UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 24, 2017

Mr. James J. Hutto
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
P.O. Box 1295, Bin 038
Birmingham, AL 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – REQUEST FOR ADDITIONAL INFORMATION (CAC NOS. MF8861, MF8862, MF8916, MF8917, MF8918, MF8919)

Dear Mr. Hutto:

By letter dated November 22, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16336A024), the Southern Nuclear Operating Company, Inc., (SNC) submitted an amendment request to revise the Joseph M. Farley Nuclear Plant, Unit 1 and Unit 2, Technical Specifications (TS). Specifically, SNC requested to:

1. Revise the licensing basis to support a selected scope application of an Alternative Source Term (AST) methodology (CAC Nos. MF8861, MF8862);

2. Incorporate Technical Specification Task Force (TSTF) Traveler, TSTF-448-A, Revision 3, "Control Room Habitability" (CAC Nos. MF8916, MF8917); and,

3. Incorporate TSTF-312-A, "Administrative Control of Containment Penetrations" (CAC Nos. MF8918, MF8919).

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that a request for additional information (RAI) is needed for NRC staff to complete its review, as provided in the Enclosure. The enclosure was discussed with your staff on March 15, 2017. SNC agreed to respond within 75 days of this letter to RAI No. 15 and RAI No. 40, and within 60 days for the remaining RAIs. Please note that the NRC staff's review is continuing and further requests for information may be developed.

Sincerely,

Shawn Williams

Shawn Williams, Project Manager
Plant Licensing Branch, II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348, 50-364

Enclosure:
Request for Additional Information
cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

ALTERNATIVE SOURCE TERM, TSTF-448, TSTF-312

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NOS. 50-348 AND 50-364

By letter dated November 22, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession Package Number ML16336A024), Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted a License Amendment Request (LAR) to revise Joseph M. Farley Nuclear Plant, Units 1 and 2, (FNP) Technical Specifications (TS). The proposed change would revise FNP TS 3.7.10, "Control Room," TS 3.9.3, "Containment Penetrations," TS 5.5.18, "Control Room Integrity Program (CRIP)," and the current licensing basis to implement an alternative radiological source term for evaluating design basis accidents as allowed by Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.67, "Accident Source Term." The Nuclear Regulatory Commission (NRC) staff reviewed the impact of implementing an alternative radiological source term for evaluating design basis accidents (DBAs) on all DBAs currently analyzed in the FNP updated final safety analysis report (UFSAR) that could have the potential for significant dose consequences. The NRC staff determined that more information is needed to complete the review.

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 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) 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) TEDE for the duration of the accident.

2. Plant Design Criterion 19, "Control Room," provides for conformance with Section 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," states, in part, that whenever a holder of an operating license under this part, desires to amend the license, application for an amendment must be filed with the Commission, as specified in § 50.4 of this chapter, as applicable, fully describing

the changes desired, and following as far as applicable, the form prescribed for original applications.

3. Section 10 CFR Part 50, Appendix A, "General Design Criterion (GDC) for Nuclear Power Plants": GDC 19, requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. It also states that adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

4. The regulation at Subpart H of 10 CFR 20, "Standards for Protection against Radiation," provides the requirements for respiratory protections and controls to restrict internal exposure in restricted areas. Specifically, 10 CFR 20.1701 states that licenses shall use, to the extent practical, process or engineering controls to control the concentration of radioactivity in the air. Use of other controls as described in 10 CFR 20.1702 is allowed by regulation when it is not practical to apply process or other engineering controls to control the concentrations of radioactive material in the air.

5. Section 10 CFR 50.36, "Technical Specifications," contains the NRC's regulatory requirements related to the content of the Technical Specifications (TSs). This regulation requires that the TS include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notifications; and (8) written reports.

6. 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 10 CFR 50.36 (60 FR 36953). Licensees adopting portions of the improved STS to existing technical specifications should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.

7. NUREG-0800, Standard Review Plan (SRP) Section 6.4, "Control Room Habitability System," Revision 3, March 2007; Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4, March 2007; and Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms," Revision 0, July 2000.

8. 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).

9. NRC RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Revision 1, January 2007 (ADAMS Accession No. ML063560144).

10. NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," D.A. Powers and S.B. Burson, USNRC, June 1993.

11. NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," D.A Powers, et.al., USNRC, July 1996.

12. NUREG-0800, "Standard Review Plan [SRP]," 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms."

13. TSTF-448-A, Revision 3, "Control Room Habitability"

Request for Additional Information (RAI) No. 1 (Loss of Coolant Accident (LOCA))

Regulatory Basis numbered 1, 2, 3, 7, 8, and 9 apply to RAI No. 1.

In the LAR, Table 3.5a of Enclosure 1 provides the parameters and assumptions for the LOCA analysis including atmospheric dispersion factors (x/Qs) for the containment, purge, plant vent, and refueling water storage tank (RWST) leakage pathways to the control room. The NRC staff has reviewed Table 3.5a of Enclosure 1 and has noted the following:

- The plant vent x/Qs appears to include:
  - The most limiting release-receptor x/Q location from Table 3.4a of Enclosure 1 for each time period,
  - A 60 second control room isolation time, as shown by the following time periods 0 – 0.0167 hour and 0.0167 – 2 hour, and
  - The most limiting release-receptor x/Q location for the normal control room intake receptor (i.e., Unit 2 reactor to normal control room intake x/Q of 2.79E-03 seconds per cubic meters (sec/m^3)).

- The control room x/Qs for the containment leakage pathway *does not* appear to include:
  - The 60 second control room isolation time because the time periods 0-0.0167 hour and 0.0167 - 2 hour time periods are missing, and
  - The most limiting release-receptor x/Q location for the normal control room intake receptor because the 3.88E-03 sec/m^3 x/Q for the unit 2 vent to normal control room intake for the time period of 0 – 0.0167 hour is missing.

- The control room x/Qs for the RWST release pathway *does not* seem to include:
  - The 60 second control room isolation time because the time periods 0-0.0167 hour and 0.0167 - 2 hour time periods are missing.

In addition, the NRC staff reviewed the LOCA RADTRAD files and has noted the following:

- The LOCA_ESF_325 RADTRAD file does not reflect the 60 second control room isolation time (i.e., 0 – 0.0167 hour and 0.0167 – 2 hour time periods are missing), and the most limiting release-receptor x/Q location for the normal control room intake receptor is missing (i.e., unit 2 reactor to normal control room intake x/Q of 2.79E-03 sec/m^3).

- The LOCA_Contain_325 RADTRAD file does reflect the 60 second control room isolation time and the most limiting release-receptor x/Q location for the normal control room intake receptor.

There appears to be inconsistencies between the individual release pathways for the control room ventilation system operation. In addition, there are inconsistencies between the parameters and assumptions for the LOCA analysis stated in Table 3.5a of Enclosure 1 and the control room modeling in the RADTRAD files.

Please clarify the inconsistencies and explain why the containment leakage and RWST pathways don't reflect the control room isolation time and transfer of x/Qs from the most limiting normal control room intake to the most limiting control room emergency intake.

RAI No. 2 (LOCA)

Regulatory Basis numbered 1, 2, 7, and 8 apply to RAI No. 2.

RG 1.183 Appendix A position 3.2 states:

> 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.

In the LAR, Table B of Enclosure 5 states FNP's conformance with RG 1.183 Appendix A. Table B states that FNP's analysis for RG 1.183 regulatory position 3.2 is, "Conforms - An aerosol natural deposition rate of 0.1 h^{r-1} is assumed based upon values presented in Section VI of NUREG/CR-6189." Enclosure 6, "Loss of Coolant Accident Analysis," containment leakage parameters states that the natural deposition is 0.1 $hours^{-1}$ after sprays are terminated for aerosols only. Enclosure 12 provides comparison tables of the new Alternative Source Term (AST) values compared to the current licensing basis values. Table 2 of Enclosure 12 the LOCA inputs and assumptions; states that for natural deposition the current licensing basis assumes none for elemental, organic, and aerosol iodine and that the new AST assumes none for elemental, and organic iodine and 0.1 hr^{-1} in unsprayed regions only for aerosol iodine.

During the NRC staff's review of the RADTRAD LOCA_Contain_325 file, the staff noticed that:

- There are two natural deposition models, one for the sprayed portion of the containment and one for the unsprayed portion of the containment.
- Both containment natural deposition models show 2.88 $hour^{-1}$ for elemental iodine removal.

The RADTRAD modeling does not appear consistent with the inputs and assumptions in the LAR, and the LAR is not consistent between the various enclosures and tables.

Please explain what assumptions are being used for natural deposition of elemental, organic and aerosol iodine in the unsprayed and sprayed portion of the containment. Explain how the removal coefficient(s) were calculated and discuss how the assumptions are consistent with

RG 1.183. Provide enough detail to allow the NRC staff to confirm the methodology is consistent with NUREG-0800 Chapter 6.5.2 and/or NUREG/CR-6189, as applicable.

RAI No. 3 (LOCA)

Regulatory Basis numbered 1, 2, 7, and 8 apply to RAI No. 3.

In the LAR, Table 3.5a of Enclosure 1 lists the parameters and assumptions for the LOCA. Enclosure 6 contains the LOCA analysis. However, there are inconsistencies between the two Enclosures. For example:
Table 3.5a of Enclosure 1 states:

- The iodine species emergency core cooling system (ECCS) leakage released to the atmosphere is 100% elemental and 0% organic.
- ECCS leakage rate to the RWST is 1 gallon per minute.
- RWST leakage iodine flashing factors are 0% to 13.9%.

Enclosure 6 states:

- The iodine species ECCS leakage released to the atmosphere from RWST is 100% elemental and 0% organic.
- ECCS leakage rate to the RWST is 2 gallons per minute.
- RWST leakage iodine flashing factors varies, max is 265° Fahrenheit.

In the LAR, Control Room Emergency Filtration/Pressurization System (CREFS) is the only credited mitigating system for filtration of iodine, neither enclosure lists any other credited filtration systems. FNP's current licensing basis as reflected in UFSAR 15.4.1.7.4, "Penetration Room Model," states:

> During normal operation, the penetration room filtration system is aligned to the spent-fuel pool area. After a LOCA, the penetration room filtration system could be manually switched to the penetration room. Any containment leakage through the penetrations would then be filtered prior to release to the environment. However, it has been conservatively assumed that all containment leakage going into the penetration room is released to the atmosphere at ground level without being filtered for the duration of the accident.

However, the RADTRAD file, LOCA_ESF_325 shows ECCS leakage is exhausted through the penetration room filtration system with credit for removal of 89.5% of the elemental, organic, and aerosol iodine. Because of the inconsistencies, it is not apparent exactly what is being proposed as the new design basis environmental/radiological consequences associated with the LOCA.

Please provide:

1. A summary statement of <u>each</u> individual containment release pathway (i.e., containment mini-purge system, containment leakage, ECCS leakage, and RWST back leakage), in enough detail that allows the NRC to perform an independent assessment, and include any filtration systems being credited in addition to CREFS. Explain any deviations from RG 1.183. For example, if the chemical form of iodine released from the ECCS or

RWST is assumed to be 100% elemental and 0% organic this is a deviation from RG 1.183 that requires explanation.

2. State the specific assumptions and inputs for each pathway, include the specific numbers used in the analyses. Please be specific about the iodine flashing fraction for the duration of the ECCS/RWST leakage, such as flashing fraction is A% from time X until time Y, flashing fraction is B% from time Y until time Z, etc.

RAI No. 4 (LOCA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 4.

In the LAR, Enclosure 6 provides the LOCA containment leakage parameters. Specifically, it provides the total containment volume, the sprayed containment volume, and the unsprayed containment volume. During its review, NRC staff noted that the sprayed and unsprayed volumes do not equal the total containment volume.

Please confirm the containment numbers stated in the analysis and explain the reason for any missing containment volume. Please correct any errors.

RAI No. 5 (LOCA)

Regulatory Basis numbered 1, 2, 7, 8, and 9 apply to RAI No. 5.

NRC staff noted that the LOCA and fuel handling accident (FHA) analyses assume CREFS in leakage is 325 cfm and the main steam line break (MSLB), steam generator tube rupture (SGTR), control rod ejection accident (CREA), and locked rotor accident (LRA) analyses assume CREFS in leakage is 310 cfm.

Please explain the difference in CREFS in leakage between the analyses. In addition, please provide the results of the most recent tracer gas test performed on February 8, 2016, showing how the CREFS parameters presented in the LAR, are in alignment with this test.

RAI No. 6 (LOCA)

Regulatory Basis numbered 1, 2, 7, 8, and 10 apply to RAI No. 6.

RG 1.183 Appendix A regulatory position 3.3 states:

> 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). The simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3).

In the LAR, Table B of Enclosure 5 (page E5-32) states FNP's conformance with RG 1.183 Appendix A. Table B states that FNP's analysis for RG 1.183 regulatory position 3.3 is, "Conforms – Containment Spray is credited for elemental and particulate iodine removal." Enclosure 6 (page E6-5), "Loss of Coolant Accident Analysis," states that the containment spray elemental iodine removal coefficient is 13.7 hours^{-1}. Enclosure 12 (page E12-3) provides comparison tables of the new AST values compared to the current licensing basis values. Table 2 of Enclosure 12 states the LOCA inputs and assumptions; it states that the elemental iodine spray removal coefficient in the current licensing basis assumes 2.7 hours^{-1} and that the new AST assumes 13.7 hours^{-1}.

Please explain how the elemental iodine spray removal coefficient was calculated and discuss consistency with RG 1.183. Provide enough detail to allow the NRC staff to confirm the methodology is consistent with NUREG-0800 Chapter 6.5.2 and/or NUREG/CR-5966 as applicable.

RAI No. 7 (LOCA)

Regulatory Basis numbered 1, 2, 7, 8, and 9 above apply to RAI No. 7.

In the LAR, Section 3.3 of Enclosure 1, states that for the LOCA the control room normal flow rate is 2340 cubic feet per minute for less than 60 seconds. However, the timing associated with the control room isolation is not discussed. Enclosure 6 for the LOCA analysis, states that the control room pressurization mode initiation is automatic at 60 seconds; however, it doesn't provide any more information about this statement.

Please discuss the timing associated with the control room emergency filtration/pressurization system. Include in the discussion at what time after the event the detector will generate, the containment isolation actuation signal (CIAS), how long it takes the instrumentation to process the signal, how long it takes the control room ventilation system to reposition to the isolated position and/or pressurization mode. In addition, provide a comparison to the current licensing basis assumption.

RAI No. 8 (Fuel Handling Accident (FHA))

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 8.

In the LAR, it is stated for the FHA in containment that radioactivity released through the personnel airlock (PAL) mixes in a portion of the auxiliary building on the same level as the control room, and that this mixing is assumed to be instantaneous.

Regulatory Guide (RG) 1.183 Appendix B regulatory position 4.3 states:

> The radioactivity release from the fuel pool should be assumed to be drawn into the ESF [engineered safety feature] 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.

RG 1.183 Appendix B regulatory position 5.5 states:

> Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a

> case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.

RG 1.183 regulatory position 5.1.2 states:

> 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.

The NRC staff position is stated in NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," summary of issue number 3. In part, RIS 2006-04 summary of issue number 3 states:

> When implementing an AST, some licensees have proposed that certain engineered safety features (ESF) ventilation systems not be credited as a mitigation feature in response to an accident. In some cases, the licensee's revised design basis analysis introduced the assumption that normal (non-ESF) ventilation systems are operating during all or part of an accident scenario. Such an assumption is inappropriate unless the non-ESF system meets certain qualities, attributes, and performance criteria as described in RG 1.183, Regulatory Positions 4.2.4 and 5.1.2. For example, credit for the operation of non-ESF ventilation systems should not be assumed unless they have a source of emergency power. In addition, the operation of ventilation systems establishes certain building or area pressures based upon their flowrates. These pressures affect leakage and infiltration rates which ultimately affect operator dose. Therefore, to credit the use of these systems, licensees should incorporate the systems into the ventilation filter testing program in Section 5 of the TS. In summary, use of non-ESF ventilation systems during a DBA should not be assumed unless the systems have emergency power and are part of the ventilation filter testing program in Section 5 of the TS.

Considering that the auxiliary building ventilation system does not meet RG 1.183 regulatory position 5.1.2, please demonstrate how mixing occurs without the auxiliary building ventilation system, the 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 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.

RAI No. 9 (FHA)

Regulatory Basis numbered 1, 2, 7, 8, and 9 apply to RAI No. 9.

In the LAR, Section 3.4 of Enclosure 1, states that for the FHA in containment that high radiation in the control room makeup air intake results in isolation of the control room. However, the timing associated with the control room isolation is not discussed. Enclosure 7 states that the control room isolation mode initiation is automatic at 60 seconds, however it doesn't state if this assumption applies to the FHA in containment or the FHA in the spent fuel pool, or to both.

Please discuss the timing associated with the radiation detectors and the control room ventilation system. Include in the discussion at what time after the event the detector will generate the high radiation signal, how long it takes the instrumentation to process the signal, how long it takes the control room ventilation system to reposition to the isolated position and/or pressurization mode, and if the isolation occurs for both the FHA in containment and in the spent fuel pool. In addition, provide a comparison to the current licensing basis assumption.

RAI No. 10 (FHA)

Regulatory Basis numbered 1, 2, 7, 8, and 9 apply to RAI No. 10.

In the LAR, Table 3.6b of Enclosure 1, states that for the FHA that the control room emergency filtration/pressurization system (CREFS) recirculation charcoal filter efficiencies are 94.5% elemental and organic iodine. Enclosure 7, "Fuel Handling Accident Analysis," states that the CREFS recirculation charcoal filter efficiencies are 94.5% for all iodine species. These two Enclosures differ.

In addition, Tables 3.5a, 3.7a, 3.8a, 3.9a, and 3.10a of Enclosure 1, and Enclosures 6, 8, 9, 10, and 11 state that for the LOCA, MSLB, SGTR, CREA, and LRA that the CREFS recirculation charcoal filter efficiencies are 94.5% elemental and organic iodine and 98.5% particulate iodine.

If appropriate, please correct Table 3.6b of Enclosure 1 and Enclosure 7 to be consistent with the CREFS recirculation charcoal filter efficiency design or provide a discussion for using the different efficiencies presented in Table 3.5b of Enclosure 1 and Enclosure 7.

RAI No. 11 (FHA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 11.

In the LAR, Table 3.6a of Enclosure 1 states the parameters and assumptions for the FHA analysis and Enclosure 7 provides the FHA analysis. Table 3.6a of Enclosure 1 and Enclosure 7 provide the fraction of fission product inventory in the gap as:

I-131	0.08
Kr-85	0.10
Other Noble gases	0.05
Other Halogens	0.05

RG 1.183 states that 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 and that the release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. The release fractions listed in RG 1.183 Table 3 are not applicable to cores containing mixed oxide fuel and have been determined to be acceptable for use with currently approved light water reactor (LWR) fuel with a peak burnup up to 62,000 megawatt days (MWD) per metric ton of uranium (MTU) provided that the maximum linear heat generation

rate does not exceed 6.3 kilowatts (kw) per foot (ft) peak rod average power for burnups exceeding 54 gigawatt day (GWD)/MTU. The group and fractions in RG 1.183 Table 3 are:

I-131	0.08
Kr-85	0.10
Other Noble gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

RG 1.183 Appendix B regulatory position 3 states, "...Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor)."

The fraction of fission product inventory in the gap stated in Table 3.6a of Enclosure 1 and Enclosure 7 deviates from that stated in RG 1.183 Table 3 because the alkali metal gap fraction has been excluded. In addition, Section 3.4 and Table 3.6a of Enclosure 1 and Enclosure 7 do not address the fact that the fraction of fission product inventory in the gap is 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.

The NRC staff notes that the assumption that alkali metals are not released from the fission product gap does not appear to be accurate and is not equivalent to the assumption that 12% alkali metals are released from the fission product gap and are retained in the spent fuel pool or reactor cavity water.

Please explain the deviation from RG 1.183 stated above or provide analysis that conforms to RG 1.183. In addition, confirm that for the current approved LWR fuel that:

- The core does not contain any mixed oxide fuel.
- _Each rod_ does not exceed a peak burnup up of 62 GWD/MTU.
- For _each rod_ burnup that exceeds 54 GWD/MTU that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power.

RAI No. 12 (FHA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 12.

In the LAR, Tables 2 and 3 of Enclosure 7 lists the source term for the FHA. This list includes the following isotopes:

- Kr-85
- Xe-131m, Xe-133, Xe-133m, Xe-135
- I-131, I-132, I-133, I-135

Table 2 of Enclosure 6 lists the source term for the LOCA. This list includes the following isotopes:

- Kr-83m, Kr-85, Kr-85m, Kr-87, Kr-88
- Xe-131m, Xe-133, Xe-133m, Xe-135, Xe-135m, Xe-138
- I-130, I-131, I-132, I-133, I-134, I-135
- Br-82, Br-83, Br-84

- Cs-134, Cs-134m, Cs-136, Cs-137, Cs-138
- Rb-86, Rb-89

RG 1.183 Appendix B regulatory position 1.2 states:

> 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.

The source term provided in Tables 2 and 3 of Enclosure 7 deviates from RG 1.183 Appendix B regulatory position 1.2 and excludes radionuclides that are listed in Table 2 of Enclosure 6. Please explain the deviation from the RG 1.183 or conform to RG 1.183.

RAI No. 13 (FHA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 13.

In the LAR, Table C of Enclosure 5 provides FNP's conformance with RG 1.183 Appendix B. Specifically FNP's analysis for RG 1.183 Appendix B regulatory position 4.2 states that the penetration room filter (PRF) system meets the position in RG 1.52 and is required to be in service prior to the movement of irradiated fuel in the auxiliary building. Enclosure 7, "Fuel Handling Accident Analysis," also indicates that the PRF is in service prior to the FHA occurring.

FNP UFSAR Section 15.4.5.2, "Analysis of Effects and Consequences," G states, in part, that:

> Iodine escaping from the spent-fuel pool will be detected in the pool sweep ductwork and an alarm signaled to the control room operator. The normal ventilation system will be isolated automatically and the activity will be exhausted through the penetration room filtration (PRF) system. Both of the 100% capacity PRF systems will receive an automatic start signal. A single PRF system is capable of meeting all requirements of the FHA analyses. In the event of low flow in the normal ventilation system, the PRF system will automatically start, thus assuring a negative pressure inside the fuel handling area. Charcoal filter efficiencies of 89.5% for elemental and organic iodine are assumed, which have been reduced by 0.5% for bypass leakage.

The FNP UFSAR and this LAR appear to be inconsistent.

Is SNC proposing to change the current licensing basis for the FHA in the spent fuel pool such that PRF system will be placed in service prior to moving irradiated fuel in the auxiliary building? If a change to the current licensing basis is requested, please explain in what procedure is the requirement to place the PRF system in service prior to moving irradiated fuel in the auxiliary building, located and how the procedure is controlled.

If a change to the current licensing basis is not requested, please discuss the timing associated with the radiation detectors and the PRF system. Include in the discussion at what time after the event the detector will generate the high radiation signal, how long it takes the instrumentation to process the signal, how long it takes the PRF system to reposition and/or start and the amount of time that is takes to establish the required negative pressure inside the fuel handling area.

RAI No. 14 (FHA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 14.

In the LAR, Table 3.6a, "Parameters and Assumption for the FHA," of Enclosure 1 states that the pool water depth is 23 feet and the overall effective decontamination factor (DF) for iodine is 200. Enclosure 7, "Fuel Handling Accident Analysis," states that the overlaying pool depth is 23 feet and the pool decontamination factor for elemental iodine is 500 and for organic iodine is 1.

RIS 2006-04 summary of issue number 8 states:

> Appendix B to RG 1.183, provides assumptions for evaluating the radiological consequences of a fuel handling accident. If the water depth above the damaged fuel is 23 feet or greater, Regulatory Position 2 states that "the decontamination factors for the elemental and organic [iodine] species are 500 and 1, respectively, giving an overall effective decontamination factor of 200." However, an overall DF of 200 is achieved when the DF for elemental iodine is 285, not 500.

Please provide the plant specific method used to determine the DF for elemental iodine or provide a correction to the LAR consistent with RIS 2006-04 issue number 8.

RAI No. 15 (FHA)

Regulatory Basis numbered 1, 2, 5, and 8 apply to RAI No. 15.

RG 1.183 Appendix B regulatory position 5.3 states:

> 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.

[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 adminstrative 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.

In the LAR, Table C of Enclosure 5 states FNP's conformance with RG 1.183 Appendix B. Table C states that FNP's analysis for RG 1.183 regulatory position 5.3, "Conforms - The FHA radiological release is over a two-hour period." However, the NRC staff could not find two evaluations for the FHA occurring in containment. The first evaluation missing is an evaluation of the FHA in containment with the PAL open and the equipment hatch closed. The second evaluation missing is an evaluation of the FHA in containment with the equipment hatch open and the PAL closed.

FNP TS 3.9.3 allows three different configurations during core alterations and during movement of irradiated fuel assemblies within containment. The configurations are: (1) equipment hatch open and capable of being closed and held in place by four bolts, and PAL with one or more door(s) closed, (2) equipment hatch closed and held in place by four bolts, and PAL open with one door in the air lock capable of being closed, and (3) equipment hatch open and capable of being closed and held in place by four bolts and PAL open with one door in air lock capable of being closed. SNC submitted a FHA in containment analysis that addresses the configuration of the equipment hatch open and the PAL open.

Consistent with the allowances of TS 3.9.3 please provide:

1. An evaluation that is consistent with RG 1.183 and meets the limits in RG 1.183, SRP 15.0.1 and 10 CFR 50.67 and evaluates the FHA in containment with the equipment hatch open and the PAL closed.

2. An evaluation that is consistent with RG 1.183 and meets the limits in RG 1.183, SRP 15.0.1 and 10 CFR 50.67 and evaluates the FHA in containment with the equipment hatch closed and the PAL open.

If it is SNC's intent to be able to move irradiated fuel assemblies within the containment with *only* the configuration of equipment hatch open and PAL open, please discuss TS 3.9.3 and limits to other configurations.

RAI No. 16 (FHA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 16.

In the LAR, Section 3.1.3 of Enclosure 1 states that the control room atmospheric dispersion 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. Table 3.4b of Enclosure 1 provides the x/Q values based on 4 ½ years of data from 1999 to 2003 which was used in the FHA. FNP current licensing basis (CLB) for the FHA in containment is described in license amendment numbers 178 and 171 (ADAMS Accession No. ML082730007). The CLB lists x/Q values for the control room from the release point PAL as:

Boundary	Time Interval	Personnel Air Locks x/Q Value (sec^2/m^3)
Control Room	0 – 30 seconds 30 seconds – 2 hours	5.06×10^{-3} 1.66×10^{-3}

The LAR states:

> 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...
> 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.

RG 1.183 regulatory position 5.1.2 states:

> 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.

The NRC staff position is stated in NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," summary of issue number 3. In part, RIS 2006-04 summary of issue number 3 states:

> When implementing an AST, some licensees have proposed that certain engineered safety features (ESF) ventilation systems not be credited as a mitigation feature in response to an accident. In some cases, the licensee's revised design basis analysis introduced the assumption that normal (non-ESF) ventilation systems are operating during all or part of an accident scenario. Such an assumption is inappropriate unless the non-ESF system meets certain qualities, attributes, and performance criteria as described in RG 1.183, Regulatory Positions 4.2.4 and 5.1.2. For example, credit for the operation of non-ESF ventilation systems should not be assumed unless they have a source of emergency power. In addition, the operation of ventilation systems establishes certain building or area pressures based upon their flowrates. These pressures affect leakage and infiltration rates which ultimately affect operator dose. Therefore, to credit the use of these systems, licensees should incorporate the systems into the ventilation filter testing program in Section 5 of the TS. In summary, use of non-ESF ventilation systems during a DBA should not be assumed unless the systems have emergency power and are part of the ventilation filter testing program in Section 5 of the TS.

In addition, Table A of Enclosure 5 states FNP's conformance with RG 1.183 Section C. Table A states that FNP's analysis for RG 1.183 regulatory position 5.1.2 is, "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."

The auxiliary building ventilation system appears to depart from acceptable methods in RG 1.183 regulatory position 5.1.2 in that a loss of offsite power is assumed concurrent with the start of the FHA in containment. Please describe how radioactivity released through the PAL to the auxiliary building will exit the auxiliary building through the vent stack. In addition, explain which x/Qs are being used for this release pathway, include whether they are new or part of the CLB.

RAI No. 17 (FHA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 17.

In the LAR, Enclosure 14, "Summary of Regulatory Commitments," states:

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.

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.

3. With the personnel airlock open during fuel handling operations or core alterations, the containment purge system will be in operation.

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.

5. In the event of an FHA, control room occupants will use the secondary door to the control room for ingress and egress.

SNC plans to establish administrative controls to ensure awareness of open containment penetration flow paths, and personnel will be designated and readily available to isolate any open containment penetration flow path. Please discuss plans to isolate the containment penetration flow paths immediately upon a detection of a FHA.

Please explain why SNC does 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.

RAI No. 18 (Main Steam Line Break (MSLB) Accident)

Regulatory Basis numbered 1, 2, 3, 7, 8, and 9 apply to RAI No. 18.

In the LAR, Section 3.5 of Enclosure 1, states that for the MSLB accident that the control room is automatically realigned into the emergency mode upon receipt of a safety injection signal. However, the timing associated with the control room isolation is not discussed. Enclosure 8 states that the control room pressurization mode is initiated at the start of the accident.

Please explain if it is assumed that the CREFS is operating in pressurization mode at the start of the accident, and discuss if CREFS is normally operated in pressurization mode. Please discuss the AST analysis regarding normal design and operation of the control room ventilation system.

Please discuss the timing associated with the CIAS and the CREFS. Include in the discussion at what time after the event the CIAS will be generated, how long it takes the instrumentation to process the signal, how long it takes the control room ventilation system to reposition to the isolated position and/or pressurization mode, and how long it takes to start the CREFS trains for pressurization. In addition, provide a comparison to the current licensing basis assumptions and update Table 3.7a of Enclosure 1 and Table 8 of Enclosure 8 to show the timing and atmospheric dispersion factors used for the normal control room intake prior to its operation in pressurization mode.

<u>RAI No. 19</u> (MSLB)

Regulatory Basis numbered 1 and 2 apply to RAI No. 19.

In the LAR, Table 3.7a of Enclosure 1 states the parameters and assumptions for the MSLB accident. Table 3.7a states the following:

Steam releases from Intact SG [steam generator] to environment

0 – 2 hours	316,715 lbm [pounds mass]
2 – 8 hours	703,687 lbm
8 – 24 hours	948,000 lbm

Steam mass released from faulted SG to the environment 439,145 lbm

Table 1 of Enclosure 8 provides the MSLB flow rates. Table 1 and its notes are as follows:

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	4.81 E+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:

- RCS Leakage of 1 gpm - Volumetric leakage (gpm) from RCS is divided by 7.48 gal/ft^3 [gallons per cubic foot].
- Feedwater - Mass release from feedwater to intact SGs is multiplied by 1.1. Flow is the release (lbm) multiplied by 453.6 g [grams]/lbm and divided by time duration (hr).
- 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).
- Faulted SG - Mass release from the faulted SG is multiplied by 1.1. Flow is conservatively high.

Table 15.4-23 in FNP UFSAR, Revision 21, May 2008, states the parameters used in steam line break analyses. Table 15.4-23 states:

Initial steam release from faulted steam generator (lb)(min)	461,000 (0-30)
Steam release from two intact steam generators (lb)(h)	333,000 (0-2) 739,000 (2-8) 995,000 (8-24)
Feedwater flow to two intact steam generators (lb)(h)	459,000 (0-2) 747,000 (2-8) 995,000 (8-24)

Table 3.7a of Enclosure 1 and Table 1 of Enclosure 8 are not consistent with Table 15.4-23 in FNP UFSAR, Revision 21, May 2008. Applying the notes stated in Table 1 of Enclosure 8 to the data stated in FNP UFSAR doesn't yield the results in Table 1 of Enclosure 8.

Please explain the differences between Table 3.7a of Enclosure 1 and Table 1 of Enclosure 8 and Table 15.4-23 in FNP UFSAR, Revision 21, May 2008.

RAI No. 20 (MSLB)

Regulatory Basis numbered 1 and 2 apply to RAI No. 20.

In the LAR, Table 5 of Enclosure 8 contains the initial alkali metal concentrations in the secondary system. However, it appears that there may be an error in the calculation of activity for rubidium 89 for the intact steam generators. Specifically, 2.0E-02 micro curies per gram (μCi/g) multiplied by 9.82E08 g multiplied by 1.0E-06 Ci/μCi equals 19.64 Ci, not 0.2 curies.

Please review the data in Table 5 of Enclosure 8 and provide an update to Table 5 and the resultant dose results, if necessary.

RAI No. 21 (Steam Generator Tube Rupture (SGTR) Accident)

Regulatory Basis numbered 1, 2, 3, 7, 8, and 9 apply to RAI No. 21.

LAR, Section 3.6 of Enclosure 1, states that for the SGTR accident that the control room is automatically realigned into the emergency ventilation mode upon receipt of a safety injection signal. However, the timing associated with the control room isolation is not discussed. Enclosure 9 states that the control room pressurization mode is initiated at the start of the accident.

Please explain if it is assumed that the CREFS is operating in pressurization mode at the start of the accident, and discuss if CREFS is normally operated in pressurization mode. Please discuss the AST analysis normal design and operation of the control room ventilation system.

Please discuss the timing associated with the safety injection signal and the CREFS. Include in the discussion at what time after the event the safety injection signal will be generated, how long it takes the instrumentation to process the signal, how long it takes the control room ventilation system to reposition to the isolated position and/or pressurization mode, and how long it takes to start the CREFS trains for pressurization. In addition, provide a comparison to the current licensing basis assumptions and update Table 3.8a of Enclosure 1 and Table 8 of Enclosure 9 to show the timing and atmospheric dispersion factors used for the normal control room intake prior to its operation in pressurization mode.

RAI No. 22 (Control Rod Ejection Accident (CREA))

Regulatory Basis numbered 1, 2, 7, 8, and 11 apply to RAI No. 22.

In the LAR, Section 3.7 of Enclosure 1 states that for the CREA that credit is taken for natural deposition of aerosols in containment. Enclosure 12 provides comparison tables of the new AST values compared to the current licensing basis values. Table 6 states the CREA inputs and assumptions; it states that for natural deposition in containment the current licensing basis assumes 50% plateout of the reactor coolant system release and that the new AST assumes an aerosol removal rate of 2.74E-2 hr^{-1} and no removal of elemental iodine. Table 6 states that the reason for the change is natural deposition is credited per RG 1.183 Appendix H Section 6.1.

RG 1.183 Appendix H regulatory position 6.1 states that a reduction in the amount of radioactive material available for leakage from containment that is due to natural deposition may be taken into account and it refers to RG 1.183 Appendix A for guidance on acceptable methods and assumptions for evaluating this mechanism. RG 1.183 Appendix A states that reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited and that acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2 of NUREG-0800 and in NUREG/CR-6189. Enclosure 10, "Control Rod Ejection Accident Analysis," states that credit is taken for natural deposition in containment is per NUREG/CR-6189 (Table 36).

Table 36 of NUREG/CR-6189 lists five specific time intervals and their correlations. However, Enclosure 10 lists one value for natural deposition and doesn't provide enough information for the NRC staff to determine that this value reflects a methodology consistent with NUREG/CR- 6189.

Please provide a summary of the methodology used in enough detail to allow the NRC staff to determine consistency with NUREG/CR-6189.

RAI No. 23 (CREA)

Regulatory Basis numbered 1, 2, 3, 7, 8, and 9 apply to RAI No. 23.

In the LAR, Section 3.7 of Enclosure 1, states that for the CREA accident that the control room ventilation system is automatically realigned into the emergency ventilation mode upon receipt of a safety injection signal. However, the timing associated with the control room isolation is not discussed. Enclosure 10 states that the control room pressurization mode is initiated at the start of the accident.

Please explain if it is assumed that the CREFS is operating in pressurization mode at the start of the accident, and discuss if CREFS is normally operated in pressurization mode. Please discuss the AST analysis normal design and operation of the control room ventilation system.

Please discuss the timing associated with the CIAS and the CREFS. Include in the discussion at what time after the event the CIAS will be generated, how long it takes the instrumentation to process the signal, how long it takes the control room ventilation system to reposition to the isolated position and/or pressurization mode, and how long it takes to start the CREFS trains for pressurization. In addition, provide a comparison to the current licensing basis assumptions and update Table 3.9a of Enclosure 1 and Table 6 of Enclosure 10 to show the timing and atmospheric dispersion factors used for the normal control room intake prior to its operation in pressurization mode.

RAI No. 24 (CREA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 24.

RG 1.183 Appendix H, "Assumptions for Evaluating the Radiological Consequences of a pressurized water reactor (PWR) Rod Ejection Accident," regulatory position 4 states:

> 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.

In the LAR, Table G in Enclosure 5, Section H-4, states FNP's conformance with RG 1.183 Appendix H. Table G states that FNP's analysis for RG 1.183 regulatory position 4 is:

> 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.

Because containment sprays are not actuated and no credit is taken for the containment sump pH being controlled at values of 7 or greater, the iodine species needs to be evaluated on a plant specific basis to determine that 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide is a conservative assumption.

Provide the plant specific evaluation that determined that the chemical form of radioiodine released to the containment atmosphere of 95% CsI, 4.85% elemental iodine, and 0.15% organic iodide is conservative at FNP and that the iodine does not re-evolve. 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.

RAI No. 25 (CREA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 25.

RG 1.183 Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," regulatory position 5 states:

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

In the LAR, Table G in Enclosure 5, Section H-5, states FNP's conformance with RG 1.183 Appendix H. Table G states that FNP's analysis for RG 1.183 regulatory position 5 is:

> 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.

Using the containment iodine species specified in RG 1.183 Appendix H regulatory position 4 instead of the steam generator iodine species specified in RG 1.183 Appendix H regulatory position 5 is a deviation from the RG 1.183 Appendix H. This deviation does not result in a conservative postulated dose. NUREG-1465 section 3.5, "Chemical form," states:

> In an aqueous environment, as expected for LWRs [light water reactors], iodine is expected to dissolve in water pools or plate out on wet surfaces in ionic form as I^-. Subsequently, iodine behavior within containment depends on the time and pH of the water solutions. Because of the presence of other dissolved fission products, radiolysis is expected to occur and lower the pH of the water pools. Without any pH control, the results indicate that large fractions of the dissolved iodine will be converted to elemental iodine and be released to the containment atmosphere.

Provide the plant specific evaluation that determined that the chemical form of radioiodine released <u>from the steam generators</u> of 95% CsI, 4.85% elemental iodine, and 0.15% organic iodide is conservative and show that the iodine does not re-evolve. 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.

<u>RAI No. 26</u> (CREA)

Regulatory Basis numbered 1 and 2 apply to RAI No. 26.

In the LAR, Table 2 of Enclosure 10 contains the containment and reactor coolant system activities for the control rod ejection accident. However, it appears that there may be an error in the calculation for the gap release activity for the alkali metals, specifically the rubidium and cesium isotopes experiencing cladding failure. Page E10-4 of Enclosure 10 states that the fractions of fission product inventory contained within the fuel rod gap for alkali metals is 0.12. However, Table 2 lists the alkali metal gap fraction to be 0.05 and therefore the calculated activity for the gap release activity is lower than expected.

Please review the data in Table 2 and provide an update to Table 2, as necessary. In addition, provide the updated dose results that correspond with this change or explain the deviation from RG 1.183.

<u>RAI No. 27</u> (CREA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 27.

RG 1.183 Section C, Table 6, states the analysis release duration for each accident and/or case, in addition to the exclusion area boundary and low population zone dose criteria; it states that the release duration is until cold shutdown is established for the secondary pathway for the pressurized water reactor rod ejection accident.

RG 1.183 Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," regulatory position 7.1 states:

> 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.

In the LAR, Section 3.7, "Control Rod Ejection Accident," states in part that, for releases from the secondary system that the primary to secondary leakage duration is 2500 seconds and in keeping with previous evaluations of the CREA, the secondary system mass releases to the environment last for 98 seconds.

The NRC staff agrees that this assumption is consistent with FNP CLB as stated in UFSAR 15.4.6.4.3, "Assumptions for Regulatory Guide 1.77 analysis." However, the assumptions in RG 1.77 differ from those in RG 1.183. RG 1.183 assumes that the primary to secondary leakage continues until shutdown cooling is in operation and releases from the steam generators have been terminated.

Please discuss the CREA release from the secondary system accident analysis and accident response, explain how it is consistent with RG 1.183, Appendix H regulatory position 7.1 stated above or explain why the deviation from RG 1.183 is conservative with respect to the assumptions and resultant radiological doses.

RAI No. 28 (CREA)

Regulatory Basis numbered 1, 2, 7, 8, and 10 apply to RAI No. 28.

In the LAR, Section 3.7 of Enclosure 1 states that for the CREA no credit is taken for removal by containment sprays or for deposition of elemental iodine on containment surfaces. Table G of Enclosure 5 states FNP's conformance with RG 1.183 Appendix H. Table G states that FNP's analysis for RG 1.183 Appendix H (page E5-52) regulatory position 6.1 is, "Conforms – Radioactive material removal from the containment atmosphere by sprays and other engineered safety features is not credited..." Enclosure 12 provides comparison tables of the new AST values compared to the current licensing basis values. Table 6, CREA inputs and assumptions; states that for iodine/particulate removal by containment sprays the current licensing basis assumes none, and that the new AST assumes an iodine/particulate removal by containment sprays removal rate of 5.0 hr^{-1}. Table 6 states that the reason for the change is "Particulate removal by containment sprays is per RG 1.183 Appendix H Section 6.1."

RG 1.183 Appendix H regulatory position 6.1 states that a reduction in the amount of radioactive material available for leakage from containment that is due to containment sprays may be taken into account and refers to RG 1.183 Appendix A for guidance on acceptable methods and assumptions for evaluating this mechanism. RG 1.183 Appendix A states that 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 may be credited and that acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and in NUREG/CR-5966.

Enclosure 12 doesn't provide sufficient information for the NRC staff to determine that the iodine and/or particulate removal by containment sprays methodology is consistent with NUREG/CR-5966. In addition, Enclosure 12 is in conflict with Enclosure 1 and Enclosure 5.

Please explain if SNC is proposing to credit iodine and/or aerosol remove by the containment spray system during a CREA. If SNC is proposing to credit iodine and/or aerosol removal by the containment spray system during a CREA, then provide a summary of the methodology used such that the NRC staff can determine consistency with NUREG/CR-5966.

RAI No. 29 (CREA)

Regulatory Basis numbered 1, 2, 7, 8, and 11 apply to RAI No. 29.

In the LAR, Section 3.7 of Enclosure 1 states that for the CREA that *no* credit is taken for removal by deposition of elemental iodine on containment surfaces. Table G of Enclosure 5 states FNP's conformance with RG 1.183 Appendix H. Table G states that FNP's analysis for RG 1.183 Appendix H regulatory position 6.1 is, "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." Enclosure 12 provides comparison tables of the new AST values compared to the current licensing basis values. Table 6 states the CREA inputs and assumptions; it states that for natural deposition in containment the current licensing basis assumes 50% plateout of the reactor coolant system release and that the new AST assumes an aerosol removal rate of 2.74E-2 hr^{-1} and *no* removal of elemental iodine. Table 6 states that the reason for the change is "Natural deposition is credited per RG 1.183 Appendix H Section 6.1."

RG 1.183 Appendix H regulatory position 6.1 states that a reduction in the amount of radioactive material available for leakage from containment that is due to natural deposition may be taken into account and it refers to RG 1.183 Appendix A for guidance on acceptable methods and assumptions for evaluating this mechanism. RG 1.183 Appendix A states that reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited and that acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2 of NUREG-0800 and in NUREG/CR-6189. SNC's discussion of the removal of elemental iodine by natural deposition in Section 3.7 of Enclosure 1 and Table 6 of Enclosure 12 differs from that stated in Table G of Enclosure 5.

Please explain if SNC is proposing to credit removal of elemental iodine by natural deposition during a CREA. If SNC is proposing to credit removal of elemental iodine by natural deposition during a CREA then please provide a summary of the methodology used in enough detail that the NRC staff can determine consistency with NUREG/CR-6189.

RAI No. 30 (CREA)

Regulatory Basis numbered 1, 2, 3, 7, and 8 apply to RAI No. 30.

In the LAR, Section 3.7 of Enclosure 1 and Enclosure 10 discuss the CREA analysis and two separate release pathways are evaluated. The radiological dose results provided are 3.8 rem at the exclusion area boundary (EAB), 2.7 rem at the low population zone (LPZ), and 3.7 rem in the control room. However, it does not state if these results are specific to the containment release pathway or the secondary release pathway.

Please provide the resultant radiological doses at the EAB, LPZ, and control room for each CREA pathway analyzed.

<u>RAI No. 31</u> (CREA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 31.

RG 1.183 Section C, regulatory position 3.1 states:

> ...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.

In the LAR, Table 2 of Enclosure 10 contains the containment and reactor coolant system activities for the CREA. Page E10-7 of Enclosure 10 states that the fuel melt activity is the product of core activity in column 2, margin factor in column 3, core release in column 4 and fuel melt fraction of 0.0025. However, it appears to be contrary to RG 1.183 regulatory position 3.1 in that there the calculation for the fuel melt activity does not take into account the radial peaking factor. In addition, the calculation double counts the gap release from cladding failure because it fails to remove the percentage of cladding failure. An example of a typical calculation for fuel melt activity would be the product of core activity, radial peaking factor, any margin factor used, percentage of fuel cladding failure, percentage of fuel melt, and fuel melt release fraction; therefore I-131 fuel melt activity would be equal to 1.63E+04 (7.50E+07 * 1.7 * 1.02 * 0.1 * 0.0025 * 0.5)

Please review the data in Table 2 of Enclosure 10 and provide an update to Table 2 as necessary. In addition, provide the updated dose results that correspond with this change or please explain the deviation from RG 1.183.

<u>RAI No. 32</u> (Locked Rotor Accident (LRA))

Regulatory Basis numbered 1, 2, 3, 7, 8, and 9 apply to RAI No. 31.

In the LAR, Section 3.8 of Enclosure 1, states that the LRA analysis assumes that the control room isolates and enters the emergency ventilation mode at the onset of the accident and that for conservatism, an assessment is being performed for a delayed manual CREFS initiation. However, the timing associated with the control room isolation is not discussed nor is the manual isolation time. Enclosure 11 states that the control room pressurization mode is initiated at the start of the accident and that it is under reassessment for delayed manual start.

Please explain if it is assumed that the CREFS is operating in pressurization mode at the start of the accident, and discuss if CREFS is normally operated in pressurization mode. Please discuss the AST analysis normal design and operation of the control room ventilation system.

Please discuss the timing associated with the CIAS and the CREFS. Include in the discussion at what time after the event the CIAS will be generated, how long it takes the instrumentation to process the signal, how long it takes the control room ventilation system to reposition to the isolated position and/or pressurization mode, and how long it takes to start the CREFS trains for pressurization. In addition, provide a comparison to the current licensing basis assumptions and update Table 3.10a of Enclosure 1 and Table 6 of Enclosure 11 to show the timing and atmospheric dispersion factors used for the normal control room intake prior to its operation in pressurization mode.

Or provide the assessment of the delayed manual start, including an update Table 3.10a of Enclosure 1 and Table 6 of Enclosure 11 to show the timing and atmospheric dispersion factors used for the normal control room intake prior to its operation in pressurization mode. In addition using NUREG-0711, Human Factors Engineering Program Review Model, include the human factors assessment of the delayed manual start.

RAI No. 33 (LRA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 33.

RG 1.183 Section C, Table 6, states the analysis release duration for each accident and/or case, in addition to the exclusion area boundary and low population zone dose criteria; it states that the release duration is until cold shutdown is established for the pressurized water reactor locked rotor accident.

In the LAR, Section 3.8 of Enclosure 1, the discussion starts by stating that the LRA continues for 8 hours by which time the reactor coolant system temperature is cooled to cold shutdown condition, however it concludes by stating that for the LRA that the duration is 30 days for the containment pathway, and until cold shutdown is established for the secondary pathway. This conclusion is contrary to the LRA duration stated in RG 1.183 Section C Table 6.

Please explain if the 30 day duration for the containment pathway is in error. If not, please provide the analysis of the containment pathway.

RAI No. 34 (LRA)

Regulatory Basis numbered 1 and 2 apply to RAI No. 34.

In the LAR, Table 3.10a of Enclosure 1 states the parameters and assumptions for the LRA. Table 3.10a states the following:

Steam releases from SG to environment	
0 – 2 hours	512,325 lbm
2 – 8 hours	833,221 lbm

Table 1 of Enclosure 11 provides the LRA flow rates. Table 1 and its notes are as follows:

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/ft^3.
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).

Table 15.4-25A in FNP UFSAR revision 21 5/08 states the parameters used in reactor coolant pump locked rotor analyses. Table 15.4-25A states:

Steam release from three steam generators (lbs)	538,000 (0-2 h) 875,000 (2-8 h)
Feedwater flow to three steam generators (lbs)	728,000 (0-2 h) 887,000 (2-8 h)

Table 3.10a of Enclosure 1 and Table 1 of Enclosure 11 are not consistent with Table 15.4-25A in FNP UFSAR, Revision 21, May 2008.. Applying the notes stated in Table 1 of Enclosure 11 to the data stated in FNP UFSAR doesn't yield the results in Table 1 of Enclosure 11.

Please explain the differences between Table 3.10a of Enclosure 1 and Table 1 of Enclosure 11 and Table 15.4-25A in FNP UFSAR, Revision 21, May 2008.

<u>RAI No. 35</u> (LRA)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 35.

In the LAR, Table 3.10a of Enclosure 1 states the parameters and assumptions for the LRA. Enclosure 11 contains the LRA analysis. Table 3.10a of Enclosure 1 states that the fraction of fission product inventory in the gap is:

- I-131 0.08
- Kr-85 0.10
- Other Halogens and Noble gases 0.05
- Alkali Metals 0.12

Enclosure 11 states that the fraction of fission product inventory contained in the fuel rod gap is:

• I-131	0.08
• Kr-85	0.10
• Other Halogens and Noble gases	0.05
• Alkali Metals	0.12

These gap fractions are consistent with RG 1.183 Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap." However, Table 2 in Enclosure 11 the gap release activity calculated for the isotopes of Bromine uses the gap fraction of 0.12. This conflicts with the other halogen gap fraction stated in both enclosures, and is not consistent with RG 1.183.

Please either correct this inconsistency and provide the updated dose results or explain the deviation from RG 1.183.

RAI No. 36 (LRA)

Regulatory Basis numbered 1, 2, 3, 7, 8, 9, and 12 apply to RAI No. 36.

FNP, "General Design Criteria for Nuclear Power Plants," Criterion 19, "Control Room," states, in part:

> A control room is provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Section 10 CFR 50.67, "Accident Source Term," (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) 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) TEDE for the duration of the accident.

NUREG-0800, SRP 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms," Table 1 lists the accident dose criteria. RG 1.183 Table 6 lists the accident dose criteria. Both SRP 15.0.1 and RG 1.183 tables place a lower limit on the EAB and LPZ dose criteria for the pressurized water reactor locked rotor accident of 2.5 rem.

In the LAR, Enclosure 11 contains the LRA analysis and states that the resultant dose in the control room is less than 5 rem TEDE. In addition, it states that a reassessment is being performed assuming a delayed manual CREFS initiation and that the results of the reassessment are expected to remain less than 5 rem TEDE and be non-limiting.

Provide the resultant dose in the control room dose that is postulated to occur during a locked rotor accident. Explain if a reassessment was performed, provide the methodologies, assumptions and inputs, and resultant radiological doses for the EAB, LPZ, and control room in enough detail to allow the NRC staff to be able to perform an independent assessment of the results.

RAI No. 37 (TSTF-448)

Regulatory Basis numbered 2, 4, 5, 6, and 13 apply to RAI No. 37.

TSTF-448, Revision 3, "Control Room Habitability," modifies Condition B and Condition F of Standard Technical Specifications (STS) Limiting Conditions of Operation (LCO) 3.7.10, "Control Room Emergency Filtration System (CREFS)," in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," to state:

Condition B	One or more CREFS trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4.
Condition F	Two CREFS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.

SNC has chosen to deviate from TSTF-448 by proposing to revise Condition B in FNP Technical Specification 3.7.10 so that it includes all modes of applicability instead of just Modes 1, 2, 3, and 4 and by not revising Condition D. Proposed Condition B will state:

One or more CREFS trains inoperable due to inoperable CRE boundary.

Current Condition D states:

Two CREFS trains inoperable in MODE 1, 2, 3, OR 4.

These two deviations from TSTF-448 cause an apparent conflict in FNP Technical Specification 3.7.10, such that two conditions are required to be entered anytime the following exist:

- Two CREFS trains are inoperable because of CRE boundary while in Modes 1, 2, 3, or 4. This inoperability requires entry into Conditions B and D.
- Two CREFS trains are inoperable because of CRE boundary during core alterations. This inoperability requires entry into Conditions B and F.
- Two CREFS trains are inoperable because of CRE boundary during movement of irradiated fuel assemblies. This inoperability requires entry into Conditions B and F.

In addition, in letter dated June 18, 2009 (ADAMS Accession No. ML091690643), the Technical Specifications Task Force (TSTF) submitted Traveler TSTF-508, Revision 1, "Revise Control Room Habitability Actions to Address Lessons Learned from TSTF-448 Implementation," to the NRC staff for review and approval. TSTF-508 proposed the extension of the use of mitigating actions to Modes 5, 6, and during movement of recently irradiated fuel assemblies when one or more CREFS trains is inoperable due to an inoperable CRE boundary in Westinghouse STS 3.7.10, just as SNC has requested through the deviation in this LAR. During the review of TSTF-508, the NRC staff requested additional information (ADAMS Accession No. ML110890817). This position is applicable to this change requested by SNC and is as follows. The NRC staff expressed their view that the extension of the use of mitigating actions to Modes 5, 6, and during movement of recently irradiated fuel assemblies is not adequately justified and is not warranted for the following reasons:

- The regulation at Subpart H of 10 CFR Part 20, "Standards for Protection against Radiation," provides the requirements for respiratory protections and controls to restrict internal exposure in restricted areas. Specifically, 10 CFR 20.1701 states that licenses shall use, to the extent practicable, process or engineering controls to control the concentration of radioactivity in the air. Use of other controls as described in 10 CFR 20.1702 is only allowed by regulation when it is not practicable to apply process or other engineering controls.

- NEI 99-03, Appendix F, "Compensatory Measures Allowable On An Interim Basis," Page F-1, states:

    > The use of SCBA [self-contained breathing apparatus] and KI [potassium iodide] has been determined to be acceptable for addressing control room envelope integrity in the interim situation until the licensee remediates the issue. However, use of SCBA or KI in the mitigation of situations where in-leakage does not meet design basis limits is not acceptable as a permanent solution. <u>10 CFR 20.1701 essentially says that engineering/process controls shall be used to the extent practical. If not practical, then 10 CFR 20.1702 methods should be used. Therefore, the use of SCBAs should be a last resort.</u> [emphasis added]

- The use of KI and SCBA is not without risk. The allowance to use KI and SCBA was not previously extended to Modes 5 and 6 because another practical control (stopping fuel movement) existed. The NRC staff does not believe that the proposed compensatory measures are appropriate given that the process control of stopping fuel movement is available.

Explain how the regulations at 10 CFR 20.1701 and 10 CFR 20.1702 and the regulatory guidance in NUREG-1431 STS 3.7.10 are met or how your proposal meets the intent of these regulations and regulatory guidance. Please justify the deviation from TSTF-448 or remove the extension of the use of mitigating actions to Modes 5, 6, during core alterations, and during movement of irradiated fuel assemblies.

<u>RAI No. 38</u> (RG 1.183)

Regulatory Basis numbered 1, 2, 7, and 8 apply to RAI No. 38.

In the LAR, Table A of Enclosure 5 for RG 1.183 regulatory position 4.3 states:

> Not Applicable – This full scope AST implementation LAR is for the radiological consequences of major FNP DBAs.

RG 1.183, Regulatory Position 4.3, "Other Dose Consequences," states that:

> 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. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of [total effective dose equivalent] effective TEDE.

Please provide additional information describing how RG 1.183, Regulatory Position 4.3 has been assessed for FNP.

RAI No. 39 (RG 1.183)

Regulatory Basis numbered 1, 2, 3, 8, and 9 apply to RAI No. 39.

In the LAR, Table A of Enclosure 5 states FNP's conformance with RG 1.183 Section C. Table A states that FNP analysis for RG 1.183 regulatory position 4.2.1 is:

> 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 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 (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.

The NRC staff understands from the discussion presented above that the resultant dose will not be significant from these paths, but RG 1.183 considers all sources of radiation that causes exposure to control room personnel.

Please provide the radiation dose results calculated from the following pathways and the final resultant dose to the control room from all sources of radiation during LOCA or provide a technical basis for deviating from RG 1.183.

- 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.

<u>RAI No. 40</u> (RG 1.183)

Regulatory Basis numbered 1, 2, 3, 7, 8, and 9 apply to RAI No. 40.

Section 10 CFR 50.67(b)(2) requires that the licensee's analysis demonstrates with reasonable assurance that *<u>adequate radiation protection is provided to permit access to</u>* and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

In the LAR, SNC has provided the resultant radiation dose associated with occupancy of the control room. However, the LAR appears to be missing a discussion and/or calculation that accounts for the control room personnel radiation exposure received, for the duration of the accident, upon ingress/egress from the site boundary to the control room. In order to meet 10 CFR 50.67 and 10 CFR 50 Appendix A GDC 19, the radiation dose for accessing the control room must be evaluated from the site boundary to the control room for both ingress and egress for the duration of the accident.

Please provide an analysis of the radiation dose received from ingress and egress to the control room in enough detail that will enable the NRC staff to be able to perform an independent calculation.

<u>RAI No. 41</u> (RG 1.183)

Regulatory Basis numbered 1, 2, and 8 apply to RAI No. 41.

In the LAR, Table 1 in Enclosure 6 lists the core cycle-to-cycle augments. Table 1 of Enclosure 6 states:

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

Enclosure 10 provides the CREA analysis and Enclosure 11 provides the LRA analysis. Enclosure 10 and Enclosure 11 state that core fission product inventories are taken from an equilibrium cycle based upon a power level of 2831 MWt and 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

Please explain why there is a difference between the factor 1.03 for halogen and other noble gases stated in Enclosure 6 and factor 1.02 for iodine isotopes and other noble gases stated in Enclosures 10 and 11. Explain why they are not the same. If they are supposed to be the same, then please provide an update to the respective enclosure(s) and update the resultant radiological doses.

RAI No. 42

As described in NUREG-0800, "Standard Review Plan," Chapter 15.0.1, Rev. 0, "Radiological Consequence Analyses Using Alternative Source Terms," part of the human factors review consists of reviewing issues related to emergency operating procedures (EOPs).

In the license amendment request dated November 22, 2016, FNP does not appear to address EOPs. Please describe, if applicable, whether FNP will be updating any EOPs and, if so, describe any operator training associated with those updates.

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION (CAC NOS. MF8861, MF8862, MF8916, MF8917, MF8918, MF8919) DATED MARCH 24, 2017

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