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

July 21, 2017

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

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

Dear Mr. Hutto:

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

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

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

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

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that a request for additional information (RAI) is needed for NRC staff to complete its review, as provided in the Enclosure. The enclosure was discussed with your staff on July 12, 2017. SNC agreed to respond by September 7, 2017. Please note that the NRC staff's review is continuing and further requests for information may be developed.

Sincerely,

Shawn Williams

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

Docket Nos. 50-348, 50-364

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

REQUEST FOR ADDITIONAL INFORMATION

ALTERNATIVE SOURCE TERM, TSTF-448, TSTF-312

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NOS. 50-348 AND 50-364

By letter dated November 22, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession Package Number ML16336A024), Southern Nuclear Operating Company (SNC), the licensee, submitted a License Amendment Request (LAR) to revise Joseph M. Farley Nuclear Plant Units 1 and 2, (FNP) Technical Specifications (TS). The proposed change would revise FNP TS 3.7.10, "Control Room," TS 3.9.3, "Containment Penetrations," TS 5.5.18, "Control Room Integrity Program (CRIP)," and the current licensing basis to implement an alternative radiological source term for evaluating design basis accidents as allowed by Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.67, "Accident Source Term." The Nuclear Regulatory Commission (NRC) staff reviewed the impact of implementing an alternative radiological source term for evaluating design basis accidents (DBAs) on all DBAs currently analyzed in the FNP updated final safety analysis report (UFSAR) that could have the potential for significant dose consequences. The NRC staff requested additional information (RAI) in letter dated March 24, 2017 (ADAMS Accession Number ML17065A201). SNC responded to the NRC staff's RAI letter dated May 23 and June 8, 2017 (ADAMS Accession numbers ML17143A447 and ML17159A847, respectively). The NRC staff reviewed SNC's response letters and determined that additional information is needed to complete the review.

Regulatory Analysis Basis

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

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

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

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

2. NUREG-1431, "Standard Technical Specifications Westinghouse Plants Revision 4.0," Volume 1, Specifications dated April 2012 contains the improved standard technical specifications (STS) for Westinghouse plants. The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 *Federal Register* (FR) 39132), which was subsequently codified by changes to 10 CFR 50.36 (60 FR 36953). Licensees adopting portions of the improved STS to existing TSs should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.

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

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

5. NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," D. A Powers, et.al., USNRC, July 1996 (ADAMS Accession No. ML100130305).

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

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

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

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

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

SNC's response to RAI No. 2 states:

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

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

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

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

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

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

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

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

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

Supplement Request for RAI No. 17 (FHA)

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

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

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

> Existing administrative controls for open containment airlock doors will be expanded to ensure specified individuals are designated and readily available to isolate any open penetration flow path(s) in the event of an FHA inside containment.

SNC's response to RAI No. 17 states:

> Item #2, above, is the provision for the isolation of the flow path, which is consistent with TSTF-312.
>
> Under SNC's LLRT [local leak rate test] procedures, personnel are stationed at the containment penetration being tested. Therefore, if an FHA were to occur at the same time an LLRT is being conducted, the LLRT personnel would be immediately available to isolate the penetration.
>
> The proposed change also includes the addition of text to the LCO [limiting conditions of operation] discussion in Bases 3.9.3 stipulating that the administrative controls that are put in place when penetrations flow path(s) are unisolated ensure that: (1) appropriate personnel are aware of the open status of the penetration flow path during core alterations or movement of irradiated fuel assemblies within the containment, and (2) specified individuals are designated and readily available to isolate the flow path in the event of an FHA.

TSTF-312 and NUREG-1431 states:

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

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

Supplement Request for RAI No. 25 (CREA)

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

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

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

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

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

> Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.
>
> Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.

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

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

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

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

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

Supplement Request for RAI No. 31 (CREA)

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

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

SNC's response to RAI No. 31 states:

> RG 1.183, Appendix H, regulatory position 1 indicates that of the iodine contained within the fraction of core that experiences fuel melt, 25% is available for release from the containment and 50% is available for release from the reactor coolant. To simplify the dose analysis for Farley, it is assumed that 50% of the iodine in the melted fuel is available for release from both the reactor coolant and the containment. To offset this conservatism of a factor of two (50% instead of 25%) in containment release, the radial peaking factor of 1.7 is not applied to the melted fuel. While this is non-conservative for the reactor coolant source, which is eventually released to the environment via the secondary system, the dose contribution from this pathway is small compared to the containment release because the secondary release terminates at 225 sec while the containment release continues for 30 days...

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

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

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

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

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

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

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

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

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

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – REQUEST FOR ADDITIONAL INFORMATION (CAC NOS. MF8861, MF8862, MF8916, MF8917, MF8918, MF8919) DATED JULY 21, 2017

DISTRIBUTION:
PUBLIC
LPL2-1R/F
RidsACRS_MailCTR Resource
RidsNrrDorlDpr Resource
RidsNrrDorlLpl2-1 Resource
RidsRgn2MailCenter Resource
RidsNrrLAKGoldstein Resource
RidsNrrDe Resource
RidsNrrPMFarley Resource
KBuschltz, NRR
DKi, NRR
CTilton, NRR
SSun, NRR
RHarvey, NRO
JWhite, NRO
SJones, NRR
JBettle, NRR

ADAMS Accession No. ML17194A787 *by e-mail

OFFICE	DORL/LPL2-1/PM	DORL/LPL2-1/LA	DRA/ARCB/BC*	DORL/LPL2-1/BC	DORL/LPL2-1/PM
NAME	SWilliams	KGoldstein	KHsueh	MMarkley	SWilliams
DATE	07/21/17	07/14/17	07/19/17	07/21/17	07/21/17

OFFICIAL RECORD COPY