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Joseph M. Farley Nuclear Plant
License Amendment Request to Revise Technical Specification
Section 5.5.17 "Containment Leakage Rate Testing Program"
Responses to NRC Requests for Additional Information

Ladies and Gentlemen:

On November 15, 2016, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) to revise the Joseph M. Farley Nuclear Plant, Unit 1 and Unit 2, Technical Specifications (TS) 5.5.17, "Containment Leakage Rate Testing Program." On March 15, 2017, the Nuclear Regulatory Commission (NRC) staff, upon a determination that additional information was needed to complete its review, issued a request for additional information (RAI) letter. Enclosed is a partial response to the RAI. These responses stand alone and will not be impacted by the answers to the remaining RAI requests. A response to the remaining RAI requests will be provided by August 31, 2017.

Also enclosed is a new marked-up TS page and clean copy TS page for the changes to TS 5.5.17. Since SNC submitted the original LAR, SNC has implemented Amendment No. 203 and Amendment No. 199 for Farley, Units 1 and 2, respectively, which revised TS 5.5.17. See ML 15233A448. Consequently, SNC is submitting an updated marked-up and clean copy TS page to incorporate the proposed changes in this LAR into the TS page most recently implemented.

Amendments 203/199 allow the performance of the visual examinations of the containment pursuant to ASME Code Section XI, Subsections IWE and IWL, in lieu of the visual examinations performed pursuant to RG 1.163. Therefore, the visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of frequency specified by the ASME Section XI Code, Subsection IWL. In addition, the visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE. Supplemental inspections will not be required. This information provides an update to Section 3.4.3 of the original LAR.

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

Mr. J. J. Hutto states that he is the Regulatory Affairs Director for SNC, is authorized to execute this oath on behalf of SNC and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

J. J. Hutto
Regulatory Affairs Director

JJH/efb/lac

Sworn to and subscribed before me this 22 *day of* June, *2017.*

Laura L. Crumpton
Notary Public

My commission expires: 10-8-2017

Enclosures:
1. Responses to NRC Request for Additional Information
2. Proposed TS 5.5.17 - Marked-Up TS Page
3. Proposed TS 5.5.17 – Re-Typed (Clean Copy) TS Page

cc: NRC Regional Administrator, Region II
NRC NRR Project Manager – Farley
NRC Senior Resident Inspector – Farley
SNC Records RTYPE: CFA04.054

Joseph M. Farley Nuclear Plant
License Amendment Request to Revise Technical Specification Section 5.5.17
"Containment Leakage Rate Testing Program"
Responses to NRC Requests for Additional Information

Enclosure 1

RAI 1

Regarding Attachment 1, Table 5-7: "Farley Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT [Integrated Leak Rate Test] required 1/10", please validate the value for Accident Class 1.

Under EPRI Methodology, Table 5-7 has a value of 9.63E-06/ year (yr), however, NRC staff calculated a value of 9.37E-6/yr, by taking the corresponding Class 1 value from Table 5-5 of 1.01 E-5/yr and subtracting both Accident Class 3a and 3b and multiplying the result by 3.333:

NRC Calculation:
Class 1 value from Table 5-5 - (3a value from Table 5-5 + 3b value from Table 5-5) * 3.333

1.01 E-5/yr- (1.74E-7/yr + 4.36E-8/yr) * 3.333 =should be 9.37E-6/yr instead of 9.63E-6/yr

It appears that SNC may have only subtracted Class 3a instead of 3a and 3b.

This same calculation was made in Class 1 on Table 5-7 for both Electric Power Research Institute (EPRI) Methodology and EPRI Methodology Plus Corrosion.

These results appear to be carried forward in Tables 5-8, 5-9, 5-10 and carried through in subsequent calculations.

- Table 5-8 value of 1.02E-5/yr for Accident Class 1 Frequency (for both EPRI and EPRI Corrosion). NRC estimates 1.00E-5/yr.

- Table 5-9 value of 9.26E-6/yr for Accident Class 1 Frequency (for both EPRI and EPRI Corrosion). NRC estimates 9.01 E-6/yr.

- Table 5-10 value of 9.89E-6/yr for Accident Class 1 Frequency (for both EPRI and EPRI Corrosion). NRC estimates 9.70E-6/yr.

Please validate the results in submittal dated November 15, 2016, and, if correct as originally provided, provide sufficient detail on how the values were obtained. If an error was made, please re-calculate and re-submit with the revised values.

The NRC staff notes that when the revised calculated values are corrected for the apparent error described above, the results in Section 5.5 Step 5 to values for ΔCCFP for both units appear to exceed the threshold for "small" increases as per Regulatory Guide (RG) 1.174. For instance instead of the change in CCFP of 0.92% for extending the test interval to fifteen years from the original three in ten years for FNP Unit 1 and Unit 2; NRC staff calculates values of 2.06% for Unit 1 and 2.07% for Unit 2, in excess of the acceptance maximum of 1.5%. (NRC staff also calculated exceedances for the change from three in ten years to ten years for both units, albeit by lesser amounts). For instance:

Unit 1

NRC calculation: $CCFP_{3\ Unit\ 1}$ = [1 – (1.01E-5/yr + 1.74E-7/yr)/1.91E-5/yr] * 100% = 46.2%
SNC calculation: $CCFP_{3\ Unit\ 1}$ = 46.02%

NRC calculation: $CCFP_{10\ Unit\ 1}$ = [1 – (9.37E-6/yr+5.81E-7/yr)/1.91E-5] * 100% = 47.9%
SNC calculation: $CCFP_{10\ Unit\ 1}$ = 46.56%

NRC calculation: $CCFP_{15\ Unit\ 1}$ = [1 – (9.01 E-6/yr + 8.72E-7/yr)/1.91 E-5] * 100% = 48.26%
SNC calculation: $CCFP_{15\ Unit\ 1}$ = 46.94%

NRC calculation: $\Delta CCFP_{Unit\ 1} = CCFP_{15} - CCFP_3$ = 48.26% - 46.2% = 2.06%
NRC calculation: $\Delta CCFP_{Unit\ 1} = CCFP_{15} - CCFP_{10}$ = 48.26% - 47.9% = 0.36%
NRC calculation: $\Delta CCFP_{Unit\ 1} = CCFP_{10} - CCFP_3$ = 47.9% - 46.2% = 1.7%

Unit 2

NRC calculation: $CCFP_{3\ Unit\ 2}$ = [1– (1.07E-5/yr+1.60E-7/yr)/1.75E-5/yr] * 100% = 37.94%
SNC calculation: $CCFP_{3\ Unit\ 2}$ = 38.02%

NRC calculation: $CCFP_{10\ Unit\ 2}$ = [1– (1.00E-5/yr+5.32E-7/yr)/1.75E-5] * 100% = 39.82%
SNC calculation: $CCFP_{10\ Unit\ 2}$ = 38.55%

NRC calculation: $CCFP_{15\ Unit\ 2}$ = [1– (9.70E-6/yr+7.99E-7/yr)/1.75E-5] * 100% = 40.01%
SNC calculation: $CCFP_{15\ Unit\ 2}$ = 38.93%

NRC calculation: $\Delta CCFP_{Unit\ 2} = CCFP_{15} - CCFP_3$ = 40.01 % - 37.94% = 2.07%
NRC calculation: $\Delta CCFP_{Unit\ 2} = CCFP_{15} - CCFP_{10}$ = 40.01 % - 39.82% = 0.19%
NRC calculation: $\Delta CCFP_{Unit\ 2} = CCFP_{10} - CCFP_3$ = 39.82% - 37.94% = 1.88%

Therefore, when recalculating for the apparent error as described above, explain and provide justification for exceeding any acceptance thresholds.

SNC Response

SNC provides the following response to RAI 1:

Regarding Attachment 1, Table 5-7: "Farley Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT [Integrated Leak Rate Test] required 1/10", SNC has validated that the value in Table 5-7 of 9.63E-06/ year for Accident Class 1 is correct, and no recalculations are needed.

SNC offers the following explanation:

Both the NRC and SNC used the EPRI methodology[1] of calculating the impact of extending the ILRT interval from 3 to 10 years by multiplying the values in Table 5-3 calculated for Containment Release Types 3a and 3b by the number 3.333. The baseline value of 1.01E-5 /yr for Accident Class 1 is calculated by subtracting the values for Class 3a and Class 3b from the Adjusted Frequency for Intact Containment as follows:

Class 1 value from Table 5-2 – (Class 3a value from Table 5-3 +Class 3b value from Table 5-3) or 1.04E-5/yr – (1.74E-07/yr + 4.36E-08) = 1.01E-5/yr. Therefore, the value of Accident Class 1 reflects the subtraction of Class 3a and Class 3b from Class 1. Additionally, the value of the sum of all containment event tree (CET) end states must be the same as core damage frequency.

These values were correctly carried forward to Table 5-5, where they are reflected in the column for Accident Class Frequency using the EPRI methodology. The value of the sum of all CET end states is the same as core damage frequency, so core damage frequency is preserved.

In Table 5-7, the EPRI methodology reflects the impact of the longer test interval by multiplying the values for the frequency of Class 3a and Class 3b by 3.333:

Class 3a value from Table 5-5 * 3.333 = Class 3a value in Table 5-7 or 1.74E-7/yr *3.333 = 5.81E-7.

Class 3b value from Table 5-5 * 3.333 = Class 3b value in Table 5-7 or 4.36E-08/yr *3.333 = 1.45E-7.

As the NRC concludes in the RAI, when these values are subtracted from the Class 1 value from Table 5-5 of 1.01E-5/yr, the result is 9.37E-6/yr.

However, when the value of 9.37E-6 is entered in Table 5-7, and the sum of Accident Classes 1 through 8 is calculated, this value does not match the core damage frequency of 1.91E-5/yr.

Specifically, the sum of Accident Class 1, 2, 3a, 3b, 7 and 8 values from Table 5-7 is: 9.37E-6/yr + 3.52E-8/yr + 5.81E-7/yr+1.45E-7/yr+8.61E-6 + 1e-7/yr = 1.89E-5/yr. And, this total release frequency does not equal the CDF, thus the CDF is not preserved. Recalling that the value of Accident Class 1 in Table 5-5 of 1.01E-5 already reflects the subtraction of the sum of Class 3a and Class 3b, to correct for this, the multiplication factor applied in Table 5-7 must be reduced by one, to account for the one "set" of Class 3a + Class 3b frequencies already applied to the original Class 1 frequency of 1.04E-5 presented in Table 5-2.

The following is a comparison of the NRC calculation and the SNC calculation:

<u>NRC Calculation:</u>

Class 1 value from Table 5-5 - (3a value from Table 5-5 + 3b value from Table 5-5) * 3.333

[1] Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325. EPRI, Palo Alto, CA: 2008. TR 1018243.

1.01E-5/yr – (1.74E-7/yr + 4.36E-8/yr) * 3.333 = 9.37E-6

SNC Calculation:

Class 1 value from Table 5-5 – (3a value from Table 5-5 + 3b value from Table 5-5) * (3.333-1)
1.01E-5/yr - (1.74E-7/yr + 4.36E-8/yr) * (3.333-1) = 9.63E-7/yr.

Using the SNC calculation, CDF is preserved. The results match those in the LAR, and the conclusions of the LAR remain unchanged.

RAI 2

In Section 4.2.4, Population Dose Estimate Methodology, FNP was compared with a reference plant, Surry. Explain if the weather conditions around the two sites (Surry vs FNP) are sufficiently comparable to render this scaling approach conservative based on power level, leakage and population. Please provide justification.

SNC Response

SNC provides the following response to RAI 2:

The EPRI methodology in its submittal template and the SNC LAR state plainly that meteorology and site differences other than population are assumed not to play a significant role in the evaluation. The EPRI methodology discusses the development of baseline population dose in Section 4.2.2 of the Technical Approach "Step Two: Develop the Baseline Population Dose". Scaling of a reference plant population dose was chosen because plant-specific PRA dose rates for Farley were not available. EPRI selected Surry as the representative for the PWR. Surry and Farley are PWRs with similar containment designs that experience similar weather conditions.

Development of baseline population dose in Section 4.2.2 of the EPRI Technical Approach "Step Two: Develop the Baseline Population Dose," does not discuss the significance of the impact of the meteorological conditions between the representative plant and the submitting plant. The methodology focuses on the population density, power level, and the allowed leakage as the relevant driving factors. Because both Farley and Surry are in areas considered by climatologists to be Humid Subtropical Climates (Köppen Cfa), the differences in weather can be considered secondary to the population density, power level, and allowed plant leakage.

Further, the impact of meteorology on the ILRT extension is negligible when compared with the definition of very small population dose of less than or equal to 1.0 person-rem per year increase.

In conclusion, the weather conditions around the two sites (Surry and Farley) are sufficiently comparable to render the scaling approach conservative based on power level, leakage, and population.

RAI 3

Please address the following questions associated with Section A.2.5, "Assessment of PRA [Probabilistic Risk Assessment] model Technical Adequacy":

a. In Table 1, "Resolution of the Farley PRA Peer Review F&Os [Fact and Observations] Associated with the 17 Not Met SRs"; the peer review finding for F&O HR-G7-01 & 02 addresses the licensee's treatment of dependency between multiple human actions. Please indicate if a specific floor value was defined *(e.g.* via post-processing) to ensure scenarios (cutsets) containing multiple human failure events/human error probabilities (HFEs/HEPs) did not result in joint HEPs (JHEPs) within the scenario (cutset) below a minimum threshold, that being 1 E-6 for internal events. If any cutsets resulted in joint HEPs lower than 1 E-6, provide a sensitivity evaluation of imposing such a minimum value and address whether this affects the conclusions drawn in the application.

b. Confirm that the response to a similar request for additional information (RAI) under National Fire Protection Association Standard (NFPA)-805 specific to the fire PRA, where use of a floor value of 1.0E-5 for any JHEP combinations with values less than 1.0E-5 was cited, remains valid and applicable for this ILRT application. This was RAI PRA 26e answered by RAI response letter dated October 30, 2013 (ADAMS accession number ML 13305A105).

SNC Response

SNC provides the following response to RAI 3:

a. A floor value of 1.0E-6 was used for HRA Dependency Analysis in the Farley internal events PRA. The floor value was defined in a recovery rule file for any human failure event combinations from which the probability was lower than 1.0E-6.

b. The floor HEP value of 1.0E-5 remains valid and applicable for this ILRT application.

RAI 4

The submittal did not include the complete list of Internal and External PRA Peer Review F&Os/ Findings and dispositions/resolutions. NRC staff recognizes that Table 1 of Appendix A included 17 of the "SR (Supporting Requirement) Not Met" findings and dispositions/resolutions after the peer review.

Normally the NRC expects all PRA F&Os to be submitted for License Amendment Requests. To the extent that the PRA Peer Review F&Os and resolutions/dispositions from the FNP NFPA-805 submittal dated September 25, 2012, remain applicable for this ILRT application we will treat these as part of the submittal. However, if there have been any changes in the dispositions/resolutions, including ones resulting from the processing of the NFPA-805 submittal, that could affect this ILRT submittal, or any self-assessments or peer reviews that

have generated new Findings since the NFPA-805 submittal, provide these for review. Include their dispositions/resolutions. If none, please confirm.

SNC Response

SNC provides the following response to RAI 4:
No self-assessment or peer reviews have been performed for Internal Events (including Internal Flooding) and Fire PRA models since the NFPA-805 submittal. Therefore, new Findings have not been generated for these models since the NFPA-805 submittal.

Fire PRA:
The disposition of the Fire PRA Findings was not changed as a result of processing of the NFPA-805 submittal. Therefore, reference made to the NFPA 805 submittal remains valid for the Fire PRA model.

Internal Events (including Internal Flooding) PRA:
Because several peer review findings for the Plant Farley Internal Events (including Internal Flooding) PRA were characterized as "pending" in the NFPA 805 submittal and because these findings have been dispositioned after the NFPA 805 submittal, all findings and their resolution are provided in response to RAI 04. The following table and appendix summarizes Findings (and associated resolutions) resulting from the Peer Review of the Plant Farley Internal Events (including Internal Flooding) PRA Model. NRC requires verbatim F&O Description as provided by the peer review team. Therefore, SNC has not made any attempts to make editorial changes or clarify the F&O description.

With the exception of Finding DA-C14-01, the disposition of Findings is reflected in the Revision 9 Version 3 of the PRA model. As part of the model maintenance practice, a comprehensive data update was performed in 2015 and was incorporated in the Revision 10 of the Internal Events PRA model. The Finding DA-C14-01 is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
AS-C2-01	Finding	Closed	In the discussion of large, medium, and small Loss Of Coolant Accidents (LOCAs), the operator failure to transfer to low head recirculation is discussed. For large LOCAs, this error is OAR_A_1-----H, and for other LOCAs (or event trees) the error is OAR_A_2------H. The only difference between the two errors is timing. However, the discussion of OAR_A_2------H indicates that the operator must manually align Component Cooling Water (CCW) cooling to the Residual Heat Removal (RHR) heat exchanger. The discussion of OAR_A_1------H does not include the requirement for the operator to realign CCW to the RHR heat exchanger. The two errors appear to have been modeled correctly, but the difference in the description in the AS notebook is confusing. Add the discussion of the operator realigning CCW to the RHR heat exchanger to the description of OAR_A_1------H.	The description for OAR_A_1------H in the Accident Sequence notebook was revised to note that "operator action is still required to align CCW cooling to the RHR heat exchanger" to be consistent with the description of OAR_A_2------H . This finding is considered closed.
AS-C2-02	Finding	Closed	Table 2.6-1 of the Farley AS notebook identifies events %LOSSACF and %LOSSACG as Loss of Power to 4kV Bus F and Loss of Power to 4 kV Bus G, respectively. However, the table in Section 2.6.4 identifies these events as Loss of 4160 V Bus F and Loss of 4160 V Bus G, respectively. These two events (Section 2.6.4) are not recoverable by the EDGs because of damage to the respective buses. In Section 2.6.4, the events Loss of Power to 4 kV Bus F and Loss of Power to 4 kV Bus G are labeled as %LOSPF and %LOSPG, respectively. Initiating events %LOSPF and %LOSPG are not included in Table 2.6-1. Table 2.6-1 is incomplete because it is lacking initiating events %LOSPF and %LOSPG.	The Accident Sequence notebook was revised to correctly reference the loss of bus initiating events. The descriptions of the %LOSSACF and %LOSSACG events were not changed because they are correct. Instead, the descriptions for those events were corrected and events %LOSPF and %LOSPG were

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			Table 2.6-4 incorrectly characterizes initiating events %LOSSACF and %LOSSACG. Add initiating events %LOSPF and %LOSPG to Table 2.6-1. Correct the descriptions of initiating events %LOSSACF and %LOSSACG in Table 2.6-4.	added. Documentation was revised. This finding is considered closed.
DA-C14-01	Finding	Closed	Several of the data sets used in the Farley database are based on information that is getting dated. The period over which these data were collected is 1984 through 2001, or earlier. The affected data sets include Table 4 (simultaneous maintenance on redundant equipment), offsite power recovery, and plant-specific data used for failure rates, probabilities, and unavailability. For RIR application, periodically updated plant specific data is required. These data sets need to be updated using more recent information.	The data were updated using more recent industry generic data and plant specific experience data. This finding is considered closed.
HR-D2-01	Finding	Closed	Farley develops detailed restoration errors for three events and applies this probability to most of the remaining events without any specific evidence through procedures or tests that the values are events are similar enough that the same values should apply. The values for these restoration errors could be significantly over-estimated since the value applied is not shown to be directly applicable to the event analyzed. Detailed analysis should only be applied to the event analyzed or to directly applicable events where procedures and actions are similar (SW pump trains with identical restoration type errors through similar procedures). Perform detailed analysis on all events to verify the applicability used or use	The HRA notebook was revised to add more detailed explanation of the approach used. The pre-initiator approach relies on detailed THERP assessments that are mapped to similar HFEs. This finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			screening values for those events not explicitly analyzed with a detailed analysis.	
HR-D2-02	Finding	Closed	The screening probability used for unanalyzed events is 1E-4. This is significantly lower than the base screening HEP from ASEP which is median failure rate of 3E-2. Even if credit is taken for a recovery factor such as post-maintenance testing or independent verification, then the screening value would be approximately 8E-3. The screening values used are significantly below the screening values recommended in Technique for Human Error Rate Prediction (THERP) and Accident Sequence Evaluation Program (ASEP). Review the Pre-accident HRA screening values that are used and be consistent with ASEP as discussed in the SR.	The HRA notebook was revised to add more detailed explanation of the approach used. The pre-initiator approach relies on detailed THERP assessments that are mapped to similar HFEs. This finding is considered closed.
HR-G1-01	Finding	Closed	The top HRA events in the QU notebook are not developed in the HRA notebook. Example 1RTOPMANRTNSGH and OMG_A_2-------H. These events appear in several of the top 50 cutsets and are thus significant to the risk assessment Develop HRAs for these events and include in the HRA calculation.	The events were included in the HRA calculator using the values found in NUREG/CR-5500 and WCAP-15831. The finding is considered closed.
HR-G7-01	Finding	Closed	The top HRA cutset combinations in the QU notebook are not addressed in the HRA dependency analysis. These events appear in several of the top 50 cutsets and are thus significant to the risk assessment Explicitly evaluate the top HRA combinations in the dependency analysis.	An HRA Dependency Analysis was conducted and incorporated into the model quantification. This analysis has been incorporated into the HRA notebook.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
				The finding is considered closed.
HR-G7-02	Finding	Closed	Attachment C to the HRA notebook performs the dependency assessment, but the dependency factors are based upon 2004 HRA values. The multiplication factors in the rule file are to be based upon current HRA. The recovery rules seem to address dependence with factors greater than one and only then for 5 events. This is not consistent with the dependence methods. Update the HRA dependence evaluation to be consistent with industry practices.	An HRA Dependency Analysis was conducted and incorporated into the model quantification. This analysis has been incorporated into the HRA notebook. The finding is considered closed.
HR-I3-01	Finding	Closed	Assumptions are listed in the individual HRA analyses. However, some major assumptions normally associated in an HRA analysis, such as default minimum values for pre- and post-accident HRAs, are not included in the analysis. In addition, uncertainty based on using the same HRA probability for all manual valve misalignments is ripe for an uncertainty evaluation. Also, the HRA calculation does not address the different types of uncertainty that is included in other Farley document packages. Review the EPRI report on HRA uncertainties and see if any will apply to Farley. Documentation of sources of uncertainty is required by the SR Include a source of uncertainty in the HRA calculation.	A document was created to address HRA Uncertainty for the Farley model. The finding is considered closed.
IE-A10-02	Finding	Closed	The Farley IE notebook indicates that failure of the Service Water (SW) pond dam was included as a special initiator. However, a search of the model did not	The information presented in the F&O Table Appendix (following

F&O #	Level	Status	Issue and Proposed Resolution.	Resolution
			locate the dam failure. Further, the probability of a loss of the SW pond dam is estimated to be 1.9e-7 failures per year based on the FNP River Water Study (dated 1982). This analysis is based on a generic estimate of 1.9e-5 failures per year for earthen filled dams that in the opinion of Alabama Power Company should be reduced to 1.9e-7 per year due to design, monitoring, maintenance, and responsiveness of the owner to problems. Loss of the dam would result in a dual unit loss of service water. For an event of the magnitude of a dual unit loss of service water, the supporting evidence for reduction of the generic value by a factor of 100 is treated very lightly. An initiating event that would result in a dual event initiator should be included in the initiating event portion of the model. Evidence for reducing the generic dam failure probability is qualitative in nature, and the extension of this information to justify a factor of 100 reduction in the generic probability is not clear and poorly supported. Further, dam failure analysis technology has improved since 1982, and use of the newer approaches to analysis should be considered. Consider revisiting the estimation of the probability of dam failure using newer technology and better supported calculation. Add the loss of the SW pond dam to the model, if appropriate.	this table) supports SNC's conclusions that loss of SW due to a random failure of the dam as an initiating event does not need to be modeled in the internal events PRA based on the screening criteria in IE-C6 (b) of the ASME PRA Standard. This finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
IE-A5-01	Finding	Closed	There is no evidence of a system by system review of the Farley systems to verify no additional initiators exist. A systematic review of the Farley systems and trains should be performed to ensure that all potential initiators are identified and that the initiators are grouped properly on the basis of impact and frequency. Add a systematic review of the safety and non-safety systems that could cause a plant scram to verify that no additional initiators are needed.	A systematic review of the Farley safety and non-safety systems was performed and documented in the Farley Initiating Event Notebook. The notebook lists each Farley system ordered by a system group identifier, system ID, system description, impact of system loss and treatment of system loss in Farley PRA. The "treatment of system loss" addressed specifically whether the loss of a system would result in an initiating event and how the initiating event was grouped. This finding is considered closed.
IE-A7-01	Finding	Closed	Section 2 states that events occurring during Modes 3 -6 are considered to determine if they are applicable at-power. Appendix B-1 includes events that occurred at power levels less than 10%. However, the review does not seem to look at the event applicability for Mode 1. Two of the reactor trips at 0% power were due to Source Range Monitors (SRMs). These events would not be applicable to the at-power analysis since the SRM would be replaced by the APRMs for Mode 1. Clarify the review of the LPSD events included in Appendix B-1 and how they	These two events were reviewed and it was determined that they should be removed from the plant specific frequency analysis. The Initiating Events Notebook was revised to reflect the changes to the analysis. This finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			are included in the plant specific frequency analysis.	
IE-A9-01	Finding	Closed	There is no indication that IE precursors such as intake clogging have been performed. Precursor reviews generally include a significant plant event that did not cause a scram but could have if prompt action is not taken. Review significant non-scram events at the plant to determine if any precursors exist.	A search was performed using the Condition Reports database for significant non-scram events. A comparison of the results was made to Farley's initiating events list. No new initiating event precursors to plant trips were found. Added methodology and review results in the Initiating Events notebook. The finding is considered closed.
IE-B1-01	Finding	Closed	Events are grouped in general categories. It is not clear that the impact on systems are similar or that the grouped event frequency includes these events Loss of Turbine Building Cooling is grouped with loss of Service Water. However, the frequency for these events is expected to be similar and may have different impact on the PSA systems. Other potential groupings, such as the 7300 bus and 4.16 KV buses identified through the operator interviews were not clearly grouped. In other cases, the review of the events from NUREG/CR-3862 and NUREG/CR-5500 are not directly tied to an initiating event class. Include the impact of the initiator	Table C-1 "Farley Initiating Event Identification Analysis" was created and documented in the Farley Initiating Event Notebook. This table lists each Farley system ordered by a system group identifier, system ID, system description, impact of system loss and treatment of system loss in Farley PRA. The treatment of

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			(especially the transient events) on the PSA systems in the model.	"system loss" addressed specifically whether the loss of a system would result in an initiating event and how the initiating event was grouped. This finding is considered closed.
IE-C1-01	Finding	Closed	In section 4 of the initiating event notebook, Farley discusses the quantification of the vessel rupture frequency. They present the WASH-1400 median frequency of 1E-07 with the associated error bounds but then proceed to treat that value as a mean. This is mathematically incorrect and introduces a slight non-conservative bias. It is not likely to impact the overall results. Calculate the mean from the median and error factor and use that in the quantification. (Mean should be about 2.7E-07.) There is also a newer generic source that has a better number.	Revised Reactor Vessel Rupture Frequency in the Initiating Events notebook and added a reference in the reference section to include a more current data source. This finding is considered closed.
IE-C15-01	Finding	Closed	Table 7 of the Farley Initiating Events Notebook presents the initiating event frequencies for the special initiators but does not characterize the uncertainty. The special initiators are quantified using fault tree analysis so the uncertainty intervals inherently can be quantified based on the uncertainty data for basic events. However, the variance is not presented and there is no discussion of this beyond stating that the frequencies are calculated using fault trees. This is a documentation issue. There is no indication that the uncertainty was not included in the overall model quantification. Document how the uncertainty for the	This is a documentation issue. As discussed in the finding, the uncertainty of special initiating event is evaluated during quantification process. The finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			special initiators was characterized/quantified as part of the discussion in section 3.3 of the Initiating Events Notebook	
IE-C5-01	Finding	Closed	Farley did calculate their initiating event frequencies on a reactor year basis. However, they did not modify the resultant frequencies to address plant availability. Discussions with the Farley staff indicated that the adjustment was not made as part of quantification either. The frequencies are slightly conservative. The initiating event frequency should be modified to address plant availability. This can be done by multiplying each initiating event frequency by the availability factor or the adjustment can be done as part of the quantification	The adjustment has been made as part of the model quantification. The Initiating Events notebook contains the development of the annual average availability factor. The finding is considered closed.
IE-D1-01	Finding	Closed	Farley did document their initiating event analysis. However, the structure and content of the documentation was such that it was often difficult to trace the identification, grouping and quantification of the IEs is an easy to follow manner. This issue was identified in virtually all Technical Elements of the Farley PRA. It was often difficult to determine what Farley had done to address a given SR and required detailed evaluation of the model and many discussions with the Farley PRA staff. One part of the problem was that in several places, the documentation reflected an earlier version of the model (Version 8 versus Version 9) or did not match the model (treatment of miscalibration errors). This made the PRA difficult to review. However, of greater concern, the documentation could only support applications or updates if a knowledgeable/experienced engineer was involved. This touches on virtually all PRA documents.	Many documents including initiating event notebook and documentation reflecting an earlier version have been updated since the peer review was performed. The finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			1. Ensure that the documentation reflects the latest version of the model. 2. Review the documentation to see if it has sufficient content and is structured such a less experienced engineer can understand the analysis.	
IFEV-B3-01	Finding	Closed	The Farley PRA flooding analysis indicates that sources of uncertainty were not documented because of the low contribution to CDF and LERF from flooding. Although this is true, this SR requires that sources of model uncertainty and related assumptions associated with the internal flood-induced initiating events be documented. The Farley PRA flooding analysis indicates that sources of uncertainty were not documented because of the low contribution to CDF and LERF from flooding. Although this is true, the SR requires that a discussion of uncertainty be provided. Include a discussion of uncertainty and assumptions related to internal flood initiating events. This finding is related to other internal flooding SRs that discuss documentation of uncertainty.	New text concerning uncertainty and assumptions has been incorporated into the appropriate sections of the Flooding notebook. The finding is considered closed.
IFPP-B2-02	Finding	Closed	The IF notebook provides descriptions about flood areas within four (4) buildings, such as auxiliary building, diesel building, service water intake structure, and turbine building. There is no description about the other buildings. Even though they are not risk-significant, the description about the reason why those buildings are not analyzed is	New text concerning screened/eliminated areas and buildings has been incorporated into the Section 3.1 of the Flooding notebook. The

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			needed. The screened/eliminated areas are not considered in the analysis. Possible resolution is to add information about the screened/eliminated areas and buildings in terms of internal flooding analysis.	finding is considered closed.
IFPP-B2-03	Finding	Closed	Even though Farley has areas that are common between both units the documentation of how multi-unit impacts were addressed could not be located. Discussions with the Farley PRA staff did reveal that Farley had considered multi-unit effects, However, the documentation of how Farley explicitly considered the potential for multi-unit floods is not well presented. The IF Notebook needs to be updated to address the potential for multi-unit floods or the propagation of a flood in one unit to the other unit via shared spaces. Farley needs to explicitly describe how they dealt with the evaluation multi-unit effects for areas where there shared spaces. The basis for any screening of such areas should be explicitly described in the text as well as in the screening table.	New text concerning multi-unit impacts has been incorporated into the Section 3 and 12.5 of the Flooding notebook. The finding is considered closed.
IFQU-A11-01	Finding	Closed	Internal Flooding Analysis Notebook Appendix A does not describe the information related with human reliability analyses and screening decisions. Possible resolution is to provide the room for 1) operator mitigation action and 2) reason of screening decisions. Supplemental Comments: In according to ASME Standard, IFQU-A11, human actions (and human reliability analysis) modeled for each flood area's quantification are verified via flood walk downs. Also, the reason of screening decision should be verified via walk downs.	Although qualitative screening is documented in Table 6-1 to 6-4 of the Internal Flooding notebook, the tables did not include any human actions. Section 6 of the Internal Flooding notebook lists the screening criteria which includes human mitigating actions

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			Proposed resolution: The walk down sheet for each flood area add two more sections as follows: G. related human actions H. Screening Decision In case of screening, table 6-1 in the notebook would be a good reference. In case of HRA, table 10-2 and 10-3 would be a good reference.	as a criterion (criterion d.). However, most of the flood locations in Tables 6-1 to 6-4 were qualitatively screened based on criterion a or b. None were screened on criterion d, which shows that although human actions were considered as a screening criterion, there were no applicable areas in the Farley flooding PRA. The finding is considered closed.
IFQU-A6-01	Finding	Closed	HRA for flooding event was performed but the base is different from internal event HRA. It seems that there is version mismatch. Possible resolution is to update the HRA for flooding events like as HRA for internal events.	PRA-BC-F-10-004 describes the HRA methodology for flooding PRA and flooding human failure events were added to the HRA Calculator database. The finding is considered closed.
IFQU-A7-01	Finding	Closed	Quantification of flooding event does not perform uncertainty analysis and dependency analysis. Section 10.1.7 explains the dependencies between human interactions, and Farley performed dependency analysis when quantifying the flood CDF. However, there is no description of calculation results about the dependencies. Technical Items are missing. Possible resolution is to perform and provide uncertainty analysis and	An HRA Dependency Analysis was conducted and incorporated into the model quantification. This analysis has been incorporated into the HRA notebook. The finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			dependency analysis, even though the flood risk is not significant.	
IFQU-B3-01	Finding	Closed	The Farley PRA flooding analysis indicates that sources of uncertainty were not documented because of the low contribution to CDF and LERF from flooding. Although this is true, this SR requires that sources of model uncertainty and related assumptions associated with the internal flood-induced initiating events be documented. The Farley PRA flooding analysis indicates that sources of uncertainty were not documented because of the low contribution to CDF and LERF from flooding. Although this is true, the SR requires that a discussion of uncertainty be provided. Include a discussion of uncertainty and assumptions related to internal flood initiating events. This finding is related to other internal flooding SR that discusses documentation of uncertainty.	New text concerning uncertainty and assumptions has been incorporated in the Flooding notebook. The finding is considered closed.
IFSN-A2-01	Finding	Closed	The flood analysis does discuss the potential effect of: alarms, structure such as curbs and sumps, drains, sump pumps, watertight doors; However, any direct application of these factors was hard to find. The factors most often explicitly credited was the credit for jacketed piping eliminating spray considerations and air/water tight doors stopping propagation. The remarks column in table 7-1 does seem to reference hatches as propagation paths but it is not clear that impact of drains and curbs or the like were considered for propagation. Farley should update the flood documentation to provide more information on plant features that can	New text has been incorporated in the Flooding notebook. The finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			impact the propagation or retention for each flood scenario, especially anywhere that non-watertight doors, berms or curbs are credited	
IFSN-A4-01	Finding	Closed	In the IF Notebook, there was extensive discussion with respect to treatment of drains, there was explicit evidence that drains were considered as propagation paths for several flood scenarios. However, no explicit estimation of drain capacities could be found. This is a direct violation of SR. Farley should consider adding a table that explicitly includes drain capacities.	New text has been incorporated in the Flooding notebook. The finding is considered closed.
IFSN-B3-01	Finding	Closed	The IF Notebook did not seem to include assumptions related to the flood scenario selection in a coherent fashion nor did there seem to be any discussion concerning sources of uncertainty. Farley needs to include a section in the IF Notebook to discuss the IF assumptions and sources of uncertainty. A section on assumptions could be included in each section (such as was done for section 9 and 11) or a single section encompassing all tasks could be added	New text concerning uncertainty and assumptions has been incorporated the Flooding notebook. The finding is considered closed.
MU-B4-01	Finding	Closed	There is no reference to a peer review for upgrades. Did not find a section which addressed upgrades (not updates) to the PRA specifically involving changes to key PRA software. This is a direct violation of an SR. Revise either NL-PRA-001 or NL-PRA-002 to explicitly require a peer review for PRA upgrades (i.e. methodology change or major software change etc.)	Department procedure was revised to require a peer review following an upgrade of the PRA model. The finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
QU-F1-01	Finding	Closed	The maintenance related mutually exclusive events are stated to be based on Tech Spec disallowed maintenance conditions. The mutually exclusive logic was based upon FNP-0-ACP-52.1 but was not referenced as the source of the mutually exclusive logic. The review of the QU notebook referenced the incorrect document the development of the mutually exclusive logic. Update the documentation to reflect the actual references.	Documentation in reference has been updated in the Quantification Notebook. The finding is considered closed.
QU-F4-01	Finding	Closed	The information regarding the assumptions and sources of uncertainty is located in Appendix D of the Farley QU notebook. However, this appendix is not referenced in the QU notebook, neither is it included in the notebook's table of contents. The only reference to the appendix is in the Revision 9 Roadmap and Quality Self-Assessment document, and in this document, it is misidentified as Appendix A. References to this document are either non-existent or incorrect. Even though the document contains a lot of good information, it is almost impossible to locate. Correct the QU notebook table of contents to include Appendix D and its title. Add information to Section 2 that references the appendix. Correct the reference to the appendix in the Revision 9 Roadmap.	Added Appendix D to the Quantification Notebook. Corrected references to the Appendix in the Roadmap. This finding is considered closed.
SC-A2-01	Finding	Closed	The maximum core temperature of two cases of Medium LOCA (CL3-MLO-S2 and CL5-MLO-S1) exceeds 1800F early times after accident, but they are considered as success. In the MAAP analysis notebook describes "only exceeded 1800°F for less than 6 min; considered success." (Appendix B, Table B-1). In addition, there are two SGR cases (S1 and S2) in which the	While the core damage criteria of 1800 °F was exceeded for a short period of time, these 2 MAAP cases are considered successful pertaining to the

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			core temperature is oscillating unstably, exceeding 1800 F in some of the later oscillations. It is not clear that these or successes or that a stable configuration has been achieved. It is not clear that the identified cases cannot meet the success criteria for core damage. First possible resolution is to perform analysis using another tool instead of MAAP (e.g., a more detailed model that would allow a higher core damage temperature as a success criterion) for these two cases. Second one is to describe the details in the notebook why the analyst assumes these cases as success.	core damage success criteria. The Success Criteria notebook has been revised to address the maximum core temperature of two cases of Medium LOCA (CL3-MLO-S2 and CL5-MLO-S1). The finding is considered closed.
SC-A5-01	Finding	Closed	There are two SGR cases, S1 and S2, for which the maximum core temperature is oscillating wildly beyond 24 hours, sometimes exceeding 1800 F. These cases are evidently considered as successes, though it is not evident that a stable configuration has been reached at 24 or even 30 hours. In addition, there are cases for which the mission time is listed as less than 24 hours without explanation. Either do additional calculations to show the two SGR cases are successes or provide adequate explanation of why they are considered successes and a stable condition has been reached. In addition, provide additional explanation of the mission times that are shorter than 24 hours.	MAAP analysis was performed using MAAP 4.0.8 to address two SGR cases. The results show that the maximum core temperature did not oscillate and exceed 1800 °F for the cases. The finding is considered closed.
SC-B3-01	Finding	Closed	The current success criteria for LOCAs are based on plant capabilities and system responses. Although the definitions for small, medium and large break LOCAs are reasonable based on this criteria, the specific break sizes associated with the transitions between the LOCA definitions have not been	MAAP analyses were performed for a 6" break LOCA which is a lower end of large LOCA spectrum and upper end of the medium LOCA spectrum.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			adequately justified. Currently the break sizes are based on the original IPE criteria and no thermal hydraulic analyses of the break sizes have been performed. Per the requirement, thermal hydraulic evaluations are required at a level of detail to support the definitions/break sizes so that the appropriate initiating event frequencies can be determined. Several utilities' PRAs were dramatically impacted when the MAAP code was used to determine actual break sizes and some utilities determined that an additional fourth size LOCA was required to adequately model their plant. This has the potential to dramatically impact the CDF. Supplemental Comments: This comment is a general comment on thermo-hydraulic analysis for Farley. More plant-specific analysis would be required. According to your notebook, break sizes for MAAP analysis are as follows: - Large LOCA : 8.25 ft2 (about 39 in diameter) - Medium LOCA : 2.18E-02 ft2, 4.91E-02ft2, 1.36E-01ft2 (2 in, 3in, 5 in diameter) - Small LOCA : 7.64E-04ft2, 5.45E-03ft2, 2.18E-02ft2 (0.37 in, 1 in, 2 in diameter) The above break sizes are different from NUREG/CR-6928. Furthermore, they do not appear to explicitly cover the full range of potential LOCAs (from 5 inches up to 39 inches does not appear to be explicitly addressed). According to NUREG/CR-6928, the break sizes for LOCA are defined as follows: - LLOCA : greater than 6 inches inside diameter (D.2.2) ---> about 0.2 ft2 - MLOCA : between 2 and 6 inches	The MAAP analyses shows that the LOCA is able to mitigated by either medium LOCA success criteria or large LOCA success criteria. The finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			inside diameter (D.2.4) ---> about 0.02 ft2 ~ 0.2 ft2 - SLOCA : between 0.5 and 2 inches inside diameter (D.2.19) ---> about 0.00005 ft2 ~ 0.02 ft2 The success criteria change for the different break classes but there is no analysis to show that the success criteria are appropriate for both the upper and lower end of the break spectrums. For example, the primary difference between LLOCA and MLOCA is typically the number of accumulators required and possibly the number of pumps required. The primary difference between MLOCAs and SLOCAs is that secondary side heat removal is needed for small LOCAs. However, the MAAP analyses do not show that for LOCAs greater than 2 inches, the break is sufficient to remove decay heat while below 2 inches secondary heat removal is required. More and appropriate selection of break size would be required, such as 6 inches, 0.5 inches, etc. Develop LOCA break sizes based on Farley specific flow capacities and required systems.	
SC-B5-01	Finding	Closed	This SR requires that the reasonableness and acceptability of the SC results be verified. Although there was a table added the Success Criteria (SC) notebook (during the last few days prior to the peer review) that compares the SCs for Farley to SCs for Summer and Turkey Point, there was no text discussing the table, how the comparison was done, and the reasonableness/acceptability of any differences between Farley and either Summer or Turkey Point. This is a documentation issue rather than a technical issue since the comparison was apparently done. However, there is no basis in the documentation to	New text concerning the reasonableness of the SCs has been incorporated the Success Criteria notebook. The finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			determine whether the work to actually verify the reasonableness of the SCs was completed in accordance with the intent of the standard. Add discussion to the system notebook that references Table B and, at least at a high level, explains how the comparison was done and what was done if differences were found. At least, provide a couple of examples to illustrate this process.	
SY-A23-01	Finding	Closed	The system model nomenclature did not consistently use the fault tree guideline definitions in the naming convention. Examples include: guide has FW as feedwater system but model uses MF as system designator, RF component type identifier is not match the guide, room coolers are modeled with the system supporting. The room cooler system designator is the same as the ECCS pump. Farley should review their naming convention and make sure it is applied consistently in all models.	The naming conventions have been updated in the Farley Fault Tree Analysis Guidelines notebook. Specifically, the identifier FW was changed to MF for Main Feedwater and the component description for identifier RF was changed to Refrigeration Unit. The finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
SY-A8-01	Finding	Closed	In the diesel generator model, the diesel generator, the output breakers, the fuel oil transfer pumps, the sequence relays and the Local Control Panel are all modeled individually. However, Farley uses NUREG/CR-6928 as the source of their generic diesel generator data and collects plant specific data in accordance with the 6928 component boundaries. The NUREC/CR-6928 diesel generator component boundary explicitly includes the output breaker and the fuel oil system (without much definition). Thus, the component boundaries as used in the Farley diesel generator system model do not match the component boundaries used for collecting the failure data. Furthermore, the component boundaries used to derive the generic common cause boundaries do not match the component boundaries used to develop the generic failure rates. For the most part, Farley has made the appropriate adjustments to match the two divergent data sets. However, the generic common cause data for diesel generators had an event whose description was such that it could be interpreted as either involving fuel oil transfer pumps or not. The decision was made to include the event as a diesel failure because it would be conservative. The component boundary definitions in the Systems and Data Analysis Notebooks were not very detailed so this was difficult to identify. Farley needs to adjust their data collection and quantification to collect and quantify the diesel generator system failure data consistent with how the system is modeled. Farley also needs to review their component boundary definitions to ensure that they are sufficiently detailed to identify exactly	The modeling approach is valid because: i) The modeled fuel oil transfer pumps are external fuel oil pumps to makeup day tanks, which are not sufficient to supply fuel oil to DGs for 24 hours mission time. The pumps are required to makeup fuel oil from storage tank. ii) DG Output circuit breaker, sequence relays, and the Local Control Panel are modeled separately because three of five Farley DGs are shared by two units. Even though explicit modeling of the circuit breaker is somewhat conservative, the proper dependency model is required. The finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution
			what is included within each component and that are consistent from the model to the system notebooks to the data analysis notebook to the common cause failure analysis.	
SY-A9-01	Finding	Closed	The system model boundary is not clearly defined between the notebook and the model. Example is room cooling for Emergency Core Cooling System (ECCS) system is model as part of the system but is listed as a dependent system to the ECCS. AFW discussion of boundary includes condensate tank and steam supply up to steam generators, but later in the notebook defines condensate and steam supply as support systems. See also SY-A8-01 for diesel boundary issues.	The system notebooks were reviewed and modified as needed to reflect the boundary of the system as shown in the model. The support system sections were reviewed and corrected as needed to reflect the support systems as modeled. The finding is considered closed.

F&O #	Level	Status	Issue and Proposed Resolution	Resolution .
SY-B6-01	Finding	Closed	The room heatup calculations for the Engineered Safety Features (ESF) pump rooms and ESF electrical equipment rooms are excellent. But some calculation results are mismatched with documents and references, the others are conservatively applied into fault tree model. Description of Ref.12 and HVAC system notebook are mismatched with Ref.4. The calculations results show the temperature of the ESF equipment rooms during 30 days after loss of Heating, Ventilation and Air Conditioning (HVAC) condition. Some document errors occurred using 30-day calc. results. If the calc. results for ESF pump rooms and electrical equipment rooms would be checked for 24 hours, temperature of some rooms will be lower than the limit. If then, the system fault trees do not develop "room cooling failure" any more for those cases. Descriptions should be matched.	This is a documentation issue. The references were corrected. The model was checked for conservative room cooler failure modeling as a result of interpretation of the calculation results. The HVAC model for the Engineered Safety Features (ESF) pump rooms and ESF electrical equipment rooms were updated based on up-to-date room heatup calculations. This finding is considered closed.
SY-C1-01	Finding	Closed	In section 6.1.7 of the system notebooks for AFW, CCW, Containment Cooling, Containment isolation, Containment Spray, ECCS, IA, MS, SW incorrect reference information to test and maintenance is provided. Farley needs to correct the references for test and maintenance information.	This is a documentation issue. The references were corrected. This finding is considered closed.

F&O Table Appendix

F&O IE-A10-02 Service Water Pond Dam Failure

F&O IE-A10-02

"The Farley IE notebook indicates that failure of the Service Water (SW) pond dam was included as a special initiator. However, a search of the model did not locate the dam failure. Further, the probability of a loss of the SW pond dam is estimated to be 1.9e-7 failures per year based on the FNP River Water Study (dated 1982). This analysis is based on a generic estimate of 1.9e-5 failures per year for earthen filled dams that in the opinion of Alabama Power Company should be reduced to 1.9e-7 per year due to design, monitoring, maintenance, and responsiveness of the owner to problems. Loss of the dam would result in a dual unit loss of service water. For an event of the magnitude of a dual unit loss of service water, the supporting evidence for reduction of the generic value by a factor of 100 is treated very lightly. An initiating event that would result in a dual event initiator should be included in the initiating event portion of the model. Evidence for reducing the generic dam failure probability is qualitative in nature, and the extension of this information to justify a factor of 100 reduction in the generic probability is not clear and poorly supported. Further, dam failure analysis technology has improved since 1982, and use of the newer approaches to analysis should be considered.

Consider revisiting the estimation of the probability of dam failure using newer technology and better supported calculation. Add the loss of the SW pond dam to the model, if appropriate."

F&O Summary

This F&O questioned the results of the Farley River Water Study performed in 1982. It also challenged the validity of the analysis methods used to determine the frequency of the SW Pond dam failure and the rationale for screening loss of SW due to a random failure of the dam as an initiating event in the internal events PRA, stating that evidence for reducing the generic dam failure frequency is not clear and poorly supported.

Response

The information presented below supports SNC's conclusions that loss of SW due to a random failure of the dam as an initiating event does not need to be modeled in the internal events PRA based on the screening criteria in IE-C6 (b) of the ASME PRA Standard.

<u>SW Pond Dam Licensing Basis:</u>

Section 9.2.1.2.3.2 of the FSAR states: "The storage pond dam was designed and constructed and is maintained in compliance with applicable industry standards. Additionally, a detailed analysis was performed to demonstrate the reliability of the pond dam and the results of this analysis indicated the possibility of a dam failure is approximately 1.9 x 10^{-7} failures per year. Therefore, the loss of the storage pond dam is not considered to be a credible event and such

an event is not postulated as part of the design basis of the station cooling water system." Also, in the Safety Evaluation in FSAR section 9.2.5.3 performed to show that the SW storage pond has the capability to provide sufficient cooling for at least 30 days it is noted that "a loss of the pond dam is not considered to be a credible failure."

The failure frequency of 1.9E-07/year is in the Farley FSAR as referenced above and as a result has been previously reviewed and accepted as part of the design basis for which the plant is licensed. As such, this frequency is used in the analysis for screening loss of SW due to a random failure of the dam as an initiating event in the internal events PRA.

Re-calculation of Failure Frequency of a Random Dam Failure using Generic Information:

A re-analysis which calculates the failure frequency of a random failure, using current Industry methods and generic information in references provided by an Industry expert in external flooding issues, is provided below.

According to experts, dam failure may result from one of three reasons: (1) hydrologic failures (2) seismic failures and (3) sunny day (or internal dam failures). Hydrologic failures are associated with dam overtopping. This occurs when precipitation events drive the water level of the dam to exceed its capacity. Unique features of the Farley SW pond dam which prevent overtopping and the potential for erosion from overfill are provided in a section below which discusses specifics which distinguish the Farley SW pond dam from generic dams. The potential for Seismic failures is negligible due to the fact that the SW pond dam is a Seismic Category I Safety-related structure and Farley is in a low seismic zone. Therefore, seismic failures and hydrologic failures resulting from precipitation above the maximum design storm are not considered as random failures in the re-calculation of the random dam failure frequency using generic information.

The third failure mode, associated with dam failure due to random/unknown causes, is considered in more detail below.

Dam Internal Failure Frequency Assessment
In Technical Report TR-70, "Preliminary Safety Evaluation of Existing Dams," John A. Blume Earthquake Engineering Center, Stanford University, M. McCann provides guidance for ranking dams due to risk of dam failure. This report is intended to help prioritize resources for dam improvements/upgrades. Included in this report is a means to evaluate the dam failure frequency for earthen dams based on dam condition. Four condition evaluations are performed for the dam. These are:

1. Evaluation of Piping and Outlet Works
2. Evaluation of Foundation
3. Evaluation of Slope Stability
4. Evaluation of Dam Piping

A ten-point rating scale is provided, with 1 being the highest rating, 10 the lowest. The rating scales are presented below for the top categories.

Earth, Rockfill Dams

Evaluation Scale for Piping Associated with Outlet Works

Evaluation* Scale	Description
	Good Condition
1 2 3	Outlet works are in good condition. No indication of corrosion. No clogging or obstructions.
	Neutral Condition
4 5 6 7	Outlet works are in fair condition. Evidence of corrosion and seepage around the pipe (i.e. vegative cover, pipe always has water in it), possible minor erosion.
	Poor Condition
8 9 10	Outlet works in poor condition. Evidence of extensive corrosion or structural damage (i.e. cracking, deformation, offsets, etc.) Excessive seepage into and/or around the outlet works.

The evaluation scale is grouped into 3 general categories; good, neutral, and poor. In this case the engineer identifies what the condition of the dam is, and adjusts the evaluation within the category.

Earth, Rockfill Dams

Evaluation Scale for Foundation and Miscellaneous Failures

Evaluation* Scale	Description
1	Good condition. No signs of cracking, heaving, unusual movements, or pressures. No signs of erosion. Good factor of safety (if known).
2	Dam in good condition. Minor erosion of abutment and foundation soils.
3	Good condition of the dam and foundation. No evidence of differential settlements. Possible indication of fractured rock or potentially weak foundation material.

TABLE 4.4

Earth, Rockfill Dams

Evaluation Scale for Slope Stability

Evaluation Scale	Description
1	Good condition of dam. No unanticipated movements or performance. Mild operating conditions and climate.
2	Good condition of dam, but dam subjected to extreme weathering or drawdown cycles are likely.
3	Minor erosion of slope face.
4	Evidence of weak material or organic matter.
5	Evidence of minor erosion and weak material.

TABLE 4.3

Earth, Rockfill Dams

Evaluation Scale for Piping

Evaluation Scale	Description
1	Good embankment, with no signs of defects or distress. Any seepage which exists is in agreement with design expectations (if available).
2	Generally good embankment, but evidence of minor defects, such as a few burrow holes, organic matter in embankment, etc.
3	Embankment in good condition, minor seepage observed.
4	Localized vegetative growth which may indicate slight seepage free of soluble contents or suspended solids. Inadequate filter system.
5	Numerous burrow holes or tree growth along downstream slope only. Evidence of rotting debris in embankment, foundation, or abutments.

Once the evaluation is complete the dam internal failure frequency can be estimated using the table below (Table 4.10 of the Reference).

TABLE 4.10

Earth, Rockfill Dams

Frequency of Dam Failure

Evaluation Scale	Piping	Slope Stability	Piping/Outlet Works	Foundation and Misc.
1	7.9×10^{-6}	1.1×10^{-6}	5.9×10^{-7}	3.9×10^{-6}
2	1.1×10^{-5}	2.8×10^{-6}	7.9×10^{-7}	1.0×10^{-5}
3	1.2×10^{-5}	3.3×10^{-6}	8.0×10^{-7}	1.1×10^{-5}
4	2.5×10^{-5}	5.9×10^{-6}	1.6×10^{-6}	1.9×10^{-5}
5	3.8×10^{-5}	9.2×10^{-6}	3.4×10^{-6}	3.6×10^{-5}
6	6.3×10^{-5}	1.3×10^{-5}	6.6×10^{-6}	5.1×10^{-5}
7	9.5×10^{-5}	2.1×10^{-5}	1.1×10^{-5}	8.2×10^{-5}
8	1.9×10^{-4}	3.7×10^{-5}	2.7×10^{-5}	1.3×10^{-4}
9	6.9×10^{-4}	4.8×10^{-5}	6.8×10^{-4}	2.8×10^{-4}
10	1.1×10^{-3}	1.5×10^{-4}	1.2×10^{-4}	7.5×10^{-4}

Due to the Technical Specification requirements for maintenance and monitoring of the SW pond dam, which is a Seismic Category I Safety-related structure, it is assumed that the dam would rate 1 for all categories. Using the 1 rating a total internal dam generic failure frequency is estimated:
Total Internal Failure Dam Frequency = (7.9E-06 + 1.1E-06 + 0.59E-06 + 3.9E-06) per year
=**1.2E-05/year**
Note that this generic frequency of 1.2E-05/year is on the same order as the 1.9E-05/year frequency from the previous Farley study.
More recently, the NRC has investigated historical dam failures due to all causes and found the generic values to be on the order of 2.0E-04/year for earth filled dams (Ferrente, et al). The table summary is provided below.

Table II. Failure rates for grouped category dam types

	Grouped Category	Number of Failures	Dam-years	Failure Rate (/dam-year)
I	Earth	86	396033	2.2 x 10-4
II	Gravity	7	17277	4.1 x 10-4
III	Rockfill	7	6152	1.1 x 10-3
IV	Concrete	6	5262	1.1 x 10-3
V	Other/Unknown	42	93634	4.5 x 10-4

Note that if an evaluation scale of 5 is used, the Stanford estimated failure rate due to all causes (using Table 4.10) for an average dam would be 9E-05/year and for a poorly maintained dam (7 out of 10) would be about 2.2E-04/year. These values are in line with the NRC observations therefore a value of 1.2E-05/year calculated using the Stanford generic information is reasonable for a well maintained generic dam.

Based on the results of this re-analysis the frequency calculated is similar to the frequency estimated previously for the 1982 study. So, it is not clear that the methods of analysis for random failure of dams not caused by seismic or other external events have significantly changed or improved since 1982 to the point that the previous analysis is now invalid.

Information Supporting Reduction in Failure Frequency:

Specific aspects which distinguish the seismically designed Safety-related SW pond dam from generic dams are described below. These differences support/justify further reduction of the frequency of failure from **generic estimates** based on SW pond dam design, construction, operation, and Technical Specification requirements for monitoring. A summary of this information in the Farley FSAR and Technical Specifications is provided below.

Design and Construction:
The SW pond dam is a **Seismic Category I Safety-Related structure**. As such the dam and dikes were designed with factors of safety adequate to resist all static and earthquake dynamic loads. Rigorous analyses were performed which considered all credible failure modes, including soil liquefaction. The extensive dam design and stability analysis along with the soil material design parameters and soil testing are described in FSAR Sections 2B.7.6.3 and 2B.7.6.4.

FSAR section 2.B.7.6.12 notes the dam foundation and the embankments for the dam were constructed as quality controlled compacted fills. Throughout the embankment construction a record test program was conducted where a total of 26 one cubic foot block samples of the compacted fill were obtained and laboratory tested to ensure that the engineering properties of the fill were equal to or greater than those used for design. In addition, soil compaction tests were performed in multiple locations of the compacted layers during construction to confirm that the soil compaction of the layers met the design specifications.

Overfill Protection and Erosion Prevention:
As discussed in FSAR sections 2B.7.6.1 and 2.4.14.2, the SW pond dam is thirty feet wide at the crest, at elevation 195 feet, with a **9-ft freeboard above the normal maximum pond level of 186 feet** and includes an **uncontrolled concrete emergency spillway the top of which is also at elevation 186 feet**, designed to pass the runoff from the maximum design storm. Since the top of the spillway is well below the crest elevation of the dam, **over-topping of the dam is prevented and the potential for erosion resulting from over-filling of the pond is eliminated**. In addition, the downstream earth slopes are seeded with grass and the upstream slopes have dumped riprap erosion protection.

Impacts from River Flooding:
Although river flooding is not considered as a cause of random failure, as described in FSAR section 2B.7.6.6, during the dam's construction the excavated foundation materials were placed at the downstream toe of the dam as a fill to add to the stability of the dam. (The plan of the downstream fill is shown on drawing D-176980.) This fill was graded and seeded for erosion protection. Water from the postulated high river flood level (El 149 feet) is prevented from reaching the downstream slope of the dam by the downstream fill.

Seepage prevention:
To control seepage, prevent a buildup of ground water pressure within the lower sand stratum downstream of the dam, and to increase stability against liquefaction, gravity relief wells were provided in the dam. Gravity relief wells were also placed along the dikes. (The plan location, cross-section, and details of these gravity relief wells are shown on drawings D-176980, D-176981, D-176982, and D-176939.) In addition, piezometers and observation wells are installed in the main dam and dike sections.

Monitoring:
Periodic monitoring of the SW Pond Dam is required by the Technical Requirements Manual as follows:

FNP-0-STP-611.0 (TRS 13.7.4.1) UHS Support Structures (Spillway Channel Erosion)
FNP-0-STP-611.1 (TRS 13.7.4.2) UHS Support Structures (Spillway Structure and Channel)
FNP-0-STP-125 (TRS 13.7.4.3) UHS (Pond Ground Water Seepage)

Other required routine SW dam Inspections which include the earthen dam, the service water pond embankments, and the spillway slopes are outlined in the following:

FNP-0-ETP-1035.0 Monthly inspection by System Engineer
FNP-0-ETP-4389 SW Pond Dam Biennial Inspection
FNP-0-ETP-4384 SW Pond Deformation Monument Readings
FNP-0-ETP-4338 SW Pond Sounding Survey
FNP-0-STP-125.0 SW Pond Seepage Test
FNP-0-ETP-4381 Storage Pond Piezometer and Observation well Readings

Following excessive rainfall events if the SW pond level increases to greater than 187 feet FNP-0-AOP-21.0, Appendix IV, Excessive Rainfall Contingencies directs operations to visually inspect the spillway channel for erosion damage after the severe weather condition has passed and notify maintenance personnel to inspect the SW dam for erosion damage.

Review of some of the completed test reports confirm that the inspections have been performed with recommendations for corrections noted, as required and therefore it is concluded that the SW pond dam continues to meet the design specifications.

ASME PRA Standard Screening Evaluation:

The screening criteria in Supporting Requirement IE-C6 (b) of the ASME PRA Standard states that events can be screened if "The frequency of the event is less that 1E-06/year, and core damage could not occur unless at least two trains of mitigating systems are failed independent of the initiator.

As discussed above the frequency shown in the FSAR for loss of SW due to random failure of the SW pond dam, which has been previously reviewed and accepted as part of the design basis for which the plant is licensed, is 1.9E-07/year which is less than 1E-06/yr.

In the event of loss of the SW pond, FNP-0-AOP-31.0, "Loss of Service Water Pond," directs operators to align all the River Water System (RWS) pumps directly to the SW pump wet pit to provide flow to the suction of the SW pumps. If level in the wet pit continues to decrease due to insufficient flow from RWS, operators are then directed to also re-align both trains of SW discharge flow on both units to recirculate back to the SW pump wet pit instead of discharging to the river.

Based on the information above the screening criteria of IE-C6 (b) are met since:

- the frequency of loss of SW due to random failure is less than 1E-06/yr, and
- there are at least two independent trains of mitigating systems to provide flow to the suction of the SW pumps for mitigation of core damage using:
  1) The RWS pumps which can be aligned directly to the SW pump wet pit, and
  2) both trains of SW discharge flow, on both units, which can be re-aligned to recirculate back to the SW pump wet pit, instead of discharging to the river.

Therefore, a loss of SW due to a random failure of the SW pond dam as an initiating event does not need to be modeled in the internal events PRA based on the screening criteria in IE-C6 (b) of the ASME PRA Standard.

Conclusions:

Based on the information presented above which calculated a failure probability similar to the previous value combined with the fact that the SW pond dam was designed and is maintained as a Seismic Category 1 Safety-related structure, extensively tested during construction to ensure design requirements and material specifications were met, and is routinely inspected per the Technical Requirements Manual to detect any signs of degradation it is concluded that the results of the previous analyses remain valid. As such, there is not sufficient justification to perform a new analysis to address this F&O. Applying screening criteria of the ASME PRA Standard it is further concluded that modeling random loss of the SW pond dam, not due to an external cause such as a seismic event, as a loss of SW initiating event in the Farley Internal Events PRA is not required.

RAI 5

In Section 6.3 "Potential Impact from External Events Contribution", Table 6-2 provides core damage frequency (CDF) and large early release frequency (LERF) values for Fire Events from the FNP Fire PRA (FPRA) that credits pending modifications for NFPA 805 that will be implemented by the end of 2017. State if the fire CDFs (and LERFs) reflect changes to FPRA methods made since the safety evaluation (SE) was issued for NFPA 805 including the following:

a. New guidance on the credit taken for very early warning fire detection system (VEWFDS) is available in NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities, (DeloresVEWFIRE)" which is now available as a final report at Agencywide Documents Access and Management System (ADAMS) Accession No. ML 16343A058. The methodology in NUREG-2180 is acceptable to the NRC because it is currently the best available guidance. The guidance provided in frequently asked question (FAQ) 08-0046, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems" (ADAMS Accession No. ML 16253A111), has been retired and alternative approaches for staff evaluation are necessary to complete the safety evaluation.

   If the Fire PRA credits the future (or current) installation of VEWFDS, explain how credit (e.g., approach, methods, data, and assumptions) taken for the proposed VEWFDS is consistent with the guidance in NUREG-2180 or bounds the metrics in this that would be obtained had the guidance in NUREG-2180 been applied. If credit taken for VEWFDS in the FPRA is not consistent with or bounded by NUREG-2180 provide:

   1) The risk metrics that would be obtained had the guidance of NUREG-2180 been applied, or that would be obtained had an alternative method been used, along with a description and justification for the alternative method. Development and use of an alternative proposal may extend the time required to complete the review. The new risk results can be generated from a sensitivity study type evaluation insofar as formal incorporation of the new method into the PRA model of record is not required.

   2) Explain how any increases in the risk metrics are consistent with the acceptance criteria for this application.

b. Changes (generally increases) in fire ignition frequencies and non-suppression probabilities from NUREG-2169. For example, the frequency of fires in the Main Control Board is now twice as high as in the original NUREG/CR-6850 and six times higher than in its Supplement 1. The mean time to suppress a welding fire is nearly twice as long as in both the original NUREG/CR-6850 and its Supplement 1.

c. Possible increases in spurious operation probabilities from NUREG/CR-7150, Vol. 2. For example, intra-cable spurious operation for an ungrounded AC, with individual control

power transformers, single-break control circuit for a solenoid-operated valve has a probability of 0.64, slightly higher than the value of 0.6 from NUREG/CR-6850.

SNC Response

SNC provides the following response to RAI 5(a):

SNC is in the process of updating the Farley FPRA for its NFPA-805 "true up" License Condition. This effort is ongoing and has not yet been completed. SNC will provide a response to this RAI by August 31, 2017.

SNC provides the following response to RAI 5(b):

The Fire Human Reliability Analysis (HRA) has not been updated since the LAR submittal. The floor HEP value of 1.0E-5 remains valid and applicable for this ILRT application.

SNC provides the following response to RAI 5(c):

NUREG/CR-7150, Vol. 2 has not been applied to the Farley Fire PRA. Consequently, the CDFs and LERFs presented in this LAR do not reflect NUREG/CR-7150, Vol. 2.

RAI 6

In the application for NFPA 805, an F&O related SR IE-A 10 cited the need to address the probability of dam failure using newer techniques, possibly with inclusion of loss of the SW pond dam to the model. The resolution cited a sensitivity evaluation for which the estimated CDF and LERF were approximately 1 E-5/yr and 1 E-7/yr, respectively, for each unit. Table 6-2 of Attachment 1 for the Permanent ILRT Interval Extension Risk Impact Assessment cites "screened out" for "Other External Risk," which presumably would include this dam failure. Address whether this dam failure analysis has been updated or not. If updated, discuss whether the "screened out" citation still applies. If not, revise Table 6-2 to include the results of the cited sensitivity evaluation and address the effect on risk and delta-risk metrics.

SNC Response

SNC provides the following response to RAI 6:

The SW pond dam is a Seismic Category 1 safety-related structure and goes through routine inspections to ensure integrity is maintained. The SW pond failure analysis has not been updated using any newer approaches. The previous detailed evaluation referred to in the F&O related to SR IE-A10-02 determined a failure frequency for loss of service water, due to random failure of the dam, that is below the 1E-06/yr screening criteria of the ASME PRA standard. This evaluation is discussed in sections 9.2.1.2.3.2 and 9.2.5.3 of the Farley FSAR and as such has been previously reviewed and accepted as part of the design basis for which the plant is licensed.

Further review of the criteria and information used in this previous evaluation has been done since the NFPA 805 submittal dated 09/25/2012 to ensure the issues expressed in the F&O as written have been adequately addressed. The results of the review are summarized in the Farley Internal Events PRA Peer Review F&O resolution document, F-RIE-IEIF-U00-012, dated March 2016. Excerpts from the F&O resolution document are included in the response to RAI 4. This review concluded that the results of the previous analyses remained valid and there is no sufficient justification to perform a new analysis to address this F&O. Therefore, the basis for screening loss of SW due to a random failure of the dam as an initiating event in the Farley Internal Events PRA remains consistent with this previous analysis, and modeling random loss of the SW pond dam as a loss of SW initiating event in the Farley Internal Events PRA is not required.

In addition, overtopping is not possible during intense precipitation. As discussed in FSAR sections 2B.7.6.1 and 2.4.14.2, the SW pond dam is thirty feet wide at the crest, at elevation 195 feet, with a 9-ft freeboard above the normal maximum pond level of 186 feet and includes an uncontrolled concrete emergency spillway the top of which is also at elevation 186 feet, designed to pass the runoff from the maximum design storm. Because the top of the spillway is well below the crest elevation of the dam, over-topping of the dam is prevented and the potential for erosion resulting from over-filling of the pond is eliminated. In addition, the downstream earth slopes are seeded with grass and the upstream slopes have dumped riprap erosion protection.

Impacts from river flooding were considered in the design and construction of the SW pond dam. The SW pond was constructed to serve the plant's cooling needs during record external floods, at which time flow from the river water intake structure would not be available. The design bases concerned with the external hazard of flooding can be found in the FSAR. In order to estimate the Probable Maximum Flood (PMF) conditions which the plant should be designed to withstand, the Elba, Alabama, storm was transposed and maximized over the Chattahoochee River drainage area. The peak level that such a storm would produce is estimated at 144 ft. If high winds occurred simultaneously, wave run up would contribute up to approximately 9 ft to the peak flood level. Flood levels could then potentially reach 153 ft. The floor of the river water intake structure is flood protected to 127 ft, which exceeds the largest storm of record, but falls short of PMF. The SW pond dam begins at elevation of 158 ft and is therefore protected from projected flood effects.

In conclusion, based on this information and design of the spillway which would prevent overtopping, neither External Flooding from the river nor local intense precipitation will lead to failure of the SW pond dam.

RAI 7

In Section 6.3 "Potential Impact from External Events Contribution", Table 6-2 provides CDF and LERF values for Seismic Events from the FNP FPRA. As cited in the SE for NFPA 805 transition (ADAMS Accession No. ML 14308A048), Farley used an average of the CDF values (1.73E-5/yr per unit) from the "Safety/Risk Assessment Results for GI-199" (ADAMS Accession No. ML 100270582).

NRC Staff results show that if using 1.73E-5/yr per unit, the CDF totals on Table 6-2 would calculate to 1.03E-4/yr and 1.08E-4/yr, respectively. Both would minimally exceed the RG 1.17 4 Region II threshold of 1.00E-4/yr. NRC Staff notes that an increase in each LERF would also occur based on the seismic LERFs of 2.02E-7/yr and 2.60E-7/yr for Units 1 and 2, respectively, as cited in the NFPA-805 SE.

1) Perform a complete recalculation of Table 6-2 and subsequent calculations in Sections 6 and 7 using the values cited in the NFPA-805 SE and address all the issues identified in the preceding RAIs (RAI No. 1, 3, 5, and 6).

2) Confirm that any increases in the risk metrics as a result of the recalculation in part 1) above does not change the justification for exceeding the acceptance criteria for this application.

With respect to Table 6-2, the application stated, "the value for Total Internal and External events CDF slightly exceeds a value of 1.0E-04[/yr]. This value is expected to fall below 1.0E-04[/yr] when the Farley Internal Events PRAs for Unit 1 and Unit 2 credit the Generation III RCP shutdown seals which are already installed. Crediting the Generation III RCP seals is expected to reduce Internal Events CDF to the mid 1.0E-06/yr range on both units."

Justify the expectation of a reduction in internal events CDF after crediting the Generation III RCP seals to the mid 1.0E-6/yr range on both units by using a bounding quantification.

SNC Response

SNC provides the following response to RAI 7:

SNC is in the process of updating the Farley FPRA for its NFPA-805 "true up" License Condition. This effort is ongoing and has not yet been completed. SNC will provide a response to RAI 7 by August 31, 2017.

SNC has reviewed this RAI for its potential impact on the RAIs addressed in this letter and has determined that there will be no impact to preceding RAIs 1,3, and 6.

<u>Addressing the Issues Identified in preceding RAIs 1, 3, 5, and 6</u>

RAI 1 discusses the validity of the calculated values of Class 3a and 3b in Table 5-7 of the LAR. These values are calculated based on the internal events PRA, and are not impacted by the magnitude of the Fire and Seismic CDF or LERF.

RAI 3 involves an issue of PRA technical adequacy concerning the licensee's treatment of the dependency between multiple human actions. The response to RAI 3 is not impacted by the impacted by the magnitude of the Fire and Seismic CDF or LERF.

Any related issues in RAI 5 will be addressed in the response to RAI 7 when these two RAIs are answered by August 31, 2017.

RAI 6 involves an issue of PRA technical adequacy concerning the licensee's treatment of the probability of dam failure. The response to RAI 6 is not impacted by the impacted by the magnitude of the Fire and Seismic CDF or LERF.

RAI 8

The following items need additional clarification:

a. Section 4.2.1 of Attachment 1 lists CDF values of 1.91 E-05/yr for Unit 1 and 1.75E-05/yr for Unit 2. State if these values are for Internal Events only.

b. In Section 5.1 of Attachment 1, there appears to be an error in Class 3b Frequency for Unit 1. The calculation shown on page 29 has a value of 3.52E-07/yr for Class 2; however, Table 5-2 provides a value of 3.52E-08 for EPRI Class 2. Confirm the correct value. If not correct, state if any results change.

c. In Appendix A, Section A.2.2 "Parts of the PRA" refers to reference 34 and reference 37. However, Section 3 "Parts of the PRA" contains the same statement with different references (reference 17 and reference 10). Please clarify which references are correct.

d. In Section A.2.5 "Assessment of PRA model Technical Adequacy," it is stated that an independent peer review was conducted in 2001 and refers to reference number 31. However, in your references section, reference 31 is NEI 05-04, Rev 2, dated November 2008. Please clarify.

SNC Response

SNC provides the following response to RAI 8:

a. Yes. Consistent with the EPRI Risk Impact Assessment Template, CDF values of 1.91 E-05/yr for Unit 1 and 1.75E-05/yr for Unit 2 in Section 4.2.1 are for Internal Events only.

b. Section 5.1 of Attachment 1, contains a typographical error in the Class 3b Frequency for Unit 1. The Table 5-2 value of 3.52E-08 is correct, and this correct value was used in subsequent calculations. Therefore, the error had no downstream impact on the results.

c. Both are correct. Reference 17 and 10 refer to two references in a list of references in Section 6 of the main body of the LAR. Reference 34 and 37 refer to the same two references in a list of references at the end of the ILRT Risk Assessment in Attachment 1.

d. In the first bullet item of Section A.2.5 "Assessment of PRA model Technical Adequacy," SNC states that the independent PRA peer review in 2001 followed an Industry PRA peer review process and then refers to Reference 31. While the Industry PRA peer review process is mentioned in the first paragraph of the Introduction to NEI 05-04, Rev 2, there does not appear to be any significant value in referencing NEI 05-04 at the end

of the first bullet item in Section A.2.5. The correct reference lacks specificity, however it may simply be listed as:

Farley PRA Peer Review Report, Westinghouse Owners Group (WOG), 2001.

RAI 9 (restatement of informal request for clarification by NRC in May 24, 2017 email)

In section 3.1.3 "Containment Overpressure on ECCS Performance," SNC does not explicitly state that SNC does not rely on containment overpressure for ECCS NPSH. In addition, in Table 3.3.1-1, in answer to whether containment over pressure is relied upon for ECCS performance, SNC responds with a statement describing the "maximum long-term suppression pool temperature." Because the suppression pool is a BWR system, and Farley is a PWR, there appears to be a copy/paste or typographical error.

Please clarify item 4 of Table 3.3.1-1 and provide a succinct statement or explanation stating whether containment over pressure is relied upon for ECCS performance.

SNC Response

SNC provides the following response to RAI 9:

SNC regrets the copy/paste error in item 4 of Table 3.3-1 and submits the following statement as the replacement language for the FNP response:

Plant Farley Units 1 and 2 do not rely on containment overpressure for ECCS NPSH.

Joseph M. Farley Nuclear Plant
License Amendment Request to Revise Technical Specification
Section 5.5.17 "Containment Leakage Rate Testing Program"
Proposed TS 5.5.17 - Marked-Up TS Page

Enclosure 2

NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008 as modified by the following exceptions

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":~~

1. **The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.**

2. **The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.**

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.8 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

Leakage rate acceptance criteria are:

a. **Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and C tests, and $\leq 0.75\ L_a$ for Type A tests;**

b. **Air lock testing acceptance criteria are:**

   1. **Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.**

   2. **For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.**

c. **During plant startup following testing in accordance with this program, the leakage rate acceptance criterion for each containment purge penetration flowpath is $\leq 0.05\ L_a$.**

(continued)

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Section 5.5.17 "Containment Leakage Rate Testing Program"
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Enclosure 3

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008 as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.8 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

c. During plant startup following testing in accordance with this program, the leakage rate acceptance criterion for each containment purge penetration flowpath is $\leq 0.05\ L_a$.

(continued)

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":~~

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.8 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

c. During plant startup following testing in accordance with this program, the leakage rate acceptance criterion for each containment purge penetration flowpath is $\leq 0.05\ L_a$.

(continued)

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008 as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.8 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

c. During plant startup following testing in accordance with this program, the leakage rate acceptance criterion for each containment purge penetration flowpath is $\leq 0.05\ L_a$.

(continued)