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U. S. Nuclear Regulatory Commission
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Joseph M. Farley Nuclear Plant Units 1 and 2
Response to the Second Request for Additional Information Regarding
Alternative Source Term License Amendment Request

Ladies and Gentlemen:

By letter dated November 22, 2016 (Accession Number ML16336A024), Southern Nuclear Operating Company requested the Nuclear Regulatory Commission to review and approve proposed revisions to the licensing basis of Farley Nuclear Plant. The revisions would support application of Alternative Source Term (AST) methodology. The Proposed Technical Specification changes, supported by the AST Design Basis Accident radiological consequence analyses, were included in the license amendment request. 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.

By letter dated July 21, 2017 (ML17194A787), the NRC issued a request for additional information (RAI). Enclosure 1 responds to this request. Enclosure 2 contains revised pages for the original license amendment request (LAR) reflecting changes made by RAIs 25 and 31 (control rod ejection accident).

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at 205.992.7369.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 7, 2017.

Respectfully submitted,

J.J. Hutto
Regulatory Affairs Director

JJH/CBM/CG

Enclosures:
1. SNC Response to the Second Request for Additional Information
2. Revised LAR pages reflecting changes made by RAIs 25 and 31 (control rod ejection accident)

cc: Regional Administrator
NRR Project Manager – Farley
Senior Resident Inspector – Farley
RTYPE: CFA04.054

Joseph M. Farley Nuclear Plant Units 1 and 2
Response to Request for Additional Information Regarding
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Enclosure 1

SNC Response to the Second Request for Additional Information

Regulatory Analysis Basis

1. Section 10 CFR Part 50.67, "Accident Source Term," allows licensees seeking to revise their current accident source term in design basis radiological consequence analyses to apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents (DBAs) previously analyzed in the safety analysis report. Section 50.67(b)(2) requires that the licensee's analysis demonstrates with reasonable assurance that:

   (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

   (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

   (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

2. NUREG-1431, "Standard Technical Specifications Westinghouse Plants Revision 4.0,"Volume 1, Specifications dated April 2012 contains the improved standard technical specifications (STS) for Westinghouse plants. The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 Federal Register (FR) 39132), which was subsequently codified by changes to 1O CFR 50.36 (60 FR 36953). Licensees adopting portions of the improved STS to existing TS should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.

3. NUREG-0800, Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003734190).

4. NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

5. NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," D. A Powers, et.al., USNRC, July 1996 (ADAMS Accession No. ML 100130305).

6. Technical Specification Task Force (TSTF) Traveler, TSTF-312, Revision 1, "Administratively Control of Containment Penetrations," (ADAMS Accession No. ML040620147).

7. NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Volume 2, Revision 4.0, April 2012 (ADAMS Accession No. ML12100A228).

Supplement Request for RAI No. 2 (loss-of-coolant accident (LOCA))

Regulatory Basis numbered 1, 3, 4, and 5 above apply to this request.

In a letter dated March 24, 2017, RAI No. 2, the NRC staff asked SNC to explain how the removal coefficient(s) were calculated, to discuss how the assumptions are consistent with RG 1.183, and to provide enough detail to allow the NRC staff to confirm the methodology is consistent with NUREG/CR-6189 as applicable.

SNC's response to RAI No. 2 states:

> "...An aerosol natural deposition rate of 0.1 hr^{-1} is applied based upon values presented in Section VI NUREG/CR-6189."

However, SNC's response did not explain how this value is consistent with Section VI NUREG/CR-6189. The NRC staff reviewed NUREG/CR-6189 and determined that an aerosol natural deposition rate of 0.1 hr^{-1}does not seem to be consistent with NUREG/CR-6189 Section VI because it overestimates the aerosol natural deposition rate in four out of the five gap time intervals stated in Table 36 of NUREG/CR-6189. NUREG/CR-6189 provides a simplified model of aerosol removal by natural processes in reactor containments which applies to both the control rod ejection accident (CREA) in containment and loss-of-coolant accident (LOCA).

In response to RAI No. 22, SNC provided the calculated effective decontamination coefficient correlations, also known as the aerosol natural deposition rates, for the five gap time intervals stated in Table 36 of NUREG/CR-6189. These deposition rates were provide for the CREA analysis. In the CREA analysis, SNC chose to use the lowest aerosol natural deposition rate which is a conservative assumption because it removes the least amount of aerosols from the containment atmosphere. However, in the LOCA analysis, SNC is applying an aerosol natural deposition rate of 0.1 hr^{-1} for all five gap time intervals which does not appear to be a conservative assumption. The simplified approach used in NUREG/CR-6189 does not vary with different DBAs. Therefore, the aerosol natural deposition rates calculated for the CREA also apply to the LOCA analysis. The calculated aerosol natural deposition rates for the LOCA analysis should reflect each of the NUREG/CR-6189 gap time intervals. It is non-conservative to apply a later time-period (13680 to 49680) aerosol natural deposition rate to the earlier and later time-periods (0-1800, 1800-6480, 6480-13680, and 49680-80000).

Please explain the technical safety basis for applying an aerosol natural deposition rate of 0.1 hr^{-1} *or* provide aerosol natural deposition rate(s) for LOCA that are consistent with NUREG/CR-6189 and provide the revised LOCA onsite and offsite resultant doses that reflect the aerosol natural deposition rate(s).

SNC RESPONSE:

In the evaluation of the dose consequences for the LOCA, a natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region of containment as well as in the sprayed region when sprays are not operating. Industry Degraded Core Rulemaking (IDCOR) Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," December 1983, documents results from Containment Systems Experiment testing. These tests show that settling of aerosols due to gravity is the dominant natural mechanism for fission product retention. This report finds that significant removal by sedimentation would be expected even at very low particulate concentrations. Figure 4-2 of IDCOR Program Technical Report 11.3 shows a ten-fold reduction in the airborne cesium concentration over a 7-hour period at relatively low concentrations. This represents an aerosol removal rate of 0.33 hr^{-1}. A more conservative value of 0.1 hr^{-1} is used in the analysis.

Examples where this same technical basis has been applied for other approved LAR submittals include those for the St. Lucie Unit 2 License Amendment No. 152 in September 2008 (ADAMS Accession No. ML082060400) and for the Palisades Nuclear Plant License Amendment No. 226 in September 2007 (ADAMS Accession No. ML072470667).

Other examples are provided in the table below:

Plant	Accession No.	Comments
Indian Point	SER: ML003727500	Section 3.1
Seabrook	Submittal: ML032890198 SER: ML050250200	Enclosure 2, Section 2.1.2.4 Section 3.4.1
Shearon Harris	Submittal: ML012350079	Section 2.22.4.3.1
Point Beach	Submittal: ML083450683 SER: ML110240054	Table 18 Section 2.1.2.2.1
D. C. Cook	Submittal: ML14324A209 SER: ML16242A111	Table 3.1-2 Section 3.3.2
Turkey Point	Submittal: ML101800220 SER: ML110800666	Section 2.1.2.4 Section: 3.1.1.2.1
St. Lucie Unit 1 – AST	Submittal: ML072000250 SER: ML082682060	Attachment 5, Section 2.1.2.4 Section 3.1.1.2.1
St. Lucie Unit 1 – EPU	Submittal: ML101160193 SER: ML12181A019	Attachment 5, Table 2.9.2-12 Section 2.9.2.1.2.2.1
St. Lucie Unit 2 – EPU	Submittal: ML110730299 SER: ML12235A463	Attachment 5, Table 2.9.2-12 Section 2.9.2.1.2.2.1

Supplement Request for RAI No. 15 (Fuel Handling Accident (FHA))

Regulatory Basis numbered 1 and 4 above apply to this request.

In letter dated March 24, 2017, RAI No. 15, the NRC staff asked SNC to provide evaluations that analyzed the FHA in containment for multiple configurations with regard to the allowances of TS 3.9.3 and are consistent with RG 1.183 and meet the limits in RG 1.183, SRP 15.0.1, and 10 CFR 50.67.

SNC's response to RAI No. 15 states that the calculated control room dose for the FHA in the spent fuel pool is 0.1 rem. However, in the license amendment request, results and acceptance limits in Enclosure 7 states that the calculated control room dose for the FHA in the spent fuel pool is 0.2 rem.

Please clarify which calculated control room dose for the FHA in the spent fuel pool result, 0.2 rem or 0.1 rem, is correct. If the correct result is that stated in SNC's response to RAI No. 15, then please explain why the result changed.

SNC RESPONSE:

For the FHA in the spent fuel pool area, the calculated control room dose is 0.1 REM. The Enclosure 7 statement is a typographical error and is incorrect.

Supplement Request for RAI No. 17 (FHA)

Regulatory Basis numbered 1, 4, 6, and 7 apply to this request.

In letter dated March 24, 2017, RAI No. 17, the NRC staff asked SNC to explain why SNC did not appear to provide for a provision to manage flow paths to isolate any open containment penetration flow paths immediately upon a detection of a FHA or a provision to isolate flow paths upon a FHA.

In the license amendment request dated November 22, 2016, Regulatory Commitment #2, states:

> 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.

SNC's response to RAI No. 17 states:

> Item #2, above, is the provision for the isolation of the flow path, which is consistent with TSTF-312.
>
> Under SNC's LLRT [local leak rate test] procedures, personnel are stationed at the containment penetration being tested. Therefore, if an FHA were to occur at the same time an LLRT is being

> conducted, the LLRT personnel would be immediately available to isolate the penetration.
>
> The proposed change also includes the addition of text to the LCO [limiting conditions of operation] discussion in Bases 3.9.3 stipulating that the administrative controls that are put in place when penetration 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 and NUREG-1431 states:

> The allowance to have containment personnel airlock doors open and penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement and CORE ALTERATIONS is based on... (2) <u>*commitments from the license to implement acceptable administrative procedures that ensure*</u> in the event of a refueling accident (even though the containment fission product control function is not required to meet acceptable dose consequences) that the open airlock can and will be promptly closed following containment evacuation and that <u>*the open penetration(s) can and will be promptly closed*</u>. [Emphasis added]

Your application states that individuals are "designated and readily available to isolate," in the event of an FHA. TSTF-312 states that the open penetration "can and will be promptly closed." Please clarify if the open penetration flow path(s) will be promptly closed in the event of an FHA.

ANSWER:
The intention of the LAR submittal and previous RAI response has been to fully comply with the requirement outlined in TSTF-312. Specifically, SNC is committing to implement acceptable administrative procedures that ensure in the event of a refueling accident that the open airlock can and will be promptly closed following containment evacuation and that the open penetration(s) can and will be promptly closed.

<u>**Supplement Request for RAI No. 25**</u> (CREA)

Regulatory Basis numbered 1 and 4 above apply to this request.

In letter dated March 24, 2017, RAI No. 25, the NRC staff asked SNC to provide the plant specific evaluation that determined that the chemical form of radioiodine released from the steam generators of 95% cesium iodide, 4.85% elemental iodine, and 0.15% organic iodide is conservative and show that the iodine does not re-evolve.

SNC's response dated May 23, 2017, to RAI No. 25 states:

> ...As the releases from both primary and secondary systems are directly to the environment, no credit is taken for iodine deposition within containment. With no iodine removal, re-evolution is not a concern for the secondary system release scenario.
>
> Given that particulates are more likely than other chemical forms to be removed via deposition, assuming the iodine to be mostly in particulate form would be non-conservative if deposition were credited. As deposition is not credited, however, the release to the environment is not affected by the speciation of iodine. The speciation becomes relevant only when calculating the control room dose, as all iodine species do not have the same control room recirculation filter efficiency. As indicated in Table 3.9a of LAR Enclosure 1, elemental and organic iodine have a recirculation filter efficiency of 94.5%, whereas particulate iodine has an efficiency of 98.5%. With most of the iodine removed by the intake filter, which has the same efficiency for all forms of iodine, the small difference in recirculation filter efficiency has a negligible impact on control room dose...

RG 1.183 Appendix H provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a control rod ejection accident at pressurized water reactors. RG 1.183 Appendix H regulatory positions 3 and 5 state:

> 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.
>
> Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.

Two release cases are considered for the CREA per RG 1.183: (1) release to the containment; and, (2) release to the secondary system. Evaluating both pathways shows that the regulatory limits are met for each possible pathway, and therefore public and operator health and safety is maintained, regardless of the release pathway. The secondary release path does not flow into containment but flows from the RCS through the steam generators in the secondary system. Although, it is conservative to assume the release is directly from the RCS to the environment without filtering or hold up, it is not conservative to assume a speciation of 95% cesium iodide, 4.85% elemental iodine, and 0.15% organic iodide instead of 97% elemental iodine and 3% organic iodine. Especially when it is considered that the control room recirculation filter efficiency is higher for removal of particulates than elemental iodine. Assuming a speciation of 95%

cesium iodide, 4.85% elemental iodine, and 0.15% organic iodide results in an inaccurate and non-conservative control room dose result.

In the response to RAI No. 25, SNC also stated:

> The impact of slightly over-crediting the recirculation filter efficiency is also insignificant in light of the conservatism in assuming that all accident-induced leakage from primary to secondary system goes directly to the environment, neglecting the benefits of mixing and holdup in the steam generators before being released to the environment through the main steam safety valves.

Using one conservatism to offset another is not consistent with RG 1.183. RG 1.183 does allow use of sensitivity or scoping evaluations. When there is no direct correlation between two assumptions, it is difficult to justify that one conservatism will offset another, without providing the technical analysis (e.g., sensitivity or scoping evaluation) that determined that the overall dose result will be conservative.

Please provide the technical analysis (e.g., sensitivity or scoping evaluation) that determined that the overall dose result will be conservative, while assuming a speciation of 95% cesium iodide, 4.85% elemental iodine, and 0.15% organic iodide *or* revise the CREA secondary release pathway to assume a speciation of 97% elemental iodine and 3% organic iodine.

SNC RESPONSE:

SNC has taken the above statements into account and has revised the CREA analysis to better address the RG-1.183 regulatory position. The revised analysis uses a speciation of 97% elemental iodine and 3% organic iodine for the secondary release pathway. The final results of the revised analysis are provided in the response to RAI #31.

Supplement Request for RAI No. 31 (CREA)

Regulatory Basis numbered 1 and 4 above apply to this request.

In letter dated March 24, 2017, RAI No. 31, the NRC staff asked SNC to review the data in Table 2 of Enclosure 10 and provide an update to Table 2 as necessary.

SNC's response to RAI No. 31 states:

> RG 1.183, Appendix H, regulatory position 1 indicates that of the iodine contained within the fraction of core that experiences fuel melt, 25% is available for release from the containment and 50% is available for release from the reactor coolant. To simplify the dose analysis for Farley, it is assumed that 50% of the iodine in the melted fuel is available for release from both the reactor coolant and the containment. To offset this conservatism of a factor of two

> (50% instead of 25%) in containment release, the radial peaking factor of 1.7 is not applied to the melted fuel. While this is non-conservative for the reactor coolant source, which is eventually released to the environment via the secondary system, the dose contribution from this pathway is small compared to the containment release because the secondary release terminates at 225 sec while the containment release continues for 30 days...

RG 1.183 establishes an acceptable accident source term and identifies the significant attributes of other accident source terms that may be found acceptable by the NRC staff. It also provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an accident source term. Using one conservatism to offset another is not consistent with RG 1.183.

The NRC staff has reviewed the information in the submittal dated November 22, 2016, and in supplements dated May 23 and June 8, 2017. The proposed accident source term for the CREA does not appear to be consistent with RG 1.183. In addition, the evaluated dose results are non-conservative. An accident source term equivalent to that discussed in RG 1.183 has not been provided for the CREA. The radial peaking factor is applied to all the nuclides, not just the iodine isotopes. Therefore, the rest of the nuclides are non-conservative and the activity from the fuel melt is a factor of 1.7 to low for the CREA release in containment. In addition, the source term for the secondary system release case for all nuclides appears to be non-conservative and the activity from the fuel melt is a factor of 1.7 to low.

Please provide a source term for both pathways that accounts for the radial peaking factor *or* provide an accident source term equivalent to that discussed in RG 1.183 for CREA that is of the same level of quality as the source terms in NUREG-1465.

SNC RESPONSE:

SNC has taken the above statements into account and has revised the CREA analysis to better address the RG-1.183 regulatory position. The revised analysis separates the offsite and control room doses for each pathway. The results are as follows:

Pathway	CR Dose (Rem)	EAB Dose (REM)	LPZ Dose (REM)
Primary Release	2.7	2.5	1.9
Secondary Release	0.054	0.49	0.18

Supplement Request for RAI No. 35 (Locked Rotor Accident (LRA))

Regulatory Basis numbered 1 and 4 above apply to this request.

In letter dated March 24, 2017, RAI No. 35, the NRC staff asked SNC to both correct the gap fraction for the isotopes of bromine and provide the updated onsite and offsite dose results or explain the deviation from RG 1.183.

SNC's response dated May 23, 2017, to RAI No. 35 states:

> Table 2 of Enclosure 11 is incorrect. The LR [locked rotor] analysis documented in Enclosure 11 was performed using gap fractions consistent with RG 1.183, Table 3.

Please clarify if the dose results provided in the original submittal and those in RAI No. 36 reflect the 0.05 gap fraction for the isotopes of bromine consistent with RG 1.183, Table 3. In addition, please provide the gap release activities used for the isotopes of bromine in the LRA analysis.

SNC RESPONSE:

The original (LOCADOSE) and revised (RADTRAD) LRA dose consequences evaluations used a gap fraction of 0.05 for the bromine isotopes. The available gap release activities used by the RADTRAD code are shown in the table below:

Table 7-2: RCS Activities

Isotope	Core Activity (Ci) Note 1	Margin Factor Note 2	Gap Fraction Note 3	RCS Activity (Ci/MWt) Note 4
Br-82	3.8E+05	1.03	0.05	2.350E+00
Br-83	9.7E+06	1.03	0.05	6.000E+01
Br-84	1.7E+07	1.03	0.05	1.051E+02
Br-85	2.1E+07	1.03	0.05	1.299E+02
Br-86	1.5E+07	1.03	0.05	9.278E+01
Br-87	3.4E+07	1.03	0.05	2.103E+02
Br-88	3.6E+07	1.03	0.05	2.227E+02

Notes:

1. Core Activity – From Westinghouse document.
2. Margin Factor – Accounts for cycle variations.
3. Gap Fraction – From RG 1.183
4. RCS Activity (Ci/MWt) = {(Eq. Core Activity) * (Margin Factor) * (Gap Fraction) * (Failed Fuel Fraction, 0.2) * (Radial Peaking Factor, 1.7)}/(Core Thermal Power, 2831 MWth)
5. Br-85, Br-86, Br-87, Br-88 have very short half-lives. Also, the FGR 11 and 12 databases do not contain dose conversion factors for these isotopes. Therefore, in the

RADTRAD calculations, these isotopes do not significantly contribute to the dose results.

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Enclosure 2

Revised LAR pages reflecting changes
made by RAIs 25 and 31 (control rod ejection accident)

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	25%
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	325 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
Containment release	2.5	1.9	2.7
Secondary release	0.5	0.2	< 0.1
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

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.	Conforms - Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.
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 containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.	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.

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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 3

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)		
	Containment Release	Secondary Release	Limit
EAB	2.5	0.5	6.3
LPZ	1.9	0.2	6.3
Control Room	2.7	<0.1	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 25% 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	Activity in Core (Ci)	Margin Factor	Core Release		Gap Fraction	Activity (Ci)				
						Fuel Melt		Gap	Total	
			Ctmt	RCS		Ctmt	RCS	Ctmt/RCS	Ctmt	RCS
I-131	7.5E+07	1.02	0.25	0.50	0.10	8.1E+04	1.6E+05	1.3E+06	1.4E+06	1.5E+06
I-132	1.1E+08	1.02	0.25	0.50	0.10	1.2E+05	2.4E+05	1.9E+06	2.0E+06	2.1E+06
I-133	1.6E+08	1.02	0.25	0.50	0.10	1.7E+05	3.5E+05	2.8E+06	2.9E+06	3.1E+06
I-134	1.7E+08	1.02	0.25	0.50	0.10	1.8E+05	3.7E+05	2.9E+06	3.1E+06	3.3E+06
I-135	1.5E+08	1.02	0.25	0.50	0.10	1.6E+05	3.3E+05	2.6E+06	2.8E+06	2.9E+06
Kr-83m	9.7E+06	1.02	1.00	1.00	0.10	4.2E+04	4.2E+04	1.7E+05	2.1E+05	2.1E+05
Kr-85m	2.1E+07	1.02	1.00	1.00	0.10	9.1E+04	9.1E+04	3.6E+05	4.6E+05	4.6E+05
Kr-85	7.2E+05	1.15	1.00	1.00	0.10	3.5E+03	3.5E+03	1.4E+04	1.8E+04	1.8E+04
Kr-87	4.0E+07	1.02	1.00	1.00	0.10	1.7E+05	1.7E+05	6.9E+05	8.7E+05	8.7E+05
Kr-88	5.7E+07	1.02	1.00	1.00	0.10	2.5E+05	2.5E+05	9.9E+05	1.2E+06	1.2E+06
Kr-89	6.9E+07	1.02	1.00	1.00	0.10	3.0E+05	3.0E+05	1.2E+06	1.5E+06	1.5E+06
Xe-131m	8.4E+05	1.02	1.00	1.00	0.10	3.6E+03	3.6E+03	1.5E+04	1.8E+04	1.8E+04
Xe-133m	4.8E+06	1.02	1.00	1.00	0.10	2.1E+04	2.1E+04	8.3E+04	1.0E+05	1.0E+05
Xe-133	1.5E+08	1.05	1.00	1.00	0.10	6.7E+05	6.7E+05	2.7E+06	3.3E+06	3.3E+06
Xe-135m	3.0E+07	1.02	1.00	1.00	0.10	1.3E+05	1.3E+05	5.2E+05	6.5E+05	6.5E+05
Xe-135	3.5E+07	1.02	1.00	1.00	0.10	1.5E+05	1.5E+05	6.1E+05	7.6E+05	7.6E+05
Xe-137	1.4E+08	1.02	1.00	1.00	0.10	6.1E+05	6.1E+05	2.4E+06	3.0E+06	3.0E+06
Xe-138	1.3E+08	1.02	1.00	1.00	0.10	5.6E+05	5.6E+05	2.3E+06	2.8E+06	2.8E+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.

<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

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	315 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