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

Joseph M. Farley Nuclear Plant Units 1 and 2
Response to 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 (SNC) requested Nuclear Regulatory Commission (NRC) review and approval of proposed revisions to the licensing basis of FNP that support a full scope application of an Alternative Source Term (AST) methodology. Proposed TS changes, which are supported by the AST Design Basis Accident radiological consequence analyses, were included in the license amendment request (LAR). In addition, the proposed amendment incorporated 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.

The NRC staff is reviewing the submittal. By letter dated March 24, 2017, the NRC informed SNC that NRC has determined that additional information is needed to complete its review. The Enclosure provides the SNC response to the additional information to 39 of the 42 questions requested by the NRC staff. The remaining three responses will be provided by June 7, 2017.

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

Mr. Justin T. Wheat states that he is the Nuclear Licensing Manager of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

Justin T. Wheat
Nuclear Licensing Manager

JTW/CBM/LC

Sworn to and subscribed before me this 23 *day of* May, *2017.*

Notary Public

My commission expires: 10-8-2017

Enclosures:
1. SNC Response to Request for Additional Information

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
Alternative Source Term License Amendment Request

Enclosure 1

SNC Response to Request for Additional Information

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.

SNC Response to RAI No. 1:

The Plant Farley control room ventilation system achieves pressurization mode within 60 seconds following the onset of the accident. During this process, the supply of outside air into the control room is automatically realigned from the normal intake on the east side of the control room to the emergency intakes on the west side. In general, the atmospheric dispersion factors (X/Qs) for the normal intake are higher than those for the emergency intakes for this event since the release points of interest are also located on the east side of the control room. For the containment leakage and containment purge contributions to the total LOCA dose, the X/Q inputs to the RADTRAD models reflect this shift in the intake location because nuclide releases occur coincident with the beginning of the event. However, for the Emergency Safety Features (ESF) leakage and RWST release contributors, the nuclide releases begin 20 minutes after the start of the event. At this time, the control room ventilation system is fully established in the pressurization mode with the outside air being supplied through the emergency intakes. Since no activity has been released to the environment for these pathways while the normal intake is in service, the RADTRAD input is simplified to only apply the emergency intake X/Qs.

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.

SNC Response to RAI No. 2:

Section 3.2 of Appendix A to Reg. Guide 1.183 allows for reduction in airborne radioactivity in containment by natural deposition using the guidance provided in NUREG-0800, Section 6.5.2 and NUREG/CR-6189. Section III.4.C.i of NUREG-0800 (Section 6.5.2) provides the following equation for determining the wall deposition of **elemental** iodine:

$$\lambda_w = \frac{K_w A}{V}$$

Where:
λ_w = removal rate coefficient for wall deposition
K_w = mass transfer coefficient (4.9 m/hr per Section III.4.C.i of NUREG-0800, Section 6.5.2)

A = surface area for wall deposition
V = containment building net free volume

The total surface area in containment credited for deposition is 363,200 ft^2. Therefore, using the total containment volume of 2,030,000 ft^3, the wall deposition rate is calculated as:

$$\lambda_W = \frac{(4.9\ m/hr)(3.281\ ft/m)(363{,}200\ ft^2)}{2{,}030{,}000\ ft^3} = 2.88\ hr^{-1}$$

Note that this value is calculated using inputs corresponding to the total containment volume. Since this total volume is represented using two separate compartments in the RADTRAD model (sprayed and unsprayed), this single value is applied in both compartments.

However, while wall deposition of **elemental** iodine is credited throughout the event, **aerosol** deposition is modeled only in unsprayed regions of containment. This is done because it is conservatively assumed that aerosols tend to be retained by the liquids in containment. As such, the particulates can be easily washed from deposited surfaces, carried to the containment sump, and returned to the containment atmosphere via containment sprays. Consequently, **aerosol** deposition is only credited in the unsprayed RADTRAD compartment at the start of the event, and in the sprayed compartment only after sprays have been terminated at 8 hours. An **aerosol** natural deposition rate of 0.1 hr^{-1} is applied based upon values presented in Section VI NUREG/CR-6189.

No credit is taken for wall deposition of **organic** iodine at any time in this analysis.

<u>RAI No. 3</u> (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.

<u>SNC Response to RAI No. 3:</u>

The Farley LOCA analysis considers four independent release pathways:

1.) Primary Containment Leakage – Inputs and methods applied to this release pathway follow the guidance of Section 3 of Appendix A to RG 1.183.

2.) ESF Leakage - Inputs and methods applied to this release pathway follow the guidance of Section 5 of Appendix A to RG 1.183.

3.) Containment Purge – This pathway accounts for the potential for routine containment purging during power operations as identified in Item 3.8 of Appendix A to RG 1.183.

4.) RWST Backleakage – Item 5.2 of Appendix A to RG 1.183 states that consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to the atmosphere, e.g. emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank. Guidance for the analysis of this release pathway is taken from NUREG/CR-5950 as discussed in Section 5 of NRC Regulatory Issue Summary 2006-04.

Details regarding the inputs and assumptions for each of these release paths are provided below.

Primary Containment Leakage Pathway
The containment leakage pathway involves a release of nuclides from the core into the containment building, with direct leakage to the environment at a rate of 0.15% per day without further filtration. This leak rate is reduced by 50% after 24 hours consistent with Item 3.7 of Appendix A to RG 1.183. The chemical form of iodine released to containment is assumed to be 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide as directed by Section 2 of Appendix A to RG 1.183.

ESF Leakage Pathway
The ESF leakage pathway represents fluid from the containment sump which leaks into the Auxiliary Building through valve packing glands, pump shaft seals, etc. An actual leakage rate of 20,000 cc/hr (0.0118 cfm) is assumed, which is doubled to 0.0236 cfm in the analysis as required by Item 5.2 of Appendix A to RG 1.183. The leakage begins coincident with the transfer of ECCS pump suction to the containment sump at 20 minutes into the event, and the amount of this leakage which flashes to vapor is calculated based upon the constant enthalpy process described in Item 5.4 of Appendix A to RG 1.183. The resultant flashing fraction of 5.5% is based upon a maximum sump temperature of 265 °F as well as a peak post-LOCA containment pressure of 58.5 psia, and is conservatively applied for the duration of the event. Since particulates remain with the liquid, only iodine is considered for release through this pathway (RG 1.183, Appendix A, Item 5.3). The iodine that is postulated to be available for release is assumed to be 97% elemental and 3% organic consistent with Item 5.6 of Appendix A to RG 1.183.

ESF leakage into the Auxiliary Building begins 20 minutes after the start of the event and is initially released directly to the environment without filtration. After 30 minutes from the start of the event, the penetration room filtration system is

placed in service. The efficiency of the penetration room filters is 89.5% for all forms of iodine.

Containment Purge Pathway
The Containment Purge pathway simulates the release from the containment atmosphere via normal containment ventilation at the start of the event. This pathway represents a flow rate of 2850 cfm through the mini-purge system for the first 30 seconds of the event until the ventilation is isolated. One hundred percent (100%) of the nuclide activity in the RCS is distributed into the containment volume and is immediately available for release to the environment without filtration. Section 3.5 of RG 1.183 directs that the radioiodine released from the RCS to containment have an iodine composition of 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide.

RWST Backleakage Pathway
The RWST Backleakage pathway accounts for the potential release of iodine which may be transported from the containment sump to the RWST Containment Purge pathway through isolation valves in ECCS systems. An actual leakage rate of 1 gpm is assumed, which is doubled to 2 gpm in the analysis to be consistent with Item 5.2 of Appendix A to RG 1.183.

One hundred percent (100%) of the iodine in solution in the RWST is considered available for release; however, the fraction of this iodine which re-evolves into elemental iodine is derived using the guidance of NUREG/CR-5950. This document defines a relationship for the fraction of total iodine in solution which is released in the form of volatile elemental iodine as function of pH and total iodine concentration. This relationship is illustrated in Figure 3.1 of NUREG/CR-5950, which shows that very little of the iodine in solution is converted to elemental iodine for pH values greater than 7.0, but substantially more volatile iodine will re-evolve in more acidic solutions such as that in the RWST. In addition, Section 3.3 of this reference identifies a temperature-dependent partition coefficient which relates the relative concentrations of elemental iodine in the liquid and vapor phases of a pool or tank. This partition factor is used in combination with the iodine release model to determine the amount of volatile iodine in the RWST vapor space with respect to the total amount of iodine in the tank.

In addition, since Section 2 of Appendix A to RG 1.183 identifies that 0.15 percent of the iodine in the containment sump is in the organic chemical form, it is assumed that this portion of the iodine in the sump does not re-evolve as elemental iodine and is included with the volatile iodine released from the RWST. Time dependent values for the elemental iodine fraction and partition coefficient used in the analysis are listed below.

Note that the RADTRAD model for this pathway makes no further distinction in the chemical form of the iodine since all of the re-evolved elemental plus initial organic iodine is released, and the control room filter efficiencies are the same for both forms.

RWST Elemental Iodine Partition Coefficient

Time (hr)	Temperature (°F)	Temperature (°K)	Partition Coefficient
0.00	100.0	310.9	45.41
0.333	100.0	310.9	45.41
0.50	100.1	311.0	45.33
1.0	100.3	311.1	45.15
5.0	102.3	312.2	43.46
10.0	104.5	313.4	41.68
15.0	106.3	314.4	40.27
25.0	109.3	316.1	38.04
50.0	114.7	319.1	34.32
75.0	118.4	321.2	31.98
100.0	121.1	322.7	30.37
125.0	123.0	323.7	29.29
150.0	124.5	324.5	28.47
200.0	126.5	325.7	27.40
250.0	127.6	326.3	26.84
300.0	128.3	326.7	26.48
350.0	128.8	326.9	26.23
400.0	129.1	327.1	26.08
450.0	129.3	327.2	25.98
500.0	129.3	327.2	25.98
550.0	129.3	327.2	25.98
600.0	129.2	327.2	26.03
650.0	129.1	327.1	26.08
700.0	128.9	327.0	26.18
720.0	128.8	326.9	26.23

RWST Elemental Iodine Release Fraction

Time (hr)	Total RWST Iodine Concentration (g-atom/l)	RWST pH	I_2 Release Fraction
0.00	0.000E+00	4.500	0.0000
0.333	0.000E+00	4.500	0.0000
0.50	3.125E-08	4.500	0.0007
1.0	1.245E-07	4.501	0.0028
5.0	8.573E-07	4.508	0.0181
10.0	1.740E-06	4.517	0.0345
15.0	2.589E-06	4.526	0.0485
25.0	4.191E-06	4.542	0.0707
50.0	7.714E-06	4.581	0.1053
75.0	1.068E-05	4.616	0.1234
100.0	1.321E-05	4.649	0.1329
125.0	1.540E-05	4.680	0.1373
150.0	1.731E-05	4.708	0.1386
200.0	2.047E-05	4.760	0.1360
250.0	2.299E-05	4.807	0.1301
300.0	2.505E-05	4.849	0.1228
350.0	2.676E-05	4.887	0.1152
400.0	2.820E-05	4.922	0.1077
450.0	2.943E-05	4.954	0.1006
500.0	3.050E-05	4.984	0.0939
550.0	3.143E-05	5.012	0.0876
600.0	3.225E-05	5.038	0.0819
650.0	3.298E-05	5.063	0.0766
700.0	3.363E-05	5.087	0.0717
720.0	3.387E-05	5.096	0.0699

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.

SNC Response to RAI No. 4:

In Enclosure 6, the unsprayed compartment volume, as used in the analysis, should have been 361,340 ft^3, rather than 361,240 ft^3.

$$Sprayed\ Compartment\ Volume = 1{,}668{,}660\ ft^3$$

$$Unsprayed\ Compartment\ Volume = 361{,}340\ ft^3$$

$$\boldsymbol{Sprayed + Unsprayed\ Volume = 1{,}668{,}660\ ft^3 + 361{,}340\ ft^3 = 2{,}030{,}000\ ft^3}$$

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.

SNC Response to RAI No. 5:

The CREFS unfiltered in-leakages are different between the analyses in part due to the assumed nature of the in-leakages. In preparation for the LAR, SNC reperformed the LOCA and FHA analyses and chose to use a higher unfiltered in-leakage (325 cfm). In comparison to the most recent in-leakage tests, performed in February of 2016, 300 or 325 CFM is still very conservative. In the pressurization mode, the worst as-tested leakage was 54 CFM and for the isolation mode, the worst as-tested leakage was 41 CFM. Therefore, the assumed unfiltered in-leakage is many times higher than the as-tested unfiltered in-leakage.

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.

<u>SNC Response to RAI No. 6:</u>

Table 2 of Enclosure 12 of the original submittal incorrectly states that the licensing basis value for the elemental iodine spray removal coefficient is 2.7 hr^{-1}. The correct value, as stated in the FSAR is 10 hr^{-1}. The AST analysis assumes 13 hr^{-1}.

The elemental spray removal coefficient of 13.7 hr^{-1} is a historical value provided by Westinghouse in WCAP-11611, prepared for Plant Farley to address the removal of the Spray Additive Tank from the Containment Spray System.. The methodologies in the WCAP are based upon a "reference plant" and were developed using a method which pre-dated Revision 2 of NUREG-0800 Chapter 6.5.2.

While the details of the method used to generate the elemental spray removal coefficient are not specified in WCAP-11611, the value of 13.7 hr^{-1} is conservative with respect to the Farley specific coefficient which is calculated based upon Revision 4 to NUREG-0800 Chapter 6.5.2 as illustrated below.

Section III.4.C.i of NUREG-0800 Chapter 6.5.2 provides the following relationship for the spray removal coefficient for elemental iodine:

$$\lambda_s = \frac{6K_g TF}{VD}$$

Where:

λ_s = Elemental iodine removal coefficient (hr^{-1})

K_g = Gas-phase mass-transfer coefficient (m/sec)

T = Spray drop fall time (sec)

F = Spray volumetric flow rate (gpm)

V = Sprayed compartment volume (ft^3)

D = Mass mean spray drop diameter (μm)

The gas-phase mass transfer coefficient (K_g) is given in WCAP-11611 as 9.84 ft/min. Converting to SI units yields a coefficient of 0.05 m/sec. The mass mean spray drop diameter (D) is listed in FSAR 6.2.2.2.1 as 700 μm. Using this drop size with Model C of Figure 16 from NUREG/CR-5966, the spray drop terminal velocity is approximately 525 cm/sec. Based upon a fall height of 100 ft and a terminal velocity of 525 cm/sec, the spray drop fall time (T) is:

$$Spray\ Drop\ Fall\ Time = \frac{(100\ ft)(12\ in/ft)(2.54\ cm/in)}{525\ cm/sec} = 5.81\ sec$$

The volume of the sprayed compartment (V) is 1,668,660 ft^3. Applying a minimum containment spray flow rate (F) of 2,290 gpm, the Farley-specific elemental iodine removal rate for sprays is calculated as:

$$\lambda_s = \frac{6(0.05\ m/sec)(5.81\ sec)(2{,}290\ gal/min)(1.0E06\ \mu m/m)(60\ min/hr)}{(1{,}668{,}660\ ft^3)(700\ \mu m)(7.4805\ gal/ft^3)} = 27.1\ hr^{-1}$$

Note that Section III.4.C.i of NUREG-0800 Chapter 6.5.2 requires that λ_s be limited to 20 hr^{-1}. Therefore, a value of 20 hr^{-1} would be applicable to the Farley containment and confirms the value of 13.7 hr^{-1} used in the analysis is conservative.

<u>RAI No. 7</u> (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.

SNC Response to RAI No. 7:

The Farley Control Room HVAC system is designed such that a containment isolation actuation signal (CIAS) from the engineered safety features actuation system (ESFAS) automatically switches the control room HVAC system from normal to emergency mode of operation. In addition, high radiation levels entering the control room will automatically isolate the normal air systems with the pressurization and recirculation systems being manually initiated by the operator.

A CIAS phase A initiation signal (or high radiation entering the control room intakes) causes the normal makeup air to be isolated and all control room isolation valves to be closed. In this event the positive pressure in the control room is maintained by the startup of one of the emergency pressurization systems, leading to the pressurization mode discussed in the LAR submittal. Emergency pressurization is automatically initiated by the CIAS signal. As shown in the FNP FSAR Table 7.3-16, the longest time that is required to generate an SI signal is 27 seconds. The signal processing time for a CIAS from an SI signal is less than one second. This signal happens in the LOCA, MSLB, SGTR, and CRE accidents. The operators must manually initiate pressurization mode in the Locked Rotor Accident and the Fuel Handling Accident. In those two accidents, SNC assumes a 20 minute operator action time to achieve pressurization.

It is worth noting that in the normal mode of operation, the control room is kept at a slightly positive pressure to the surrounding areas of the control room. SNC has not taken credit for this normal condition in the evaluation of operator doses for the DBAs.

From the time the controlling instrument (which can vary depending upon the accident) senses the condition that would initiate an SI or CIAS signal, less than one second passes before pressurization begins. The control room becomes pressurized in less than 1 minute from the accident. Therefore, since transport from the radiation release point to the control room intakes is not instantaneous, contributions from the release to operator doses are insignificant for that time.

In the LOCA accident the following timing is assumed:

The accident happens at time = 0 hours. The Normal CR HVAC is running allowing 2340 CFM of unfiltered flow into the CR. At 1 minute (0.0167 hours) the pressurization of the CR is achieved and 375 CFM of filtered air is allowed into the CR. The CR HVAC stays in this mode for the duration of the accident. In addition, separate to these flows, 325 CFM of unfiltered in-leakage is assumed to enter the CR through the HVAC system. This in-leakage is assumed to be drawn from the area outside the building at the CR emergency air intakes – it is contaminated. This unfiltered in-leakage starts at time = 0 hours and lasts the entire duration of the accident (720 hours). The 325 CFM of unfiltered in-leakage

assumes 10 CFM for ingress/egress by the operators. Note that the area around the CR doors in the auxiliary building is not contaminated.

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.

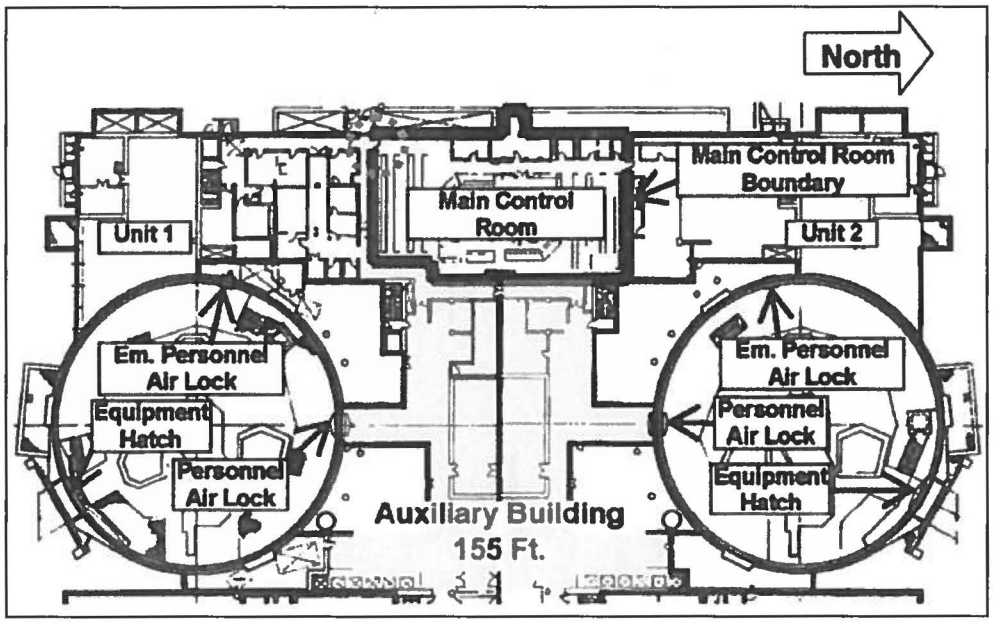
Considering that the auxiliary building ventilation system does not meet RG 1.183 regulatory position 5.1.2, please demonstrate how mixing occurs without the auxiliary building ventilation system, the evaluation should consider the magnitude of the building volume and exhaust rate, the potential for 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.

SNC Response to RAI No. 8:

Without the operation of the Containment and Auxiliary Building ventilation systems, there would be no significant release of radioactivity following a fuel handling accident (FHA) inside Containment. The assumption of their operation is not to mitigate the event, but to provide a means of releasing the radioactivity to the environment. No credit is taken for their filtration capabilities. Mixing is assumed to occur instantaneously in the Auxiliary Building. No credit is taken for isotopic decay during mixing.

The Auxiliary Building volume between the open personnel airlock (PAL) and the Main Control Room (MCR) entrance was not modeled to mitigate the effects of the FHA inside Containment. It was modeled to maximize the dose contribution from the 10 CFM unfiltered in-leakage due to MCR ingress/egress.

As shown below, the Containment Operating Deck and PAL are at the same elevation as the MCR (EL 155'-0"). In the event of a FHA in Containment with the PAL open, the volume of the Auxiliary Building between the PAL and the MCR may contain unfiltered radionuclides.



The assumed Auxiliary Building volume at this elevation was selected to provide the minimum volume between the open PAL and the MCR entrance. This provided a path for the radionuclides to the MCR, while minimizing dilution in the Auxiliary Building. The flow rate through the open PAL and this Auxiliary Building volume was selected, based on sensitivity studies, to maximize the MCR dose contribution from the 10 CFM ingress/egress unfiltered in-leakage. It is conservative to assume holdup in the Auxiliary Building for this MCR dose contribution.

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.

SNC Response to RAI No. 9:

As described in the response to RAI No. 7, the FHA does not cause an SI or CIAS signal. Contamination from the accident will reach the radiation monitors at the control room intakes, the radiation monitor signal will initiate the automatic isolation of the control room, and then the operators will manually initiate the emergency pressurization mode of CREFS operation. The FHA analysis evaluated the time from the event to the closure of the isolation dampers (from either FHA location) and found the time to be substantially less than 1 minute. The analysis conservatively assumes that the radiation monitor initiates isolation mode 1 minute after the event for either FHA location, and the operators take twenty minutes from that time to initiate the pressurization mode.

The current licensing basis assumes the operators take ten minutes to initiate pressurization mode, in keeping with the general assumption of time to complete a manual action from the control room. The AST FHA analysis conservatively assumes a longer time, with acceptable results in comparison to regulatory limits.

With respect to operator performance times for the manual accident to achieve CREFS pressurization mode for Plant Farley, recent testing has shown a maximum time of 9 minutes, 55 seconds to achieve the action.

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.

SNC Response to RAI No. 10:

The filter efficiencies assumed in each of the analyses described above are appropriate for the analyses. For FHA, 94.5% was conservatively assumed for all iodine species and adds margin to the analysis results.

RG 1.183 Appendix B regulatory position 2.0 states that the iodine above the refueling cavity or spent fuel pool water is composed of 57% elemental and 43% organic species. There is no mention of particulate iodine.

In addition, RG 1.183 Appendix B regulatory position 3 states that particulate radionuclides are assumed to be retained by the refueling cavity or spent fuel pool water.

Assuming a CREFS recirculation charcoal filter efficiency of 94.5% is therefore appropriate and conservative for all iodine species (elemental and organic) released by the fuel handling accident.

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.

SNC Response to RAI No. 11:

The SNC FHA dose analysis is consistent with RG 1.183. The "starting point" of the SNC FHA dose analysis was the radionuclide inventory instantaneously and uniformly distributed above the SFP or refueling cavity water. This inventory was calculated as follows:

$$\text{Inventory Above Water (Curies)} = \frac{(1 + \text{Design Margin}) \times \text{Core Inventory (Curies)} \times \frac{\text{1 Fuel Assembly}}{\text{157 Fuel Assemblies}} \times \text{COLR Radial Peaking Factor} \times \text{Gap Fraction}}{\text{Decontamination Factor}}$$

In accordance with RG 1.183 Appendix B regulatory position 3, the decontamination factor for particulates, such as alkali metals, is infinity because they are retained in the fuel pool or refueling cavity water. Inserting a decontamination factor of infinity results in the particulate inventory above the water being zero.

SNC notes that ignoring alkali metals released from the fuel rod gap is appropriate and is equivalent to assuming that the 12% of alkali metals released from the gap are all retained in the spent fuel pool or refueling cavity water.

SNC will revise Table 3.6a in enclosure 1 and table 1 of enclosure 7 of the LAR to add the alkali metals release fraction and a footnote that the alkali metals are retained in the spent fuel pool or refueling cavity water.

SNC confirms 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.

For Page E7-3 of the LAR, the following is substituted:

Fraction of Fission Product Inventory in Gap	
I-131	0.08
Kr -85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals (retained in pool)	0.12
Iodine Chemical Forms Released from Fuel to Pool	
Elemental	99.85%
Organic	0.15%
Overlaying Pool Depth	23 feet
Decontamination Factors	
Noble Gases	1
Particulates	Infinity
Overall Iodine	200
Iodine Chemical Forms Released from Pool	
Elemental	57%
Organic	43%

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.

SNC Response to RAI No. 12:

SNCs' fuel handling accident analysis is consistent with RG 1.183.

The RG 1.183 Appendix B regulatory position 1.2 identifies radionuclides (xenons, kryptons, halogens, cesiums, and rubidiums) that should be considered for the gap activity.

All radionuclides listed in RG 1.183 regulatory position 3.4, including those above, were considered. Those that would not contribute to the dose consequences were identified and eliminated for further evaluation. This was determined in a two-step screening process.

The first screen was based on the element's state.

Group	Elements	State
Noble Gases	Xenon and Krypton	Gas
Halogens	Iodine and Bromine	Gas
Alkali Metals	Cesium and Rubidium	Particulate
Tellurium Group	Tellurium, Antimony, Selenium, Barium, and Strontium	Particulate
Noble Metals	Ruthenium, Rhodium, Palladium, Molybdenum, Technetium, and Cobalt	Particulate
Lanthanides	Lanthanum, Zirconium, Neodymium, Europium, Niobium, Promethium, Praseodymium, Samarium, Yttrium, Curium,, and Americium	Particulate
Cerium	Cerium, Plutonium, and Neptunium	Particulate

Per RG 1.183 Appendix B regulatory position 3, particulates, will be retained by the water in the spent fuel pool or refueling cavity. As a result, only the noble

gases (xenons and kryptons) and halogens (iodine and bromine) screened in for further consideration.

The second screen compared the noble gas and halogen core inventories at 100 hours after shutdown to those of the major dose contributors, Xe-133, Xe-133m, and I-131. If a given radionuclide's inventory at 100 hours after shutdown was two or more orders of magnitude less than the major dose contributors' inventory, it was removed from further evaluation because its contribution to the dose consequences would be expected to be negligible.

This two-step screen resulted in the following radionuclides selected for determining the fuel handling accident dose consequences:

- Halogens: I-131, I-132, I-133, and I-135
- Noble Gases: Kr-85, Xe-131m, Xe-133m, Xe-133, and Xe-135

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.

SNC Response to RAI No. 13:

SNC is not 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.

Prior to the postulated fuel assembly drop, the SFP air supply and exhaust systems maintain a slightly negative pressure above the SFP. The PRF system is operable but is not in service. The PRF SFP isolation dampers are normally open to the SFP area to provide an open path for air flow to the PRF system. There is a pair of radiation detectors in the SFP exhaust ductwork. On a high-high radiation alarm, the SFP air supply and exhaust dampers close and the PRF exhaust fans start. The air flow rate from the SFP area is limited by the PRF SFP isolation dampers.

SNC has determined that the time delay between the radionuclides reaching the SFP exhaust system radiation monitor sample point and the exhaust isolation dampers closing is ~3.6 seconds. SNC has also determined that the time delay between the radionuclides reaching the radiation monitor sample point and then arriving at the exhaust isolation dampers is ~5.7 seconds at the exhaust design flow rate of 20,000 CFM. At the exhaust nominal operating flow rate of 13,100 CFM this time delay is ~8.8 seconds.

This ensures the fuel handling accident in the SFP release is through the PRF alone. With normal air supply isolated, the PRF maintains the slightly negative pressure.

Thus, the release to the environment begins only after the PRF starts. The instrumentation and PRF system timing prior to PRF start has no effect on the dose consequences.

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.

SNC Response to RAI No. 14:

In accordance with Regulatory Position 2 of RG 1.183 Appendix B, SNC applied an overall decontamination factor (DF) of 200 for the iodine released from a fuel handling accident.

The pool DF in Table 2 of enclosure 7 to the LAR is corrected to be consistent with the DF in Table 3.6a of enclosure 1 to the LAR as follows.

Overlaying Pool Depth	23 feet
Iodine Chemical Form Released from Fuel to Water	
Elemental	99.85%
Organic	0.15%
Overall Effective Decontamination Factor (DF) for Iodine	200
Iodine Chemical Form Released from Water	
Elemental	57%
Organic	43%

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.

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[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.

SNC Response to RAI No. 15:

SNC will provide the results for the additional analyses in a supplemental RAI response.

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	5.06×10^{-3}
	30 seconds – 2 hours	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.

SNC Response to RAI No. 16:

Without the operation of the Auxiliary Building ventilation system, there would be no significant release of radioactivity following a fuel handling accident (FHA) inside Containment. The assumption of its operation is not to mitigate the event but to provide a means of releasing the radioactivity to the environment. No credit is taken for its filtration capabilities.

The following current licensing basis x/Qs are used for the release via the Plant Vent Stack:

Time	Main Control Room	Exclusion Area Boundary	Low Population Zone
0-2 hours	1.62E-03 sec/m^3	7.6E-04 sec/m3	2.8E-04 sec/m^3
2-8 hours	1.37E-03 sec/m^3	N/A	1.1E-04 sec/m^3

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.

SNC Response to RAI No. 17:

Item #2, above, is the provision for the isolation of the flow path, which is consistent with TSTF-312.

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

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.

SNC Response to RAI No. 18:

For the Main Steam Line Break (MSLB) accident, the CR isolates and enters pressurization mode automatically upon a CIAS signal generated in response to an SI signal. As described in the response to RAI no. 7, the maximum amount of time for the SI signal to occur is 27 seconds, The SI to CIAS signal generation

time is less than one second, and pressurization occurs in less than a minute. Consequently, the time to pressurize is ignored in the analysis. This assumption has insignificant impact upon the total operator dose, which is significantly less than the regulatory limits. During the accident, the CR HVAC supplies 375 CFM of filtered air make-up to the CR. SNC assumes 310 CFM of unfiltered in-leakage from the HVAC system, taken from the contaminated air adjacent to the emergency HVAC intakes. The unfiltered in-leakage includes 10 CFM for CR ingress and egress. The analysis assumes this control room ventilation scheme for the course of the accident.

RAI No. 19 (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.

<u>SNC Response to RAI No. 19:</u>

The steam mass flow rates presented in Table 3.7a of Enclosure 1 are the mass flow rates for the MSLB calculated in the transient break analysis. The values in Table 15.4-23 in FNP UFSAR revision 21 5/08 were obtained by multiplying the values in Table 3.7a of Enclosure 1 by 1.05 to add margin. For the AST MSLB analysis SNC desired to add additional margin. As seen in the notes in Table 2 of Enclosure 8, 110% of the transient analysis mass flow rates were used for greater conservatism. Note the calculations below:

For the FSAR values:

For the steam flow rate from the intact steam generators:
316715 *1.05 = ~333,000 lbm/hr.
703687 * 1.05 = ~739,000 lbm/hr.
948000 * 1.05 = ~ 995,000 lbm/hr.
For the total steam flow from the faulted steam generator:
439145 * 1.05 = ~461,000 lbm.

For the LAR Table values in Enclosure 8

For the steam flow rate from the intact steam generators:
316715 *1.1 = ~348,000 lbm/hr.

703687 * 1.1 = ~783,000 lbm/hr.
948000 * 1.1 = ~ 1,040,000 lbm/hr.
For the total steam flow from the faulted steam generator:
439145 * 1.1 = ~483,000 lbm.

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.

SNC Response to RAI No. 20:

The statements above are correct. LAR Enclosure 8, Table 5 contains a transcription error. The correct value of 2.0E+01 Ci was used in the analysis, and is also reflected in the corrected version of Table 5 shown below:

Table 5 ~ Initial Alkali Metal Concentrations in the Secondary System

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

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

concentrations (in μCi/gm) in Column 2 of this table are multiplied by this mass and then by 1.0E-6 Ci/μCi.

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.

SNC Response to RAI No. 21:

For the Steam Generator Tube Rupture (SGTR) Accident, the CR isolates and enters pressurization mode automatically upon a CIAS signal generated in response to an SI signal. As described in the response to RAI no. 7, the maximum amount of time for the SI signal to occur is 27 seconds, The SI to CIAS signal generation time is less than one second, and pressurization occurs in less than a minute. Consequently, the time to pressurize is ignored in the analysis. This assumption has insignificant impact upon the total operator dose, which is significantly less than the regulatory limits. During the accident, the CR HVAC supplies 375 CFM of filtered air make-up to the CR. SNC assumes 310 CFM of unfiltered in-leakage from the HVAC system, taken from the contaminated air adjacent to the emergency HVAC intakes. The unfiltered in-leakage includes 10 CFM for CR ingress and egress. The analysis assumes this control room ventilation scheme for the course of the accident.

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.

<u>SNC Response to RAI No. 22:</u>

Table 36 of NUREG/CR-6189 presents decontamination coefficient correlations for five post-accident time intervals. For each time interval, correlations are provided at 90, 50, and 10 percentile confidence levels. For the Farley CREA, the following 10th percentile deposition rate associated with the time interval of 0 to 1800 sec is used:

$$\lambda_{0\text{-}1800} = 0.0182 + (3.260E\text{-}6)P = 2.74E\text{-}2\ hr^{-1}$$

P is the reactor thermal power of 2831 MW. This correlation is used because the 10th percentile correlation is the most conservative of the three confidence levels, yielding the lowest deposition rate for a given time interval. Also, as the accident time progresses from 0 to 80000 sec, the deposition rate corresponding to the 10th percentile correlation generally increases as follows:

$$\lambda_{1800\text{-}6480} = 0.0645[1 - \exp(-0.938P/1000)] = 6.00E\text{-}2\ hr^{-1}$$
$$\lambda_{6480\text{-}13680} = 0.094[1 - \exp(-0.869P/1000)] = 8.60E\text{-}2\ hr^{-1}$$
$$\lambda_{13680\text{-}49680} = 0.0811 + (10.15E\text{-}6)P = 1.10E\text{-}1\ hr^{-1}$$
$$\lambda_{49680\text{-}80000} = 0.0860[1 - \exp(-2.384P/1000)] = 8.59E\text{-}2\ hr^{-1}$$

Therefore, the lowest, most conservative deposition rate occurs at the 0-1800 sec time interval. Furthermore, a decontamination factor (DF) of 50 is applied to the deposition rate to limit the fraction of particulates airborne in containment that are removed due to deposition. The LocaDose computer program, which is used to model the Farley CREA, automatically initiates re-suspension of particulates into the air once the DF level is reached.

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.

SNC Response to RAI No. 23:

For the Control Rod Ejection (CRE) accident, the CR isolates and enters pressurization mode automatically upon a CIAS signal generated in response to an SI signal. As described in the response to RAI no. 7, the maximum amount of time for the SI signal to occur is 27 seconds, The SI to CIAS signal generation time is less than one second, and pressurization occurs in less than a minute. Consequently, the time to pressurize is ignored in the analysis. This assumption has insignificant impact upon the total operator dose, which is significantly less than the regulatory limits. During the accident, the CR HVAC supplies 375 CFM of filtered air make-up to the CR. SNC assumes 310 CFM of unfiltered in-leakage from the HVAC system, taken from the contaminated air adjacent to the emergency HVAC intakes. The unfiltered in-leakage includes 10 CFM for CR ingress and egress. The analysis assumes this control room ventilation scheme for the course of the accident.

Enclosure 6 of the LAR indicates that the time required to pressurize the control room and initiate the control room emergency filtration system (CREFS) is 60 sec and that the normal intake rate prior to this is 2340 cfm. For the CREA, it is assumed that CREFS is in operation at the start of the accident. While this is done to simplify the analysis, it has a negligible impact on control room dose, as demonstrated below.

With CREFS in operation, the filtered intake flow is 375 cfm at a filter efficiency of 98.5% while the unfiltered in-leakage is 310 cfm, including 10 cfm for ingress/egress. This is equivalent to the following unfiltered intake:

310 + (375)(100% – 98.5%) = 316 cfm

The unfiltered intake prior to pressurization is 2340 cfm, yielding the following ratio:

2340/316 = 7.4

The LocaDose computer program is used to calculate doses. The program output in the CREA calculation shows the control room dose as a function of time. While there is no dose shown at 60 sec, the dose at 98 sec is 2.5E-4 rem, which is 0.007% of the total 30-day dose of 3.7 rem. Conservatively using the 98-sec dose at 60 sec, the increase in dose if CREFS were not credited during the first 60 sec may be estimated as follows, assuming dose is proportional to intake:

(2.5E-4 rem)(7.4 – 1) = 1.6E-3 rem

This increase represents less than 0.1% of the total control room dose of 3.7 rem. Hence, assuming CREFS to be in operation at the start of the accident instead of at 60 sec has a negligible impact on control room dose.

<u>RAI No. 24</u> (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.

SNC Response to RAI No. 24:

The statement "no credit is taken for pH being controlled at values of 7 or greater" was made in error.

During the CRE accident, pressure in containment is not expected to reach the containment spray actuation set point. However, the Farley containment sump houses baskets of trisodium phosphate (TSP). SNC assumes the event introducing water into the sump causes dissolution of TSP, thereby maintaining the pH of the sump water at 7 or greater. Given that the pH condition of RG 1.183 is met, it may be assumed that iodine in the containment atmosphere is 95% particulate, 4.85% elemental, and 0.15% organic.

Furthermore, a decontamination factor (DF) of 50 is applied to the deposition rate to limit the fraction of particulates airborne in containment that are removed due to deposition. The LocaDose computer program, which is used to model the CREA, automatically initiates re-suspension of particulates into the air once the DF level is reached.

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>SNC Response to RAI No. 25:</u>

For the secondary system release pathway, regulatory position 3 of RG 1.183, Appendix H indicates that activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system. In evaluating this release for the Farley CRE accident, it is conservatively assumed that all accident-induced leakage from primary to secondary system is directly to the environment. Additionally, the pre-existing iodine in the secondary system is also assumed to be released directly to the environment. 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.

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.

RAI No. 26 (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.

SNC Response to RAI No. 26:
The information given in Table 2 of Enclosure 10 has typographical errors associated with transcription of the table from the analysis of record to the submittal with respect to core release and gap fractions for certain isotopes, as noted above. The CRE accident analysis used the following information, which is a revision of Table 2:

Isotope	Core Activity (curies)	Margin Factor	Core release	Gap fraction	Activities (curies)		
					Fuel Melt	Gap release	Total
I-131	7.50E+07	1.02	0.5	0.1	9.6E+04	1.3E+06	1.4E+06
I-132	1.10E+08	1.02	0.5	0.1	1.4E+05	1.9E+06	2.0E+06
I-133	1.60E+08	1.02	0.5	0.1	2.0E+05	2.8E+06	3.0E+06
I-134	1.70E+08	1.02	0.5	0.1	2.2E+05	2.9E+06	3.2E+06
I-135	1.50E+08	1.02	0.5	0.1	1.9E+05	2.6E+06	2.8E+06
Kr-83m	9.70E+06	1.02	1	0.1	2.5E+04	1.7E+05	1.9E+05
Kr-85m	2.10E+07	1.02	1	0.1	5.4E+04	3.6E+05	4.2E+05
Kr-85	7.20E+05	1.15	1	0.1	2.1E+03	1.4E+04	1.6E+04
Kr-87	4.00E+07	1.02	1	0.1	1.0E+05	6.9E+05	8.0E+05
Kr-88	5.70E+07	1.02	1	0.1	1.5E+05	9.9E+05	1.1E+06
Kr-89	6.90E+07	1.02	1	0.1	1.8E+05	1.2E+06	1.4E+06
Xe-131m	8.40E+05	1.02	1	0.1	2.1E+03	1.5E+04	1.7E+04
Xe-133m	4.80E+06	1.02	1	0.1	1.2E+04	8.3E+04	9.5E+04
Xe-133	1.50E+08	1.05	1	0.1	3.9E+05	2.7E+06	3.1E+06
Xe-135m	3.00E+07	1.02	1	0.1	7.7E+04	5.2E+05	6.0E+05
Xe-135	3.50E+07	1.02	1	0.1	8.9E+04	6.1E+05	7.0E+05
Xe-137	1.40E+08	1.02	1	0.1	3.6E+05	2.4E+06	2.8E+06
Xe-138	1.30E+08	1.02	1	0.1	3.3E+05	2.3E+06	2.6E+06
Br-82	3.80E+05	1.03	0.5	0.05	4.9E+02	3.3E+03	3.8E+03
Br-83	9.70E+06	1.03	0.5	0.05	1.2E+04	8.5E+04	9.7E+04
Br-84	1.70E+07	1.03	0.5	0.05	2.2E+04	1.5E+05	1.7E+05
Br-85	2.10E+07	1.03	0.5	0.05	2.7E+04	1.8E+05	2.1E+05
Br-86	1.50E+07	1.03	0.5	0.05	1.9E+04	1.3E+05	1.5E+05

Isotope	Core Activity (curies)	Margin Factor	Core release	Gap fraction	Activities (curies)		
					Fuel Melt	Gap release	Total
Br-87	3.40E+07	1.03	0.5	0.05	4.4E+04	3.0E+05	3.4E+05
Br-88	3.60E+07	1.03	0.5	0.05	4.6E+04	3.2E+05	3.6E+05
Rb-86	1.40E+05	1.03	0.5	0.12	1.8E+02	2.9E+03	3.1E+03
Rb-88	5.70E+07	1.03	0.5	0.12	7.3E+04	1.2E+06	1.3E+06
Rb-89	7.40E+07	1.03	0.5	0.12	9.5E+04	1.6E+06	1.7E+06
Rb-90	7.20E+07	1.03	0.5	0.12	9.3E+04	1.5E+06	1.6E+06
Rb-91	8.90E+07	1.03	0.5	0.12	1.1E+05	1.9E+06	2.0E+06
Cs-134m	3.60E+06	1.03	0.5	0.12	4.6E+03	7.6E+04	8.0E+04
Cs-134	1.10E+07	1.35	0.5	0.12	1.9E+04	3.0E+05	3.2E+05
Cs-136	3.30E+06	1.25	0.5	0.12	5.2E+03	8.4E+04	8.9E+04
Cs-137	7.60E+06	1.20	0.5	0.12	1.1E+04	1.9E+05	2.0E+05
Cs-138	1.40E+08	1.03	0.5	0.12	1.8E+05	2.9E+06	3.1E+06
Cs-139	1.40E+08	1.03	0.5	0.12	1.8E+05	2.9E+06	3.1E+06
Cs-140	1.20E+08	1.03	0.5	0.12	1.5E+05	2.5E+06	2.7E+06
Cs-141	9.10E+07	1.03	0.5	0.12	1.2E+05	1.9E+06	2.0E+06
Note	2	3	4	5	6	7	8

Containment and RCS Activities notes:

1. The note numbers correspond to column numbers.
2. Core Activity – At shutdown.
3. Margin Factor – Accounts for cycle variations.
4. Core Release – Applies to melted fuel rods.
5. Gap Fraction – Applies to fuel rods with cladding failure.
6. Fuel Melt Activity – Product of Activity (Ci) in Column 2, margin in Column 3, core release in Column 4 and fuel melt fraction of 0.0025.
7. Gap Activity – Product of Activity (Ci) in Column 2, margin in Column 3, gap fraction in Column 5, fuel cladding failure fraction of 0.1, and RPF of 1.7.
8. Total Activity – This is the sum of Columns 6 and 7.

RAI No. 27 (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.

SNC Response to RAI No. 27:

In the CREA analysis, it is conservatively assumed that all accident-induced leakage from primary to secondary system is directly to the environment. This leakage continues for 2500 sec, the time required for the pressures within the primary and secondary systems to equalize.

Additionally, the pre-existing activity in the secondary system is also assumed to be released directly to the environment. As indicated on Page E10-3 of Enclosure 10, the total mass of the secondary system fluid released during the accident is 468,600 lbm. Based on an analysis, it has been determined that the minimum

possible time required to release this mass through the main steam safety valves is 98 sec. In the dose analysis, the flow from the steam generators to the environment is set to a rate that effectively exhausts this mass in 98 sec.

Hence, the analysis does reflect a condition where primary-to-secondary release continues until shutdown cooling is in operation and releases from the steam generators have been terminated, consistent with RG 1.183, Appendix H, regulatory position 7.1.

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.

SNC Response to RAI No. 28:

SNC is not crediting any form of iodine or aerosol removal by the Containment Spray System for the CRE Accident analysis. The line in Table 6 of Enclosure 12 is incorrect. The line should read:

Table 6: Control Rod Ejection Accident Inputs and Assumptions			
Input/Assumption	CLB Value For Offsite and Control Room	New AST Value For Offsite and Control Room	Reason for Change
Iodine/Particulate Removal by Containment Sprays	Not Used	Not used	No Change

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.

SNC Response to RAI No. 29:

Section 3.7 of Enclosure 1 correctly states that no credit is taken for natural deposition of elemental iodine. Table G of Enclosure 5 incorrectly indicates that this removal is credited. Natural deposition credit is taken for particulates only.

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.

SNC Response to RAI No. 30:

The doses provided are the total for the accident from both the containment and secondary release pathways. The LocaDose computer program is used to calculate doses. As both pathways are modeled in a single LocaDose run, the output shows only the total doses from both pathways.

RAI No. 31 (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.

SNC Response to RAI No. 31:

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. The implications of this modeling approach are evaluated below by considering the release of I-131, the isotope contributing the most to doses.

For I-131, the activity in the primary coolant is based on the following parameters, as indicated in Table 2 of LAR Enclosure 10:

- Core activity = 7.5E7 Ci
- Margin factor = 1.02
- Core melt fraction = 0.0025
- Core release fraction = 0.50
- Gap fraction = 0.10
- Gap release fraction = 0.10
- Radial peaking factor (RPF) = 1.7

It is reasonable to assume that the rods that experience fuel melt are a subset of those that undergo cladding damage. As the activity in the gap is accounted for in the cladding damage source, the gap activity may be subtracted when considering the fuel melt source to avoid double counting. Hence, the non-gap fraction is 0.90.

The resulting I-131 activity available for release from the reactor coolant is as follows:

- Fuel melt activity = (activity)(margin)(melt frac)(non-gap frac)(core release frac)(RPF) = (7.5E7)(1.02)(0.0025)(0.90)(0.50)(1.7) = 1.46E5 Ci
- Gap activity = (activity)(margin)(gap frac)(gap release frac)(RPF) = (7.5E7)(1.02)(0.10)(0.10)(1.7) = 1.30E6 Ci
- Total activity = 1.46E5 + 1.30E6 = 1.446E6 Ci ≈ 1.4E6 Ci

The I-131 parameters for the containment release are the same as for reactor coolant except that core release fraction is 0.25 instead of 0.50. The resulting I-131 activity available for release from the containment is as follows:

- Fuel melt activity = (activity)(margin)(melt frac)(non-gap frac)(core release frac)(RPF) = (7.5E7)(1.02)(0.0025)(0.90)(0.25)(1.7) = 7.32E4 Ci
- Gap activity = 1.30E6 Ci (same as reactor coolant release)
- Total activity = 7.32E4 + 1.30E6 = 1.373E6 Ci ≈ 1.4E6 Ci

The LocaDose computer program is used to calculate doses. As the core activity values are limited to two significant figures, the LocaDose inputs for reactor coolant and containment activities are also limited to two significant figures. As demonstrated above, when rounded to two significant figures, both reactor coolant and containment activities are 1.4E6 Ci, consistent with the input to LocaDose, as shown in Table 2 of Enclosure 10. Hence, the impact of not applying the RPF to the fuel melt source is offset by that of doubling the fraction of iodine available for release from containment and double-counting the gap activity.

The sample calculation shown above is for I-131 only. It is possible that neglecting the RPF in the fuel melt source for other isotopes of iodine and noble gases are slightly non-conservative. However, there is another modeling assumption that adds significant conservatism to the analysis. RG 1.183, Appendix H, regulatory position 1 indicates that only iodine and noble gases are released from the core due to fuel melt. As indicated in Table 2 of Enclosure 10, alkali metals are also assumed to be released from the core following fuel melt, thereby introducing cesium in the fuel melt source as a significant contributor. The LocaDose output in the CREA calculation shows doses by isotope. In the control room, for example, of the total dose of 3.7 rem, 0.95 rem is due to Cs-134 and Cs-137. This conservatism of 25% offsets any impact of neglecting the RPF in the fuel melt source.

RAI No. 32 (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.

SNC Response to RAI No. 32:

It is possible that a Locked Rotor Accident will cause an SI signal to be generated. As discussed in the response to RAI no. 7, the maximum time for an SI signal to be generated is 27 seconds. The signal processing time from an SI signal to a CIAS signal is less than one second, and the pressurization mode would be initiated in less than one minute. The dose contribution of that one minute to enter the pressurization mode is insignificant.

However, for the Locked Rotor Accident, it is possible that the accident will not automatically initiate an SI signal or a CIAS signal. Therefore, the AST LRA analysis assumes that the CR HVAC supplies a normal intake flow of 2340 cfm to the CR, unfiltered until manual action is taken to initiate pressurization mode. The analysis assumes a 20 minute time for operators to achieve pressurization. After pressurization is achieved the analysis assumes a filtered make-up flow of 375 CFM from the CR HVAC system. SNC updated the analysis to assume 325 CFM of unfiltered in-leakage from the HVAC system, taken from the contaminated air adjacent to the emergency HVAC intakes. The unfiltered in-leakage includes 10 CFM for CR ingress and egress. The analysis assumes this control room ventilation scheme for the course of the accident.

With respect to operator performance times for the manual action to achieve CREFS pressurization mode for Plant Farley, testing has shown a maximum time of 9 minutes, 55 seconds to achieve the action.

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.

SNC Response to RAI No. 33:

The duration for the containment pathway was stated in error. There is no containment pathway for this accident. All releases are through the secondary side. However, in order to model instantaneous release of the noble gases to the environment, a pathway from the RCS to the environment is modeled in the analysis which has a high partition coefficient for the iodine isotopes and alkali metals, but no partitioning for the noble gases.

The revised sentence should state: "The duration of the release is 8 hours, until cold shutdown is established for the secondary side pathway. The doses at the EAB, LPZ, and the control room are calculated for 30 days post-accident."

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	0	2	5.64E+05	7.53E+01 cfm	3

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.

SNC Response to RAI No. 34:

The response to this item is similar to the response for RAI No. 19 for the MSLB accident analysis.

Table 3.10a of Enclosure 1 contains the vented steam release calculated by Westinghouse for the RSG, applicable to the locked rotor event.

Table 15.4-25A of the FSAR is based on the TID-14844 Locked Rotor Analysis performed for RSG. This analysis applied a 5% margin addition. The values presented in Table 3.10A of Enclosure 1 are multiplied by 1.05 and rounded to the nearest 1,000th to produce the values in FSAR Table 15.4-25A.

For AST, the margin on the steam and feedwater flow was increased to 10% for additional conservatism. Therefore, Table 1 of Enclosure 11 multiplies the values presented in Table 3.10a of Enclosure 1 by 1.1, and are rounded to the nearest 1,000th. The Table 1, Enclosure 11 values are input to the AST Locked Rotor Analysis.

RAI No. 35 (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.

SNC Response to RAI No. 35:

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

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.

SNC Response to RAI No. 36:

The CR dose calculated assuming instantaneous CREFS actuation is 0.36 rem TEDE. The re-assessment, which initiates CREFS at 20 minutes post-accident, calculates a 0.03 rem addition to the CR dose. Therefore, for a 20 minute CREFS initiation delay, the 30 day CR dose is 0.39 rem TEDE. This remains below the 5 rem TEDE acceptance criteria.

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.

SNC Response to RAI No. 37:

SNC is still in the process of responding to this question and will supplement this answer.

RAI No. 38 (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.

SNC Response to RAI No. 38:

SNC's LAR seeks AST implementation for radiological consequences of major DBAs, specifically: Loss of Coolant Accident, Fuel Handling Accident, Main Steam Line Break, Steam Generator Tube Rupture, Control Rod Ejection, and Locked Rotor Analysis. It also seeks to implement TSTF-312 and TSTF-448. There are no physical changes to the plant being proposed.

RG 1.183, Regulatory Position 1.3.2, "Re-Analysis Guidance," states:

> The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of

> the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid.

The only changes of assumptions or inputs are for the DBAs listed above, and those analyses' results have been submitted. All other existing analyses remain the same and, as mentioned above, there are no physical changes to the plant being proposed. If there are other changes (*e.g.*, procedural changes) after the approval of the LAR, then SNC will follow regulatory requirements (*e.g.*, 10 CFR 50.59).

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.

SNC Response to RAI No. 39:

SNC considered the potential impact of the shine sources mentioned above to the control room operator doses. However, due to the shielding, and the distance from the source (containment, plume, radiation filters), SNC made the determination that the shine would not contribute significantly to the dose. This decision is in keeping with the Standard Review Plan and Plant Farley's current Licensing Basis.

The Plant Farley Control Room is located within the Auxiliary Building of the Plant. It does not abut the containment, which consists of a concrete wall 3'9" thick. The floors and ceiling of the Control Room are 2'0" of concrete, and the exterior building walls are 21" thick. Radiation releases in the Auxiliary Building during a LOCA are assumed to occur in the Penetration Room Filtration System envelope, which is well away from the control room and does not communicate with it. The Penetration Room walls are 2'0" thick. Therefore, the radiation shine from the containment, from any releases in the Auxiliary Building, and from the radioactive plume is an insignificant contributor to operator dose.

The Control Room HVAC intake and recirculation filters are located in rooms above the control room and are separated by at least 2'0" of concrete from the occupied spaces in the Control Room. The radioactive sources are not in the

Control Room Envelope. As such, there is no significant contribution from these sources to operator dose.

RAI No. 40 (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 *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 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.

SNC Response to RAI No. 40:

SNC will perform an analysis of the dose an operator would receive during ingress and egress from the main control room during a LOCA. The LOCA accident is limiting for this analysis because the largest amount of radiation would be released during this accident. The response to this RAI will be supplemented upon completion of the analysis.

RAI No. 41 (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.

SNC Response to RAI No. 41:

The additional factors of conservatism for the various isotopes in each of the accidents analyzed do vary from analysis to analysis. These factors exist to add conservatism and margin for isotopes that are critical to the calculation of dose consequences for each DBA. Please note that these factors are used to add extra margin above the reactor concentrations for an equilibrium core.

The respective analyses were performed at different times, with the LOCA, LRA, and FHA analyses having been updated most recently. When first performed, the analyses showed that the LOCA and the FHA were bounding for dose consequences, and SNC chose to add extra margin to the LOCA and FHA calculations to enable us to address cores with higher enrichment and higher peak rod burnups – though not in excess of 5% enrichment or 62,000 MWD/MTU peak rod burnup. As stated in an earlier RAI response, cycle to cycle core designs assure that the predicted core equilibrium source term for the cycle will not exceed the analyzed source terms. For these reasons, the reported results do not need to be updated.

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.

SNC Response to RAI No. 42:

SNC is not intending to update any human factor reviews relating to emergency operating procedures (EOPs) as part of this LAR. If, in the future, SNC does change EOPs, it will do so in accordance with regulatory requirements.