.0E+00
I-133	9.4E+00	9.0E+00
I-134	1.4E+00	1.3E+00
I-135	5.2E+00	4.9E+00

Control Room Ventilation Parameters:

Parameter	Value
Pressurization Mode starts	Initiated at start of accident.
CR Make-up Flow Rate	375 cfm (throughout accident)
Pressurization Unfiltered In-leakage	300 cfm (throughout accident)
CR Ingress/egress	10 cfm (throughout accident through CR Vent)

CR Volume 114,000 cubic feet
CR Pressurization Filters
HEPA 98.5% for all particulates
Charcoal 98.5% for all iodine species
CR Recirculation Flow 2700 cfm (throughout accident)
Iodine Filter Efficiency 94.5% for organic and elemental
98.5% for particulates
CR Breathing Rates 3.5E-04 m^3/sec for 8 hours
EAB & LPZ Breathing Rates 3.5E-04 m^3/sec for 8 hours

Atmospheric Dispersion Factors:

Table 8 – Atmospheric Dispersion Factors

Time (hr)	χ/Q (sec/m^3)		
	EAB	LPZ	Control Room
0 – 2	7.60E-4	2.80E-4	1.66E-3
2 – 8	-	1.10E-4	1.38E-3

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Enclosure 10

Control Rod Ejection Accident Analysis

CONTROL ROD EJECTION ACCIDENT DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number:	SM-1080538201-004, Version 1
Method/Computer Program Used:	LocaDose Version 7.1
Regulatory Guidance:	RG-1.183, 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. 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 cold shutdown conditions are established in the RCS.

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

- Containment Leakage – Activity from fuel melting and fuel cladding damage 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 and fuel cladding damage instantaneously reaches 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)	
	Calculated	Limit
EAB	3.8	6.3
LPZ	2.7	6.3
Control Room	3.7	5

The maximum 2-hour EAB dose occurs between 0 and 2 hours.
(Note that rounding is applied to all values)

Key Assumptions and Inputs

Physical Parameters

Parameter	Value
Reactor Power Level	2775 MWt (+2% uncertainty = 2831 MWt)
Containment Volume	2.03E6 ft^3
Containment Leakage	0.15% per day for first 24 hours 0.075% per day after 24 hours
Particulate Removal	2.74E-2 per hour, credit is taken for natural deposition in Containment per NUREG/CR-6189 (Table 36)
RCS Mass	440,900 lbm (2.00E8 grams)
RCS Volume	1.02E4 cubic feet
SG Mass	1.68E5 lbm (per SG, which is assumed to be full)
SG Volume	2693 cubic feet (each)
Coolant Densities	Primary and Secondary water at 62.4 lbm/ft^3
Partition Factors	Iodine PF = 100, Alkali Metals PF = 1000 (moisture carryover = 0.1%) Noble Gases PF = 1
Primary to Secondary Leakage	1 gpm total, for the first 2500 seconds of the accident.
Secondary System Mass releases	468,600 lbm (426,000 +10% margin) in first 98 seconds.

Table 1 - Flow Rates

Flow Path	Time (hour)		Flow	Note
	From	to		
RCS to Env	0	0.694	1.34E-01 cfm	1
Containment to the Environment	0	24	2.11E+00 cfm	2
	24	720	1.06E+00 cfm	
SGs to Env	0	0.027	4.60E+03 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. Containment – Volume of 2.03E6 ft^3 is multiplied by 0.0015/day and divided by 1440 min/day for the first 24 hours. After 24 hours, the flow is halved.
3. SGs – Mass release from the intact SGs of 4.26E5 lbm is multiplied by 1.1. 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 50% of the Iodine isotopes within the melting rods are available for release form the containment and RCS for the containment and secondary system pathways.
- 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 2831 MWt. To account for potential cycle-to-cycle variations, the following margin factors are applied to the core inventory:
  - Kr-85 1.15
  - Xe-133 1.05
  - Cs-134 1.35
  - Cs-136 1.25
  - Cs-137 1.20
  - Iodine isotopes and other Noble Gases 1.02
  - Other Isotopes 1.03
- 100% of the activity released to from the core due to fuel melting and cladding failure is instantaneously released to and uniformly mixed in the containment at the onset of the accident.
- 100% of the activity released from the core due to fuel melting and cladding failure is instantaneously mixed within the RCS at the onset of the accident. Compared to the gap release, any RCS iodine activity due to spiking is negligible.
- Chemical form of iodine released to containment is 95% particulate, 4.85 elemental, and 0.15% organic. The containment distribution is used for the secondary system release pathway, because 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.
- RCS activity includes an assumption of normal operations 1% failed (leaking) fuel in accordance with current licensing basis (affects alkali metals).
- The radioiodine concentration in the secondary system is assumed to be at the Technical Specification limit of 0.1 μCi/gm DEI
- The concentrations of Alkali Metals in Secondary are based upon a ration of the concentration in the RCS: Given 0.1 μCi/gm DEI in the secondary and 0.5 μCi/gm in the RCS, the concentrations of alkali metals in the secondary are assumed to be 20% of those in the RCS.

Containment and RCS Activities

Table 2 Reports the Containment and RCS Activities.

The Activity in the RCS or Containment is the sum of the activity from Fuel melting plus the activity released from the gap of the damaged fuel.

Table 2 - Containment and RCS Activities

Isotope	Core Activity (curies)	Margin Factor	Core release	Gap fraction	Activities (curies)		
					Fuel Melt	Gap release	Total
I-131	7.50E+07	1.02	0.50	0.10	9.6E+04	1.3E+06	1.4E+06
I-132	1.10E+08	1.02	0.50	0.10	1.4E+05	1.9E+06	2.0E+06
I-133	1.60E+08	1.02	0.50	0.10	2.0E+05	2.8E+06	3.0E+06
I-134	1.70E+08	1.02	0.50	0.10	2.2E+05	2.9E+06	3.2E+06
I-135	1.50E+08	1.02	0.50	0.10	1.9E+05	2.6E+06	2.8E+06
Kr-83m	9.70E+06	1.02	1.00	0.10	2.5E+04	1.7E+05	1.9E+05
Kr-85m	2.10E+07	1.02	1.00	0.10	5.4E+04	3.6E+05	4.2E+05
Kr-85	7.20E+05	1.15	1.00	0.10	2.1E+03	1.4E+04	1.6E+04
Kr-87	4.00E+07	1.02	1.00	0.10	1.0E+05	6.9E+05	8.0E+05
Kr-88	5.70E+07	1.02	1.00	0.10	1.5E+05	9.9E+05	1.1E+06
Kr-89	6.90E+07	1.02	1.00	0.10	1.8E+05	1.2E+06	1.4E+06
Xe-131m	8.40E+05	1.02	1.00	0.10	2.1E+03	1.5E+04	1.7E+04
Xe-133m	4.80E+06	1.02	1.00	0.10	1.2E+04	8.3E+04	9.5E+04
Xe-133	1.50E+08	1.05	1.00	0.10	3.9E+05	2.7E+06	3.1E+06
Xe-135m	3.00E+07	1.02	1.00	0.10	7.7E+04	5.2E+05	6.0E+05
Xe-135	3.50E+07	1.02	1.00	0.10	8.9E+04	6.1E+05	7.0E+05
Xe-137	1.40E+08	1.02	1.00	0.10	3.6E+05	2.4E+06	2.8E+06
Xe-138	1.30E+08	1.02	1.00	0.10	3.3E+05	2.3E+06	2.6E+06
Br-82	3.80E+05	1.03	1.00	0.05	9.8E+02	3.3E+03	4.3E+03

Isotope	Core Activity (curies)	Margin Factor	Core release	Gap fraction	Activities (curies) Fuel Melt	Gap release	Total
Br-83	9.70E+06	1.03	1.00	0.05	2.5E+04	8.5E+04	1.1E+05
Br-84	1.70E+07	1.03	1.00	0.05	4.4E+04	1.5E+05	1.9E+05
Br-85	2.10E+07	1.03	1.00	0.05	5.4E+04	1.8E+05	2.4E+05
Br-86	1.50E+07	1.03	1.00	0.05	3.9E+04	1.3E+05	1.7E+05
Br-87	3.40E+07	1.03	1.00	0.05	8.8E+04	3.0E+05	3.9E+05
Br-88	3.60E+07	1.03	1.00	0.05	9.3E+04	3.2E+05	4.1E+05
Rb-86	1.40E+05	1.03	1.00	0.05	3.6E+02	1.2E+03	1.6E+03
Rb-88	5.70E+07	1.03	1.00	0.05	1.5E+05	5.0E+05	6.5E+05
Rb-89	7.40E+07	1.03	1.00	0.05	1.9E+05	6.5E+05	8.4E+05
Rb-90	7.20E+07	1.03	1.00	0.05	1.9E+05	6.3E+05	8.2E+05
Rb-91	8.90E+07	1.03	1.00	0.05	2.3E+05	7.8E+05	1.0E+06
Cs-134m	3.60E+06	1.03	1.00	0.05	9.3E+03	3.2E+04	4.1E+04
Cs-134	1.10E+07	1.35	1.00	0.05	3.7E+04	1.3E+05	1.6E+05
Cs-136	3.30E+06	1.25	1.00	0.05	1.0E+04	3.5E+04	4.5E+04
Cs-137	7.60E+06	1.20	1.00	0.05	2.3E+04	7.8E+04	1.0E+05
Cs-138	1.40E+08	1.03	1.00	0.05	3.6E+05	1.2E+06	1.6E+06
Cs-139	1.40E+08	1.03	1.00	0.05	3.6E+05	1.2E+06	1.6E+06
Cs-140	1.20E+08	1.03	1.00	0.05	3.1E+05	1.1E+06	1.4E+06
Cs-141	9.10E+07	1.03	1.00	0.05	2.3E+05	8.0E+05	1.0E+06
Note	2	3	4	5	6	7	8

Containment and RCS Activities notes:

1. The note numbers correspond to column numbers.
2. Core Activity – At shutdown.
3. Margin Factor – Accounts for cycle variations.

4. Core Release – Applies to melted fuel rods.
5. Gap Fraction – Applies to fuel rods with cladding failure.
6. Fuel Melt Activity – Product of Activity (Ci) in Column 2, margin in Column 3, core release in Column 4 and fuel melt fraction of 0.0025.
7. Gap Activity – Product of Activity (Ci) in Column 2, margin in Column 3, gap fraction in Column 5, fuel cladding failure fraction of 0, and RPF of 1.7.
8. Total Activity – This is the sum of Columns 6 and 7.

Initial Iodine Activities in the Secondary Coolant

The initial iodine activities in the secondary coolant corresponding to 0.1 µCi/g DEI are shown in the following table. These are entered in LocaDose as initial activities within nodes.

Table 3 - Initial Iodine Activities in the RCS and Secondary Coolant

Isotope	RCS Concentration (1% Leaking Fuel) (µCi/g)	Inhalation TEDE DCF (Rem/Ci)	Concentration x DCF	0.1 µCi/g DEI Concentration (µCi/g)	Activity in Secondary (Ci)
I-131	2.5	3.29E+04	8.23E+04	7.56E-02	1.7E+01
I-132	0.9	3.81E+02	3.43E+02	2.72E-02	6.2E+00
I-133	4.0	5.85E+03	2.34E+04	1.21E-01	2.8E+01
I-134	0.6	1.31E+02	7.86E+01	1.81E-02	4.1E+00
I-135	2.2	1.23E+03	2.71E+03	6.65E-02	1.5E+01
Total			1.09E+05	3.09E-01	7.1E+01
Note	2	3	4	5	6

Initial Iodine Activities Notes:

1. The note numbers correspond to column numbers.
2. RCS at 1% Leaking Fuel – These are the RCS concentrations (µCi/g) corresponding to 1% failed (leaking) fuel.
3. DCF – Inhalation TEDE DCFs are from FGR 11, multiplied by 3.17E12 to convert from Sv/Bq to Rem/Ci
4. DCR Weighted Concentration – the concentrations in Column 2 are multiplied by the DCFs in Column 3. The total for this column represents the relative dose corresponding to 1% failed (leaking) fuel.
5. Concentrations at 0.1 µCi/g DEI – the relative dose corresponding to 0.1 µCi/g DEI is (0.1 µCi/g)(3.29E4 rem/ci) = 3.29E3. In Column 5, the concentrations in Column 2 are multiplied by 3.29E3/1.09E5 to obtain the distribution corresponding to 0.1 µCi/g DEI.
6. Activity in Secondary – the concentrations (µCi/g) in Column 5 are multiplied by the mass of 2.29E8 grams and by 1.0E-06 Ci/µCi.

Alkali Metals in the Secondary System

The initial concentrations are assumed to be 20% of the RCS initial activities.

Table 4 - Alkali Metals in RCS (1% failed (leaking) fuel) and Secondary

Isotope	Concentration (μCi/g)		Activity in Secondary (Curies)
	RCS 1% Leak Fuel	Secondary	
Rb-88	3.8E+00	7.6E-01	1.7E+02
Rb-89	1.0E-01	2.0E-02	4.6E+00
Cs-134	2.6E-01	5.2E-02	1.2E+01
Cs-136	1.5E-01	3.0E-02	6.9E+00
Cs-137	1.3E+00	2.6E-01	5.9E+01
Cs-138	9.6E-01	1.9E-01	4.4E+01
Total	6.6E+00	1.3E+00	3.0E+02
Note	2	3	4

Initial Alkali Metals in the Secondary System Notes:

1. The note numbers correspond to the columns of the table.
2. RCS Concentrations – the initial RCS concentrations are those corresponding to 1% failed (leaking) fuel under normal operations.
3. Secondary concentrations – the concentrations in the secondary are the RCS concentrations (corresponding to 0.5 μCi/g DEI) multiplied by 0.2 to achieve concentrations corresponding to 0.1 μCi/g DEI.
4. Secondary Activities – the concentrations in column 3 are multiplied by the mass of 2.29E8 grams and by 1.0E-06 Ci/μCi

Radioiodine Appearance Rates

Iodine Appearance Rates in Intact SGs from Feedwater

A mass flow rate from the Feedwater to the steam generators is generated to develop an appearance rate of iodine into the SG (for steaming to the environment). The mass released (426E5 lbm) is adjusted to add a 10% margin, converted to grams, and divided by the release time to create a mass flow rate:

[(4.26E5 lbm)(1.1)(453.6 g/lbm)]/(98 sec)(3600 sec/hr)] = 7.81E9 grams/hour

The flow of feedwater into the intact SGs is modeled as an activity production term in LocaDose. The concentrations corresponding to 0.1 μCi/g from Table 4, Column 3 are multiplied by 1.0E-6 Ci/μCi and by time-dependent flow rates (g/hr) from Table 2, yielding the following appearance rates. These are entered in LocaDose as production terms.

Table 5 - Iodine Appearance Rates in Intact SGs

Isotope	Feedwater Appearance Rate (Ci/hr)
I-131	5.9E+02
I-132	2.1E+02
I-133	9.4E+02
I-134	1.4E+02
I-135	5.2E+02

Control Room Ventilation Parameters

Parameter	Value
Pressurization Mode starts	Initiated at start of accident.
CR Make-up Flow Rate	375 cfm (throughout accident)
Pressurization Unfiltered In-leakage	300 cfm (throughout accident)
CR Ingress/egress	10 cfm (throughout accident through CR Vent)
CR Volume	114,000 cubic feet
CR Pressurization Filters	98.5% for all radionuclide groups except noble gases
CR Recirculation Flow	2700 cfm (throughout accident)
Iodine Filter Efficiency	98.5% for particulates 94.5% for all other radionuclide groups except noble gases
CR Breathing Rates	3.5E-04 m^3/sec for 8 hours
EAB & LPZ Breathing Rates	3.5E-04 m^3/sec for 8 hours

Atmospheric Dispersion Factors:

Table 6 – Atmospheric Dispersion Factors

Time (hr)	X/Q (sec/m^3) EAB	LPZ	Control Room
0 – 2	7.60E-4	2.80E-4	1.66E-3
2 – 8	-	1.10E-4	1.38E-3
8 – 24	-	1.0E-05	7.20E-04
24 – 96	-	5.4E-06	5.6E-04
96 – 720	-	2.9E-06	4.21E-04

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Enclosure 11

Locked Rotor Accident Analysis

LOCKED ROTOR ACCIDENT DOSE CONSEQUENCES USING AST METHODS

Licensee Document Number: SM-1080538201-006, Version 1

Method/Computer Program Used: LocaDose Version 7.1

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 cold shutdown conditions are established in the RCS.

Results and Acceptance Limits:

Location	Dose (Rem TEDE)	
	Calculated	Limit
EAB	1.2	2.5
LPZ	0.83	2.5
Control Room	< 5	5

Note that the control room dose for this accident was not reported in the FSAR per the current licensing basis. Control room doses for this accident using the AST methods are analyzed as being less than the 5 Rem TEDE limit. However, a reassessment is being performed assuming a delayed manual CREFS initiation. The results of this reassessment are expected to remain less than 5 rem TEDE and be non-limiting.

The maximum 2-hour EAB dose occurs between 6 and 8 hours.
(Note that rounding is applied to all values)

Key Assumptions and Inputs

Physical Parameters

Parameter	Value
Reactor Power Level:	2775 MWt (+2% uncertainty = 2831 MWt)
RCS Mass:	440,900 lbm (2.00E8 grams)
RCS Volume:	1.02E4 cubic feet
SG Mass:	1.68E5 lbm (per SG, which is assumed to be full)
SG Volume:	2690 cubic feet (each)
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 = 1000 (moisture carryover = 0.1%)
Noble Gases PF = 1

Primary to Secondary Leakage: 1 gpm total.

Table 1: Flow Rates Before 10% Margin Adjustment

Pathway	Time		Release (lbm)	Flow	Note
	From	To			
RCS to SG	0	8	-	1.34E-01 CFM	1
Feedwater to SG	0	2	7.63E+05	1.73E+08 g/hr	2
	2	8	9.29E+05	7.03E+07 g/hr	
SG to Environment	0	2	5.64E+05	7.53E+01 cfm	3
	2	8	9.17E+05	4.08E+01 cfm	

Flow Rate Notes:

1. RCS – Volumetric leakage (gallons/minute) from the RCS is divided by 7.48 gal/ft3.
2. Feedwater – The Feedwater flow to the SGs is 693,629 lbm in the first two hours and 844,963 lbm from 2 to 8 hours. Mass release from the feedwater to the SG is then increased by 10% for margin. Flow is the release (lbm) multiplied by 453.6 grams/lbm and divided by the time duration (hour).
3. SG – Mass release from the SG is 512,325 lbm in the first two hours and 833221 lbm from 2 to 8 hours. The mass release from the SG is increased by 10% for margin. The flow is then the release (lbm) divided by 62.4 lbm/ft3 and divided by the time duration (min).

Radioactivity Considerations:

- 20% 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 2831 MWt. To account for potential cycle-to-cycle variations, the following margin factors are applied to the core inventory:
    - Kr-85 1.15
    - Xe-133 1.05
    - Cs-134 1.35
    - Cs-136 1.25
    - Cs-137 1.20

- o Iodine isotopes and other noble gases 1.02
- o Other Isotopes 1.03

- 100% of the activity released from the core due to cladding failure is instantaneously mixed within the RCS at the onset of the accident. Compared to the gap release, any RCS iodine activity due to spiking is negligible.
- Chemical form of iodine released to the environment 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.
- RCS activity includes an assumption of normal operations 1% failed (leaking) fuel in accordance with past CLB (Affects alkali metals)
- 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/gm DEI in the secondary and 0.5 µCi/gm in the RCS, the concentrations of alkali metals in the secondary are assumed to be 20% of those in the RCS.

Containment and RCS Activities

Table 2 - Containment and RCS Activities

Isotope	Core Activity (curies)	Margin Factor	Gap fraction	Gap release	Isotope	Core Activity (curies)	Margin Factor	Gap fraction	Gap release
I-131	7.50E+07	1.02	0.08	2.1E+06	Br-83	9.70E+06	1.03	0.12	4.1E+05
I-132	1.10E+08	1.02	0.05	1.9E+06	Br-84	1.70E+07	1.03	0.12	7.1E+05
I-133	1.60E+08	1.02	0.05	2.8E+06	Br-85	2.10E+07	1.03	0.12	8.8E+05
I-134	1.70E+08	1.02	0.05	2.9E+06	Br-86	1.50E+07	1.03	0.12	6.3E+05
I-135	1.50E+08	1.02	0.05	2.6E+06	Br-87	3.40E+07	1.03	0.12	1.4E+06
Kr-83m	9.70E+06	1.02	0.05	1.7E+05	Br-88	3.60E+07	1.03	0.12	1.5E+06
Kr-85m	2.10E+07	1.02	0.05	3.6E+05	Rb-86	1.40E+05	1.03	0.12	5.9E+03
Kr-85	7.20E+05	1.15	0.10	2.8E+04	Rb-88	5.70E+07	1.03	0.12	2.4E+06
Kr-87	4.00E+07	1.02	0.05	6.9E+05	Rb-89	7.40E+07	1.03	0.12	3.1E+06
Kr-88	5.70E+07	1.02	0.05	9.9E+05	Rb-90	7.20E+07	1.03	0.12	3.0E+06
Kr-89	6.90E+07	1.02	0.05	1.2E+06	Rb-91	8.90E+07	1.03	0.12	3.7E+06

Isotope	Core Activity (curies)	Margin Factor	Gap fraction	Gap release	Isotope	Core Activity (curies)	Margin Factor	Gap fraction	Gap release
Xe-131m	8.40E+05	1.02	0.05	1.5E+04	Cs-134m	3.60E+06	1.03	0.12	1.5E+05
Xe-133m	4.80E+06	1.02	0.05	8.3E+04	Cs-134	1.10E+07	1.35	0.12	6.1E+05
Xe-133	1.50E+08	1.05	0.05	2.7E+06	Cs-136	3.30E+06	1.25	0.12	1.7E+05
Xe-135m	3.00E+07	1.02	0.05	5.2E+05	Cs-137	7.60E+06	1.20	0.12	3.7E+05
Xe-135	3.50E+07	1.02	0.05	6.1E+05	Cs-138	1.40E+08	1.03	0.12	5.9E+06
Xe-137	1.40E+08	1.02	0.05	2.4E+06	Cs-139	1.40E+08	1.03	0.12	5.9E+06
Xe-138	1.30E+08	1.02	0.05	2.3E+06	Cs-140	1.20E+08	1.03	0.12	5.0E+06
Br-82	3.80E+05	1.03	0.12	1.6E+04	Cs-141	9.10E+07	1.03	0.12	3.8E+06
Note:	2	3	4	5	Note:	2	3	4	5

Containment and RCS Activities Notes:

1. The note numbers the column numbers and the columns are repeated.
2. Core Activity – Activity in the core for the isotope at shutdown.
3. Margin Factor – per isotope accounting for cycle variations
4. Gap Release Fraction per RG-1.183
5. The released activity is the product of the core activity multiplied by the margin factor multiplied by the release fraction multiplied by the amount of damaged fuel (20%) multiplied by the radial power peaking factor (1.7).

<u>Initial Iodine Activities in the Secondary Coolant</u>

The initial iodine activities in the secondary coolant corresponding to 0.1 µCi/g DEI are shown in the following table. These are entered in LocaDose as initial activities within nodes.

Table 3 - Initial Iodine Activities in the RCS and Secondary Coolant

Isotope	RCS Concentration (1% Leaking Fuel) (µCi/g)	Inhalation TEDE DCF (Rem/Ci)	Concentration x DCF	0.1 µCi/g DEI Concentration (µCi/g)	Activity in Secondary (Ci)
I-131	2.5	3.29E+04	8.23E+04	7.56E-02	1.7E+01
I-132	0.9	3.81E+02	3.43E+02	2.72E-02	6.2E+00
I-133	4.0	5.85E+03	2.34E+04	1.21E-01	2.8E+01

Isotope	RCS Concentration (1% Leaking Fuel) (µCi/g)	Inhalation TEDE DCF (Rem/Ci)	Concentration x DCF	0.1 µCi/g DEI Concentration (µCi/g)	Activity in Secondary (Ci)
I-134	0.6	1.31E+02	7.86E+01	1.81E-02	4.1E+00
I-135	2.2	1.23E+03	2.71E+03	6.65E-02	1.5E+01
Total			1.09E+05	3.09E-01	7.1E+01
Note:	2	3	4	5	6

Initial Iodine Activities Notes:

1. The note numbers correspond to column numbers.
2. RCS at 1% Leaking Fuel – These are the RCS concentrations (µCi/g) corresponding to 1% failed (leaking) fuel during normal operations.
3. DCF – Inhalation TEDE DCFs are from FGR 11, multiplied by 3.17E12 to convert from Sv/Bq to Rem/Ci
4. DCR Weighted Concentration – the concentrations in Column 2 are multiplied by the DCFs in Column 3. The total for this column represents the relative dose corresponding to 1% failed (leaking) fuel.
5. Concentrations at 0.1 µCi/g DEI – the relative dose corresponding to 0.1 µCi/g DEI is (0.1 µCi/g)(3.29E4 rem/ci) = 3.29E3. In Column 5, the concentrations in Column 2 are multiplied by 3.29E3/1.09E5 to obtain the distribution corresponding to 0.1 µCi/g DEI.
6. Activity in Secondary – the concentrations (µCi/g) in Column 5 are multiplied by the mass of 2.29E8 grams and by 1.0E-06 Ci/µCi.

<u>Alkali Metals in the Secondary System</u>

The initial concentrations are assumed to be 20% of the RCS initial activities.

Table 4 - Alkali Metals in RCS (1% failed (leaking) fuel) and Secondary

Isotope	Concentration (µCi/g)		Activity in Secondary (Curies)
	RCS 1% Leak Fuel	Secondary	
Rb-88	3.8E+00	7.6E-01	1.7E+02
Rb-89	1.0E-01	2.0E-02	4.6E+00
Cs-134	2.6E-01	5.2E-02	1.2E+01
Cs-136	1.5E-01	3.0E-02	6.9E+00
Cs-137	1.3E+00	2.6E-01	5.9E+01
Cs-138	9.6E-01	1.9E-01	4.4E+01
Total	6.6E+00	1.3E+00	3.0E+02
Note:	2	3	4

Initial Alkali Metals in the Secondary System Notes:

1. The note numbers correspond to the columns of the table.
2. RCS Concentrations – the initial RCS concentrations are those corresponding to 1% failed (leaking) fuel under normal operations.

3. Secondary concentrations – the concentrations in the secondary are the RCS concentrations (corresponding to 0.5 µCi/g DEI) multiplied by 0.2 to achieve concentrations corresponding to 0.1 µCi/g DEI.
4. Secondary Activities – the concentrations in column 3 are multiplied by the mass of 2.29E8 grams and by 1.0E-06 Ci/µCi

Radioiodine Appearance Rates

Iodine Appearance Rates in Intact SGs from Feedwater

The flow of feedwater into the intact SGs is modeled as an activity production term in LocaDose. The concentrations corresponding to 0.1 µCi/g from Table 3, Column 5 are multiplied by 1.0E-6 Ci/µCi and by time-dependent flow rates (g/hr) from Table 1, yielding the following appearance rates. These are entered in LocaDose as production terms.

Table 5 - Iodine Appearance Rates in Intact SGs

Isotope	Feedwater Appearance Rate (Ci/hr)	
	0 – 2 Hour	0 – 8 hour
I-131	1.3E+01	5.3E+00
I-132	4.7E+00	1.9E+00
I-133	2.1E+01	8.5E+00
I-134	3.1E+00	1.3E+00
I-135	1.2E+01	4.7E+00

Control Room Ventilation Parameters

Parameter	Value
Isolation Mode	Not modeled
Pressurization Mode starts	Initiated at start of accident (under reassessment for delayed manual start)
Normal CR Make-up Flow Rate	2340 CFM (+10 CFM ingress/egress) unfiltered until pressurization mode is started
CR Pressurization Flow Rate	375 cfm
Pressurization Unfiltered In-leakage	300 cfm
CR Ingress/egress	10 cfm (continuous unfiltered through CR Vent)
CR Volume	114,000 cubic feet
CR Pressurization Filters	98.5% for all radioactive groups except noble gases
CR Recirculation Flow	2700 cfm (manual start at 20 minutes until end of accident)
Iodine Filter Efficiency	98.5% for particulates 94.5% for all other radionuclide groups
CR Breathing Rates	3.5E-04 m^3/sec for 8 hours
EAB & LPZ Breathing Rates	3.5E-04 m^3/sec for 8 hours

Atmospheric Dispersion Factors

Table 6 – Atmospheric Dispersion Factors

Time (hr)	X/Q (sec/m^3)		
	EAB	LPZ	Control Room
0 – 2	7.60E-4	2.80E-4	1.66E-3
2 – 8	-	1.10E-4	1.38E-3

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Enclosure 12

FNP AST Accident Analysis Input Values Comparison Tables

FNP 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 Farley Nuclear Plant (FNP), 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	114,000 ft^3	114,000 ft^3	No change
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	0 cfm	2340 cfm	60 seconds of normal Control Room HVAC operation is assumed after accident initiation.
Unfiltered In-leakage	0 cfm	0 cfm	No change
Emergency Operation			
Recirculation Mode			
Filtered Make-up Flow Rate	375 cfm	375 cfm	No change
Filtered Recirculation Flow Rate	2700 cfm	2700 cfm	No change
Unfiltered Make-up Flow Rate	0 cfm	0 cfm	No change
Unfiltered In-leakage	53 cfm	325 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 iodine 98.5%	All iodine 98.5%	No change
Recirculation Filters	Elemental – 94.5% Organic - 94.5% Particulate – 98.5%	Elemental – 94.5% Organic - 94.5% Particulate – 98.5%	No change
Particulate	98.5%	98.5%	No change
Occupancy			
0-24 hours 1-4 days 4-30 days	100% 60% 40%	100% 60% 40%	No change
Breathing Rate	3.47E-4 m^3/sec	3.5E-4 m^3/sec	Rounded up for conservatism.

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	2.5% particulate, 95.5% elemental, 2.0% organic	95% cesium iodide, 4.85% elemental, 0.15% organic	Adoption of RG 1.183 methodology.
Containment Volume	2,030,000 ft^3	2,030,000 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	2.5% particulate, 95.5% elemental, 2.0% 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	1,668,660 ft^3	1,668,660 ft^3	No change
Containment unsprayed Volume	361,340 ft^3	361,340 ft^3	No change
Containment Spray Start Time	0 seconds	90 seconds	Provides additional conservatism to Containment Leakage Pathway.
Containment Spray Stop Time	8 hours	8 hours	No change
Containment Spray Flow Rate	2480 gpm injection mode	2480 gpm injection mode 2290 gpm recirculation mode	No change
Elemental Iodine Spray Removal Coefficient	2.7 hr^{-1}	13.7 hr^{-1}	Revision is consistent with RG 1.183 Appendix A RP 3.3
Aerosol Spray Removal Coefficient	5.45 hr^{-1}injection mode 5.03 hr^{-1}recirculation mode	5.45 hr^{-1}injection mode 5.03 hr^{-1}recirculation mode	No change
Organic Iodine Spray Removal	None	None	No change
Natural Deposition	Elemental, Organic, Aerosol – None	Elemental, Organic iodine – None Aerosols – 0.1 hr^{-1}in unsprayed regions only	Aerosol natural deposition is permitted per Appendix A of RG 1.183.
Containment Leakage Rate 0 to 24 hours 24 hours to 30 days	 0.15%/day 0.075%/day	 0.15%/day 0.075%/day	No change
Containment Leakage Filtration	0%	0%	No change
ECCS Leakage to the Auxiliary Building			
Iodine Chemical Form	0% aerosol, 98% elemental, 2% organic	0% aerosol, 97% elemental, 3% organic	The revised percentages are as specified in RG 1.183.
Containment Sump Volume	49,200 ft^3	49,200 ft^3	No change
ECCS Recirculation Start Time	20 minutes 20 minutes	20 minutes	No change
ECCS Leakage Flow Rate	4,000 cc/hr	20,000 cc/hr	ECCS flow rate increased to provide additional operating margin in the analysis.
ECCS Flashing Fraction	15%	10%	ECCS flashing fraction recalculated based on Section 5.4 of RG 1.183.
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	1	1	No change
Percentage of Fuel Rods Damaged	100%	100%	No change
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU	0	0	No change
Water Level Above Damaged Fuel	23 ft	23 ft	No change
Pool Decontamination Factors	Elementary – 400 Organic - 1	Elementary – 500 Organic - 1	Decontamination Factors are as described in RG 1.183 Appendix B Section 2.
Delay Before Fuel Movement	100 hours	100 hours	No change
Containment Release Filtration	0%	0%	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**
Maximum Pre-Accident Iodine Spike Concentration	30 μCi/gm Dose Equivalent I-131	30 μCi/gm Dose Equivalent I-131	No change
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	Not provided	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%	1%	No change. Note- the MSLB does not result in failed fuel. This is a leaking fuel pre-condition included for conservatism.
RCS Mass	440,000 lbm	440,900 lbm	AST calculation more precisely accounts for CVCS mass.
Steam Generator Secondary Liquid Mass	168,000 lbm/SG	168,000 lbm/SG	No change
Intact Steam Generator Steam Release	0 – 2 hrs: 316,715 lbm 2 – 8 hrs: 703,689 lbm 8 – 24 hrs: 948,000 lbm	0 – 2 hrs: 316,715 lbm 2 – 8 hrs: 703,689 lbm 8 – 24 hrs: 948,000 lbm	No change
Primary-Secondary Leak Rate	0.65 gpm to two 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 uncover does not occur with intact SGs
Time to Cool RCS to 200°F	24 hrs	24 hrs	No change
Intact Steam Generator Iodine partition factor	10	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	0.1% (Alkali Metal Partition Factor =1000)	New AST value conservatively included.

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	30 µCi/gm Dose Equivalent I-131	30 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	Not provided	0% aerosol, 97% elemental, 3% organic	Iodine chemical form is per RG 1.183 Appendix F Section 4.
Percentage of Fuel Rods Failed	0%	1%	No change. Note- the SGTR does not result in failed fuel. This is a leaking fuel pre-condition included for conservatism.
RCS Mass	441,000 lbm	440,900 lbm	AST calculation more precisely accounts for CVCS mass.
Steam Generator Secondary Liquid Mass	105,000 lbm/SG	105,000 lbm/SG	No change
Intact Steam Generator Steam Release	0 – 0.09 hours – 1133 lbm/second 0.09 – 2 hours – 422,000 lbm 2 – 8 hours – 934,000 lbm	0 – 0.09 hours – 1133 lbm/second 0.09 – 2 hours – 422,000 lbm 2 – 8 hours – 934,000 lbm	No change
Ruptured Steam Generator Steam Release	0 – 0.09 hours – 1133 lbm/second 0.09 – 0.5 hours – 79,000 lbm	0 – 0.09 hours – 1133 lbm/second 0.09 – 0.5 hours – 79,000 lbm	No change
Feedwater Flow to Intact Steam Generators	0.09 – 2 hours – 327,000 lbm 2 – 8 hours – 981,000 lbm	0.09 – 2 hours – 327,000 lbm 2 – 8 hours – 981,000 lbm	No change
Time of Reactor Trip	0.09 hours	0.09 hours	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 – 0.09 hours – 21,600 lbm 0.09 – 0.5 hours – 79,000 lbm	0 – 0.09 hours – 21,600 lbm 0.09 – 0.5 hours – 79,000 lbm	No change
Duration of Ruptured Tube Break Flow	30 minutes	30 minutes	No change
Break Flow Flashing Fraction	0 – 0.09 hours – 21% 0.09 – 0.5 hours – 15%	0 – 0.09 hours – 21% 0.09 – 0.5 hours – 15%	No change
Duration of Intact SG Tube Uncovery After Reactor Trip	0 minutes	0 minutes	No change
Time to Cool RCS to 200°F	8 hours	8 hours	No change
Intact Steam Generator Iodine Partition Coefficient	100	100	No change
Intact Steam Generator Moisture Carryover Fraction	Not provided	0.1%	Carryover is provided for per RG 1.183 Appendix F Section 5.6.

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 isotopes and noble gases – 0.10 Other halogens – 0.05 Alkali metals – 0.12	AST gap fractions are per RG 1.183 Table 3
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 Steam Generator Iodine Source Term	0.1 µCi/gm	0.1 µCi/gm	No change
Iodine Chemical Form – Secondary Release	Not Provided	95% aerosol, 4.85% elemental, 0.15% organic	Although different from RG 1.183, Appendix H Section 5, this is acceptable because the removal mechanism for all chemical forms of iodine is the same for this pathway
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	2.03E6 ft^3	2.03E6 ft^3	No change
Containment Leakage Rate 0 to 24 hours 24 hours to 30 days	 0.15% 0.075%	 0.15% 0.075%	No change
Containment Leakage Filtration	0%	0%	No change
Natural Deposition in Containment	50% plateout of RCS release	Elemental iodine – None Aerosols – 2.74E-2 hr^{-1}	Natural deposition is credited per RG 1.183 Appendix H Section 6.1.
Iodine/Particulate Removal by Containment Sprays	Not provided	5.0 hr^{-1}	Particulate removal by Containment Spray is per RG 1.183 Appendix H Section 6.1.
RCS Mass	435,000 lbm	440,900 lbm	The AST calculation reflects the liquid mass resulting from SG replacement.
Steam Generator Secondary Liquid Mass	168,000 lbm	168,000 lbm	No change
Primary-Secondary Leak Rate	150 gallons per day per SG	1 gpm total	AST value increased for additional conservatism.
Density Used for Leakage Volume-to-Mass Conversion	62.4 lbm/ft^3	62.4 lbm/ft^3	No change
Secondary Steam Release	426,000 lbm	426,000 lbm	No change
Time until Primary and Secondary Pressures Equalize	2500 seconds	2500 seconds	No change
Duration of SG Tube Uncovery Following Reactor Trip	0 minutes	0 minutes	No change
Steam Generator Iodine Partition Coefficient	10	100	RG 1.183 Appendix H Section 7.4 allows an iodine partition factor of 100 for SG releases.
Steam Generator Moisture Carryover Fraction	Not provided	0.1%	Carryover is provided for per RG 1.183 Appendix H Section 7.4.

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	Iodine – 12%	I-131 - 0.08 Kr-85 - 0.10 Other Halogens and Noble Gases - 0.05 Alkali Metals - 0.12	AST gap fractions are per RG 1.183 Table 3
Fuel Rod Peaking Factor	Not provided	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	0
Initial Steam Generator Iodine Source Term	0.1 µCi/gm	0.1 µCi/gm	No change
Iodine Chemical Form	Not provided	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	441,000 lbm	440,900 lbm	AST calculation more precisely accounts for CVCS mass.
Steam Generator Secondary Liquid Mass	168,000 lbm	168,000 lbm	No change
Primary-Secondary Leak Rate	150 gallons per day per SG	1 gpm total	AST value increased for additional conservatism.
Density Used for Leakage Volume-to-Mass Conversion	62.4 lbm/ft^3	62.4 lbm/ft^3	No change
Secondary Steam Release	426,000 lbm	426,000 lbm	No change
Time until Primary and Secondary Pressures Equalize	2500 seconds	2500 seconds	No change
Duration of SG Tube Uncovery Following Reactor Trip	0 minutes	0 minutes	No change
Steam Generator Iodine Partition Coefficient	10	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	Not provided	0.1%	Carryover is provided for per RG 1.183 Appendix G Section 5.6.

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Enclosure 13

FNP AST LAR Supporting Information

- **LOCA RADTRAD Input Files in CD Format**
- **LOCA RADTRAD Output Files in CD Format**
- **FHA RADTRAD Input Files in CD Format**
- **FHA RADTRAD Output Files in CD Format**
- **ARCON96 Files For RWST X/Qs**

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Enclosure 14

Summary of Regulatory Commitments

Enclosure 14

Summary of Regulatory Commitments

The following table identifies the regulatory commitments in this document. Any other statements in this submittal represent intended or planned actions. They are provided for information purposes and are not considered to be regulatory commitments.

REGULATORY COMMITMENTS	DUE DATE/EVENT
1. Administrative controls will be established to ensure appropriate personnel are aware of the open status of the penetration flow path(s) during core alterations or movement of irradiated fuel assemblies within the containment.	Prior to implementation of the LAR
2. Existing administrative controls for open containment airlock doors will be expanded to ensure specified individuals are designated and readily available to isolate any open penetration flow path(s) in the event of an FHA inside containment.	Prior to implementation of the LAR
3. With the Personnel Airlock open during fuel handling operations or core alterations, the Containment Purge System will be in operation.	Prior to implementation of the LAR
4. In the event of an FHA, the containment will be evacuated and the Personnel Airlock will be closed within 30 minutes of detection of the accident.	Prior to implementation of the LAR
5. In the event of an FHA, Control Room occupants will use the secondary door to the Control Room for ingress and egress.	Prior to implementation of the LAR