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

July 8, 2013

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
40 Inverness Center Parkway, Bin 038
Birmingham, AL 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION CONCERNING VOLUNTARY FIRE PROTECTION RISK INITIATIVE REQUEST (TAC NOS. ME9741 AND ME9742) (NL-12-1893)

Dear Mr. Pierce:

By letter dated September 25, 2012, the Southern Nuclear Operating Company (SNC) submitted a license amendment request for Joseph M. Farley, Units 1 and 2. The proposed amendment requests the review and approval for adoption of a new fire protection licensing basis which complies with the requirements in Sections 50.48(a) and 50.48(c) to Title 10 to the *Code of Federal Regulations* CFR (10 CFR), and the guidance in Regulatory Guide (RG) 1.205, Revision 1, *Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants*.

A response to the enclosed Request for Additional Information (RAI) is needed before the Nuclear Regulatory Commission (NRC) staff can complete the review. This request was discussed with Mr. Jeff Branum of your staff on June 12, 2013, and it was agreed that SNC would respond within 120 days of the issuance of this letter. In addition, on April 17, 2012 (ADAMS Accession No. ML12103A028) the NRC staff indicated that documents identified as necessary for analysis of the application will be identified by the NRC staff and the licensee will be formally requested to submit those documents on the docket. Enclosure 1 contains a list of those documents.

Should you have any questions, please feel free to contact me at (301) 415-2315.

Sincerely,

/RA/

Eva A. Brown, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Portal Document Request
2. RAI

cc w/enclosures: Listserv

Portal Documents for Submission

Calculation SE-C051326701-008, Farley Nuclear Plant, Units 1 and 2, NFPA 805 Fire Risk Evaluations, Version Number 1, dated September 25, 2012

Calculation PRA-BC-F-11-006, Joseph M. Farley Nuclear Plant, Units 1 and 2, Farley Fire PRA Task 13, Seismic-Fire Interactions Assessment, Version Number 1, dated September 20, 2011

Calculation PRA-BC-F-11-014, Joseph M. Farley Nuclear Plant, Units 1 & 2, Fire Scenario Development, Version Number 2, dated September 14, 2012

Calculation PRA-BC-F-11-016, Joseph M. Farley Nuclear Plant, Units 1 and 2, Human Reliability Analysis for Fire Events, Version Number 1, dated September 13, 2012

Calculation PRA-BC-F-12-004, Joseph M. Farley Nuclear Plant, Units 1 and 2, Fire PRA Sensitivity Analysis, Version Number 0, dated September 21, 2012

Calculation PRA-BC-F-11-002, Joseph M. Farley Nuclear Plant, Units 1 and 2, Component Selection for Farley Fire PRA, Version Number 3, dated June 29, 2012

Calculation PRA-BC-F-11-004, Joseph M. Farley Nuclear Plant, Units 1 and 2, Fire PRA Logic Model for Farley Nuclear Plants, Version Number 2, dated June 29, 2012

Calculation PRA-BC-F-11-014, Attachment 1, "Evaluation of Units 1 and 2 Control Room Abandonment Times at the Joseph M. Farley Nuclear Plant, September 18, 2012

REQUEST FOR ADDITIONAL INFORMATION

VOLUNTARY FIRE PROTECTION RISK INITIATIVE

SOUTHERN NUCLEAR OPERATING COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

Fire Protection Engineering (FPE) Request for Additional Information (RAI) 01

The compliance strategy for Section 3.3.8, "Bulk Storage of Flammable and Combustible Liquids" in Table B-1, of Attachment A (Table B-1) to the submittal dated September 25, 2012 (Agencywide Document Access and Management System (ADAMS) Accession Nos. ML122790258) is identified as "N/A" or not applicable. However, a compliance basis is provided. Discuss the compliance strategy.

FPE RAI 02

Section 50.48(c)(2)(vii) of Title 10 of the *Code of Federal Regulations* (10 CFR) allows the use of performance based-methods for fire protection program elements and minimum design requirements of Chapter 3 of National Fire Protection Association Standard (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", 2001 Edition (NFPA 805). Section 3.2.3(1), "Inspection, Testing, and Maintenance," of Table B-1, indicates "Complies" with a note that the current program may be modified by using the performance-based program of "Fire Protection Equipment Surveillance Optimization and Maintenance Guide, Electric Power Research Institute (EPRI), Technical Report (TR) 1006756, July 2003, (EPRI TR 1006756) for fire protection equipment surveillances in the future. Address whether EPRI TR 1006756 is intended as an alternative, and if so provide the appropriate supporting information consistent with 10 CFR 50.48(c)(2)(vii).

FPE RAI 03

The compliance basis for Section 3.3.5.2, "Electrical Raceway Construction Limits" of Table B-1, indicates formal approval is being requested in Attachment L to the submittal, Approval Request 2, "NFPA 805 Section 3.3.5.2," for existing embedded conduit configurations. However, no discussion is provided for non-embedded configurations. Describe how the requirements of Section 3.3.5.2 of NFPA 805, for non-embedded configurations are met. Also, provide the compliance basis for future installations of conduit embedded in concrete.

FPE RAI 04

Section 3.5.15, "Water Supply Hydrant Code Requirements," of Table B-1, and Attachment L, Approval Request 4, "NFPA 805 Section 3.5.15," requests the continued use of hydrants located more than 250 feet apart on the yard main system (up to 300 feet). Demonstrate that

manual firefighting capability is adequate where hydrants are spaced more than 250 feet apart. The discussion should include the amount of hose provided, distance of effective hose streams and confirmation that adequate water pressures and flows are provided at the nozzle given additional friction losses.

FPE RAI 05

Approval Request 5, "NFPA 805, Section 3.5.16," of Attachment L, requests approval for the use of the fire protection system to supply water for the manual wash down and flushing of the Circulating Water System (CWS) components. Describe the hydraulic demand required for this non-fire suppression related activity and discuss administrative controls to be used during this scenario to ensure the fire water system is available to perform its design function when needed.

FPE RAI 06

Fire protection systems and features that require NFPA code compliance are reflected in Chapter 3, "Fundamental Fire Protection Program and Design Elements of NFPA 805." Provide a complete list of committed NFPA codes and standards including identification of the edition (years) that will be in place post transition. For those codes and standards with numerous editions, identify which plant areas and systems apply to which editions.

FPE RAI 07

The compliance basis for Section 3.3.5.3, "Electrical Cable Flame Propagation Limits," of Table B-1, states that "cables purchased prior to the issue of this standard (IEEE-383) were purchased under the requirements of the applicable Insulated Cable Engineers Association (ICEA) standard and an additional prototype flame test, which met the later requirements of Institute of Electrical and Electronic Engineers (IEEE) Standard 383." Provide a technical basis for the acceptability of the ICEA cables.

FPE RAI 08

The compliance basis for Section 3.11.2, "Fire Barriers," of Table B-1, states "calculation SM-C051326701-006 defines physical boundaries to be used in the NFPA 805 project." This statement does not address how physical boundaries meet the requirements for fire barriers. Describe how the requirements for fire barriers as described in NFPA 805, Section 3.11.2, "Fire Barriers" are met.

FPE RAI 09

The compliance basis for Section 3.3.1.3.4, "Control of Ignition Sources on Portable Heaters," of Table B-1, states that procedures will be revised to "restrict portable fuel fired heaters from plant areas containing equipment important to nuclear safety". Section 3.3.1.3.4, to NFPA 805, states "portable fuel-fired heaters shall not be permitted in plant areas containing equipment important to nuclear safety". Describe how the procedural restrictions will meet the prohibition of fuel-fired equipment in plant areas containing equipment important to nuclear safety.

FPE RAI 10

Sections 3.3.1.1, 3.3.7.1, 3.3.8, and 3.4.1 of Table B-1 indicate that code compliance reviews have been performed to ensure the requirements of appropriate NFPA codes are met. Describe how any non-compliances identified during these reviews were addressed. If any non-compliances are still outstanding, describe how these will be addressed prior the completion of NFPA 805 implementation.

Safe Shutdown Analysis (SSA) RAI 01

Databases and software that integrate fire protection program structure, system, and component data; fire modeling results, and PRA analyses (e.g. ARCPlus) have a range of uses applicable to NFPA 805 implementation. These uses are subject to several NFPA 805 requirements including those that address determination of success paths; completion of the Nuclear Safety Capability Assessment (NSCA); the quality, configuration control, documentation, and verification and validation of analyses; and limitations of use. In addition, these databases and software can be used to facilitate integration of several aspects of NFPA 805 compliance. Specific applicable NFPA 805 requirements include:

NFPA 805 Section 2.2.9 "Plant Change Evaluation" states that: "In the event of a change to a previously approved fire protection program element, a risk informed plant change evaluation shall be performed and the results used as described in 2.4.4 to ensure that the public risk associated with fire-induced nuclear fuel damage accidents is low and that adequate defense-in-depth and safety margins are maintained."

NFPA 805, Section 2.2.11 "Documentation and Design Configuration Control" requires that: "The fire protection program documentation shall be developed and maintained in such a manner that facility design and procedural changes that could affect the fire protection engineering analysis assumptions can be identified and analyzed."

NFPA 805 Section 2.4.1 "Fire Modeling Calculations" requires: (2.4.1.1) "The fire modeling process shall be permitted to be used to examine the impact of the different fire scenarios against the performance criteria under consideration." (2.4.1.2.1) "Only fire models that are acceptable to the authority having jurisdiction shall be used in fire modeling calculations." (2.4.1.2.2) "Fire models shall only be applied within the limitations of that fire model." (2.4.1.2.3) "The fire models shall be verified and validated."

NFPA 805 Section 2.4.3.3 regarding fire risk evaluations states: "The PSA approach, methods, and data shall be acceptable to the AHJ. They shall be appropriate for the nature and scope of the change being evaluated, be based on the as-built and as-operated and maintained plant, and reflect the operating experience at the plant."

NFPA 805 Section 2.4.4, "Plant Change Evaluation" states: "A plant change evaluation shall be performed to ensure that a change to a previously approved fire protection program element is acceptable. The evaluation process shall consist of an integrated assessment of the acceptability of risk, defense-in-depth, and safety margins. The impact of the proposed change shall be monitored."

NFPA 805 Content requirements include:

- (2.7.1.1) "The analyses performed to demonstrate compliance with this standard shall be documented for each nuclear power plant (NPP). The intent of the documentation is that the assumptions be clearly defined and that the results be easily understood, that results be clearly and consistently described, and that sufficient detail be provided to allow future review of the entire analyses. Documentation shall be maintained for the life of the plant and be organized carefully so that it can be checked for adequacy and accuracy either by an independent reviewer or by the AHJ."

- (2.7.1.2) "A fire protection program design basis document shall be established based on those documents, analyses, engineering evaluations, calculations, and so forth that define the fire protection design basis for the plant. As a minimum, this document shall include fire hazards identification and nuclear safety capability assessment, on a fire area basis, for all fire areas that could affect the nuclear safety or radioactive release performance criteria defined in Chapter 1."

- (2.7.1.3) "Detailed information used to develop and support the principal document shall be referenced as separate documents if not included in the principal document."

NFPA 805 configuration control requirements include:

- (2.7.2.1) "The design basis document shall be maintained up-to-date as a controlled document. Changes affecting the design, operation, or maintenance of the plant shall be reviewed to determine if these changes impact the fire protection program documentation."

- (2.7.2.2) "Detailed supporting information shall be retrievable records. Records shall be revised as needed to maintain the principal documentation up-to-date."

Finally, NFPA 805 quality requirements apply to use of integration databases and software:

- (2.7.3.1) "Each analysis, calculation, or evaluation performed shall be independently reviewed."

- (2.7.3.2) "Each calculational model or numerical method used shall be verified and validated through comparison to test results or comparison to other acceptable models."

- (2.7.3.3) "Acceptable engineering methods and numerical models shall only be used for applications to the extent these methods have been subject to verification and validation. These engineering methods shall only be applied within the scope, limitations, and assumptions prescribed for that method."

- (2.7.3.4) "Cognizant personnel who use and apply engineering analysis and numerical models (e.g., fire modeling techniques) shall be competent in that field and experienced in the application of these methods as they relate to nuclear power plants, nuclear power plant fire protection, and power plant operations." "An uncertainty analysis shall

be performed to provide reasonable assurance that the performance criteria have been met."

Given the above range of requirements applicable to the use of integration databases and software, provide the following information:

a. Address how the post transition change evaluation process ensures that the potential interfaces between integration databases and software, and other databases and analyses (e.g., the cable and raceway database, the NSCA, the fire probabilistic risk assessment (FPRA), and fire modeling) are evaluated and updated, as appropriate.

b. Discuss the process that will be employed to ensure that integration databases and software are maintained in accordance with documentation and design configuration control processes and procedures.

c. Describe the processes and procedures that will be used to ensure that integration database and software analyses are conducted and updated by persons properly trained and experienced in its use.

d. Describe the processes and procedures that will be used to ensure that integration database and software analyses comply with NFPA 805 modeling, content, and quality control requirements.

SSA RAI 02

For Criteria 3.1.2.6.2 [B], "Cooling Systems" to Attachment B to the submittal describe how the need and availability of the heating, ventilation and air conditioning (HVAC) systems at the Primary Control Stations (PCS) (Hot Shutdown Panel) have been evaluated. If the HVAC system is not credited, provide the analysis or justification to demonstrate that monitoring/control can be feasibly performed at the PCS.

SSA RAI 03

On page G-9 of Attachment G clarify discrepancies in the component numbers for Battery Chargers 1A & 1B (OP-RECOV-RCBC-A & B and OP-RECOV-BCSW-A & B vs. Q1R42E0001A & B) and for 125volt direct current (VDC) Bus 1A & 1B (OP-RECOV-RCBC-A & B and OP-RECOV-BCSW-B vs. Q1R42B0001A & B) between Attachment G and calculation SE-C051326701-11, Revision 1, "NFPA 805 Recovery Actions Feasibility Evaluation which is referenced in Attachment G of the LAR.

SSA RAI 04

It appears that Pages G-14 and G-26 of Attachment G, have duplicate records for components Q1R42B0001B and Q2R42B0001B. Clarify whether the duplication was intentional, and address any omissions, as appropriate.

SSA RAI 05

In Attachment G, recovery actions (RAs) for the pressurizer pressure operated relief valves (PORV) Q1B31V0053 (Page G-9) and Q2B31V0053 (Page G-21) involve actions to open instrument air (IA) valves or lines. Describe how these actions and the resulting equipment line up will satisfy the instrument air availability requirement to perform the credited function of these valves.

SSA RAI 06

In Attachment G, RAs for several safe shutdown components (Q1N11PV3371A, Page G-8; Q1N11PV3371C, Q1B31V0053, Page G-9; Q1N11PV3371A, Q1N11PV3371B, Q1N11PV3371A, Page G-14; Q2B31V0061, Q2N11PV3371A, Page G-19; Q2N11PV3371A, Page G-20; Q2N11PV3371C, Page G-21; Q2N11PV3371B, Q2N11PV3371C, Page G-25) involve actions to align the emergency air system to support the credited function of the component. Describe the analysis or justification that demonstrates the emergency air compressor and associated tubing remain free from fire damage.

SSA RAI 07

In Attachment G, RAs for several safe shutdown components (Q1B31V0061, Page G-10; Q1B31V0061, Page G-14; Q2B31V0061, Page G-21; Q2B31V0061, Q2B31V0053, Page G-25) involve actions to align nitrogen supply to support the credited function of the component.

a. If a fixed nitrogen system is credited, provide the analysis or justification to demonstrate that the system and associated tubing remain free from fire damage.
b. If a portable nitrogen system is credited, describe how safe and stable can be achieved and maintained without the need for recharging the nitrogen bottles.

SSA RAI 08

In Attachment G – RAs for several safe shutdown components (Q1R21E0009F, Q1R42B0001B, Page G-8; Q1R42B0001A, Q1R42E0001B, Q2R42BOO01A, OP-RECOV-RCBC-A, OP-RECOV-RCBC-B, OP-RECOV-BCSW-B, OP-RECOV-BCSW-A, Page G-9; Q1R42B0001B, Page G-14; Q2R42B0001B, Page G-19; Q2R42BOO01A, Q2R42B0001B, Page G-20; Q2R42B0001B, Page G-21; Q2R42B0001A, Q2R42E0001B, Page G-25; Q2R42B0001 B, Q2R42B0001A, Page G-26) involve actions to provide alternate cooling for the Battery Charger Room. Describe how alternate cooling is achieved and provide the supporting analysis or justification that demonstrates the adequacy of the alternate cooling method.

SSA RAI 09

In Approval Request 1 in Attachment L, one basis for the approval request is stated as follows: "[b]y eliminating cables with potential shorts, this eliminates ignition sources and therefore the jacketing of cable is not relevant." Describe the process and criteria for identifying cables with potential shorts, and how these cables are eliminated. Clarify whether there are modifications or implementation items associated with these actions.

SSA RAI 10

For Table B-3 of Attachment C, and the Table G-1 Attachment G, clarify the following:

a. RAs are identified as the resolution of certain variance from deterministic requirements (VFDRs) in Table B-3. However, many of these RAs are not included in Table G-1. Address the differences and clarify whether all RAs credited in the VFDR resolutions are reflected in Table G-1. Provide a justification for the omitted RAs or provide revision to Table G-1, as appropriate.
b. Many components associated with RAs occurring at the PCS as identified in Table G-1 were not found among the components associated with VFDRs requiring RAs in their corresponding fire areas as identified in Table B-3. Provide a discussion for the differences.

SSA RAI 11

Describe the methodology used to evaluate NFPA 805 defense-in-depth (DID) and safety margin during transition. Describe changes to the plant or plant procedure(s), if any, to maintain the philosophy of NFPA 805 DID and safety margin.

SSA RAI 12

Describe the NFPA 805 safe and stable condition and the additional resources and actions, if any, that are credited for maintaining this condition. Also, provide a description for the following:

a. Systems and components (e.g., accumulators and tanks) that require replenishment.
b. Time-critical actions necessary to maintain safe and stable conditions.
c. Post-fire actions credited to achieve and maintain safe and stable that are not frequently performed by trained operators.

SSA RAI 13

Describe how non-power operation (NPO) pinch points were evaluated and resolved for each fire area. Provide a description of any modifications or actions being credited to minimize the impact of fire-induced spurious actuations of power-operated valves (e.g., air operated valves (AOVs) and motor operated valves (MOVs)) during NPO either as pre-fire conditioning or as required during fire response RAs. Describe how the RAs feasibility analysis is performed and whether these actions have been or will be factored into operating procedures.

SSA RAI 14

Item 6 of Attachment S to the submittal, states that for Fire Area 2-021 an interposing relay and fuse will be installed to protect cable 2VYDG15J from fire induced failure and to prevent the breaker from tripping. It also states that for Fire Area 1-021 a fuse will be installed for cable 1VBJ5012F to prevent fire damage and that for Fire Area 2-041 a fuse will be installed for cable 2VAJ5007L to prevent fire damage. Provide clarification that the installation of the relays and fuses is for mitigating the secondary effects of cable damage and not to protect the cables from fire damage.

Probabilistic Risk Assessment (PRA) RAI 01

In Enclosure 6 to the supplement dated December 20, 2012 (ADAMS Accession No. ML12359A051), the results are presented for both the total and delta core damage frequency (CDF) that are actually lower than previously reported in Attachment W of the LAR. The submittal, although only the credit for the electrical cabinet factor was removed. With the additional removal of credit for the main control room (MCR) very early warning fire detection system (VEWFDS), it is expected that these CDF results would increase, consistent with the increases in the large early release frequency (LERF) values. Explain why, including any key modeling assumptions that may be relevant, the increases in total and delta-CDF are now lower than before especially in light of the higher increases in total and delta-LERF.

From the submittal, address the following:

a. In Table S-2, Plant Modifications Committed, Item 1 cites modifications to panels in Fire Area 044, the MCR. Discuss the extent to which the high risk rank is dependent upon the risk reduction being credited from the VEWFDS installation.
b. In Section V.2, Sensitivity of Fire PRA Methods, specifically in Tables V.2-2 through V.2-4, discuss whether there are any delta-risk values sensitive to crediting risk reduction from installing VEWFDS in the MCR. If so, discuss how they would change if this credit were removed.
c. In Table V-1, Fire PRA Peer Review – Facts and Observations, with respect to Supporting Requirement (SR) FQ-A3, discuss whether the installation of VEWFDS was credited after updating the analysis for the MCR using Appendix L of NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," including how much risk reduction is being realized. Also, with respect to SR FSS-B2, discuss to what extent the conclusion of insensitivity is dependent upon this credit.
d. In the LAR Table W-1, Summary of Total Plant Risk, and in Enclosure 3 to the letter dated September 25, 2012, Item 3 on page 2, two of the sensitivity analyses (in LAR Attachment V) increase the CDF by ~2E-5/yr, bringing the total close to 1E-4/yr. Discuss whether all values here, both these values and the sensitivities, have taken some risk reduction credit for installing VEWFDS in the MCR. If total CDF becomes >1E-4/yr or total LERF > 1E-5/yr, address any ramifications due to these changes relative to the guidelines on total risk in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."
e. In LAR Attachment W, Table W-2, Fire Initiating Events Individually Representing at least 1% of Calculated CDF for Unit 1, discuss whether an MCR fire scenario should appear among the dominant ones. Include consideration that LAR Table W-6 indicates an MCR abandonment CDF of ~3E-7/yr, which could rise to dominate the others, if credit for MCR VEWFDS is not taken.
f. For Fire Areas U1 044 and U2 044, in LAR Table W-6, FNP Fire Area Risk Summary, discuss why CDF equals LERF and delta-CDF equals delta-LERF. Also, discuss what these values would be without the credit being taken for installation of VEWFDS in the MCR.
g. In LAR Table W-6, FNP Fire Area Risk Summary, discuss how the total risk and delta-risk estimates, including those for RAs, would change if VEWFDS credit in the MCR were removed.

h. Calculation SE-C051326701-008, Farley Nuclear Plant, Units 1 and 2, NFPA 805 Fire Risk Evaluations, Version Number 1, dated September 25, 2012, was discussed during the site audit. In Tables 2-1a through 2-2b of the calculation, Fire PRA Variant Case Results, (Non-)Abandonment Trains A (and B) Alignment, three non-suppression probabilities are assumed - 1, 0.1 and 0.02. Discuss the bases for the latter two including their use with respect to crediting VEWFDS installed in-cabinet in the MCR panels.

PRA RAI 02

Discuss how the evaluation includes the possible increase in heat release rate (HRR) caused by the spread of a fire from the ignition source to other combustibles. Summarize how suppression is included in the evaluation.

PRA RAI 03

Transient fires should at a minimum be placed in locations within the plant physical access units (PAUs) where conditional core damage probabilities (CCDPs) are highest for that PAU, i.e., at "pinch points." Pinch points include locations of redundant trains or the vicinity of other potentially risk-relevant equipment, including the cabling associated with each. Transient fires should be placed at all appropriate locations in a PAU where they can threaten pinch points. Hot work is assumed to occur in locations where hot work is a possibility, even if improbable (but not impossible), keeping in mind the same philosophy. Discuss how transient and hot work fires are distributed within the PAUs. In particular, identify the criteria which determine where an ignition source is placed within the PAUs. Also, if there are areas where no transient or hot work fires are located since those areas are considered inaccessible, define the criteria used to define "inaccessible." Note that an inaccessible area is not the same as a location where fire is simply unlikely, even if highly improbable.

PRA RAI 04

Discuss the methodology that was used to evaluate DID and safety margins. The description should include what was evaluated, how the evaluations were performed, and what, if any, actions or changes to the plant or procedures were taken to maintain the philosophy of DID or sufficient safety margins.

PRA RAI 05

The transition report discuss and justifies an initial coping time of 24 (48, 72) hours, after which, actions are necessary to maintain safe and stable beyond 24 (48, 72) hours. Provide a discussion of the actions necessary beyond 24 (48, 72) hours to maintain safe and stable conditions such as refilling fluid tanks or re-aligning systems. Evaluate quantitatively or qualitatively the risk associated with the failure of actions and equipment necessary to extend safe and stable beyond 24 (48, 72) hours given the post-fire scenarios during which they may be required.

PRA RAI 06

a. Section 10 of NUREG/CR-6850, Supplement 1, states that a sensitivity analysis should be performed when using the fire ignition frequencies in the Supplement instead of the fire ignition frequencies provided in Table 6-1 of NUREG/CR-6850. Summarize the details and provide the results of the sensitivity analysis of the impact on using the Supplement 1 frequencies instead of the Table 6-1 frequencies on CDF, LERF, delta (Δ) CDF, and ΔLERF for all of those bins that are characterized by an alpha that is less than or equal to one.

b. Calculation PRA-BC-F-11-017, Joseph M. Farley Nuclear Plant, Units 1 and 2, Farley Fire PRA Summary Report, Version Number 1, dated September 14, 2012, was discussed during the site audit. With respect to p. D-3, Table D-1, Uncertainty and Sensitivity Matrix: Regarding Task 8, discuss the uncertainty analysis performed based on the updated fire frequencies from Supplement 1 of NUREG/CR-6850. Discuss whether this went beyond just performing the sensitivity evaluation in part (a), i.e., discuss whether a parametric uncertainty evaluation was performed using distributional parameters for each bin represented in the CDF and LERF. Summarize the details and report the results.

c. On page D-17, Section D.3.1, Fire Ignition Bin Frequencies, of the Calculation PRA-BC-F-11-017 (as referenced above), although it is recognized that the sensitivity analyses performed here were done on an earlier FPRA model, one would still expect the ratio of the resultant CDFs in this table (8.43E-5/4.66E-5 = 1.81) to be roughly the same or even slightly lower than the corresponding ratio reported in Table V.2-3 of the LAR, namely 7.46E-5/5.24E-5 = 1.42. Similarly the delta-CDFs should be roughly the same, with the value in Table V.2-3 (2.22E-5/yr) perhaps slightly larger than the one here (3.77E-5/yr). Neither case is true. It is recognized that the two sensitivity evaluations may be different, namely the one here addressed ALL the bin frequencies while that in Table V.2-3 addressed only the selected bins with alpha parameters of 1.0 or less. If that is the explanation, extract the results from here, with appropriate adjustments to reflect the final FPRA model, and include in the LAR with a discussion of the difference between the two sensitivity analyses. If there is another explanation, provide it.

PRA RAI 07

a. Discuss how CDF and LERF are estimated in MCR abandonment scenarios. Discuss whether any fires outside of the MCR cause MCR abandonment because of loss of control and/or loss of control room habitability. Discuss whether screening values for post MCR abandonment were used (e.g., CCDP of failure to successfully switch control to the PCS and achieve safe shutdown of 0.1) or have detailed analyses including human reliability analysis (HRA) been completed for these activities. Provide further justification for any screening value via the results of the human failure event (HFE) quantification process discussed in Section 5 of NUREG-1921 (EPRI TR 1019196), "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," November 2009, considering the following:

b. The results of the feasibility assessment of the operator action(s) associated with the HFEs, specifically addressing each of the criteria discussed in Section 4.3 of NUREG-1921.

a. The results of the process in Section 5.2.7 of NUREG-1921 for assigning scoping human error probabilities (HEPs) to actions associated with switchover of control to an alternate shutdown location. The bases for the answers to each of the questions asked in Figure 5-4 should be addressed.
b. The results of the process in Sections 5.2.8 of NUREG-1921 for assigning scoping HEPs to actions for performing alternate shutdown once switchover is complete. The bases for the answers to each of the questions asked in Figure 5.5 should be addressed.
c. The results of a detailed HRA quantification, per Section 5.3 of NUREG-1921 in place of items 2 and 3 if a CCDP as low as 0.1 (and conditional large early release probability (CLERP) as low as 0.01) is not attainable through the scoping approach. For the detailed study, quantify the contribution via the evaluation of different scenarios upon MCR evacuation, including the sum of those scenarios in the results for the CCDP and CLERP.
d. Provide the results of a sensitivity analysis that shows the impact on the PRA results (CDF, LERF, ΔCDF, ΔLERF) of using the resultant CCDP/CLERP analysis for control room abandonment scenarios.

PRA RAI 08

a. It was stated at the 2010 industry fire forum that the Phenomena Identification and Ranking Table (PIRT) Panel being conducted for the circuit failure tests from the DESIREE-FIRE and CAROL-FIRE tests may be eliminating the credit for Control Power Transformers (CPTs) (about a factor 2 reduction) currently allowed by Tables 10-1 and 10-3 of NUREG/CR-6850, Vol. 2. Provide the results of a sensitivity analysis that removes this CPT credit from the PRA, showing the impact of this potential change on CDF, LERF, ΔCDF, and ΔLERF.

b. Calculation PRA-BC-F-11-017, Joseph M. Farley Nuclear Plant, Units 1 and 2, Farley Fire PRA Summary Report, Version Number 1, dated September 14, 2012, was discussed during the site audit. In light of part (a), update the estimates from the table on page D-17, Section D.3.2, Spurious Operation Probabilities, using the latest FPRA model, including an update to Table V.2-4 of the LAR.

PRA RAI 09

a. Attachment W to the submittal provides the ΔCDF and ΔLERF for the VFDRs for each of the fire areas, but it does not discuss either generically or specifically how ΔCDF and ΔLERF were calculated. Discuss the method(s) used to determine the changes in risk reported in the Tables in Appendix W. The description should include a summary of PRA model additions or modifications needed to determine the reported changes in risk. If any of these model additions used data or methods not included in the FPRA Peer Review, discuss the additions. Also, discuss how the Fire Risk Evaluations (FREs) considered modifications, fire procedures, and recovery actions in the determination of risk evaluations.

b. Step 3 of Section 5.3, Performance of Fire Risk Evaluation, on page 11 of Calculation SE -C051326701-008, Farley Nuclear Plant, Units 1 and 2, NFPA 805 Fire Risk Evaluations, Version Number 1, dated September 25, 2012 states that the "compliant" case returned a basic event to its nominal (random, non-fire-induced) probability. For the variant case, confirm that either or both of the following calculational techniques were employed. (1) The fire-induced failure probability was set to one (always failed), (2) The variant value included

BOTH the fire-induced failure probability AND the random failure probability via a Boolean sum which essentially reduced to the fire-induced failure probability when much greater than its random counterpart. Discuss any other calculational techniques that were used. If so, re-estimate the delta-risk, addressing any ramifications due to these changes.

c. LAR Attachment V, Pages V-5 through V-8, Section V.2, Sensitivity of Fire PRA Methods, indicate that, when crediting future plant modifications for the hypothetical compliant case, the future configuration of the plant post-modifications is not represented, but instead represents a hypothetical current version that is deterministically compliant. It is concluded that, post-modifications, the delta-risks will be lower than currently estimated. Discuss how the delta-risks have been calculated. Also, in Section V.2.3, clarify whether the conclusion is cited against the absolute or relative (percentage) increase in risk.

PRA RAI 10

If new RAs (not including post MCR abandonment which is addressed elsewhere) have been credited in the risk estimates, discuss how instrument failure is addressed in the HRA.

PRA RAI 11

Confirm that the peer reviews for both the internal events PRA (IEPRA) and FPRA considered the clarifications and qualifications from RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," to American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard. If not, provide a self-assessment of the PRA model for the RG 1.200 clarifications and qualifications and indicate how any identified gaps were dispositioned.

PRA RAI 12

Identify if any VFDRs involved performance-based evaluations of wrapped or embedded cables. If applicable, discuss how wrapped or embedded cables were modeled in the FPRA including assumptions and insights on how the PRA modeling of these cables contributes to the VFDR delta-risk evaluations.

PRA RAI 13

Provide a discussion of the procedure(s)/process(es) for plant change evaluations post-transition. Include a discussion on how post-transition guidance for plant change evaluations addresses key uncertainties, assumptions, sensitivity analyses, and peer review Facts and Observations (F&Os) (e.g., unaddressed F&Os).

PRA RAI 14

Identify any implementation items in Attachment S of the submittal that have not been completed but which have been credited directly or indirectly in the change-in-risk estimates provided in LAR Attachment W. When an implementation item has been included in the PRA but not yet implemented, the models and values used in the PRA are necessarily estimates

based on current plans. The as-built facility after implementation is completed may be different than the plans. Address how Southern will ensure that upon completion of all PRA credited implementation items, the validity of the reported change-in-risk will be verified. This discussion should include a plan of action should the as-built change-in-risk exceed the estimates reported in the LAR. Also, with respect to page S-2 of Attachment S, discuss how the current FPRA, and the version reviewed by the peer reviewers, ensures that the plant configuration that will be complete in November of 2017 is bounded by the PRA model used for this transition.

PRA RAI 15

a. Discuss whether the changes to the seismicities as a result of the United States Geological Society (USGS) re-evaluation for the central and eastern US (USGS, "2008 NSHM Gridded Data, Peak Ground Acceleration"), based on reanalysis of the New Madrid earthquakes, were considered in determining the applicability of the seismic-fire interaction analysis performed for the Individual Plant Examination for External Events (IPEEE) to the current state of seismic-fire interactions. In Section 6.3.1, "Estimate Bounding (Seismic) CDF," page 21 of Calculation SE-C051326701-008, Farley Nuclear Plant, Units 1 and 2, NFPA 805 Fire Risk Evaluations, Version Number 1, dated September 25, 2012, a value of 1.08E-5/yr is reported as the average seismic CDF based on Table 6-1a, excluding the Maximum Spectral Results (MSRs). This appears to be the average of all hazard source and calculational approaches, excluding the MSRs (nine values). Provide a sensitivity analysis, which recalculates the average using only the 2008 USGS results and discuss the ramifications given this higher value. Also provide a sensitivity study for the seismic LERF in Section 6.3.2, Qualitatively Evaluate Bounding (Seismic) LERF Contribution. Ensure any needed changes to Attachment V, pages V-5 through V-8, Section V.2, "Sensitivity of Fire PRA Methods," and Table W-1, "Summary of Total Plant Risk" to Attachment W are appropriately addressed.

b. In Section 2.0, "Evaluation" on page 3 of Calculation PRA-BC-F-11-006, Joseph M. Farley Nuclear Plant, Units 1 and 2, Farley Fire PRA Task 13, Seismic-Fire Interactions Assessment, Version Number 1, dated September 20, 2011, it is stated that the analysis focuses on:

> ...ignition sources that have a seismic failure mode at very low accelerations and that are not present in the absence of a seismic event ... and the plant response to such fires considering the effect that the seismic event could have on the detection and suppression capabilities.

Discuss how changes to the seismicities at these lower accelerations as a result of the USGS re-evaluation, discuss are considered, at least qualitatively, in determining the applicability of the seismic-fire interaction analysis performed for the IPEEE to the current state of seismic-fire interactions.

PRA RAI 16

For Calculation PRA-BC-F-11-014, Joseph M. Farley Nuclear Plant, Units 1 & 2, Fire Scenario Development, Version Number 2, dated September 14, 2012, address the following:

a. Assumption 3 on pages 2 and 2-7 in Section 2.2, Assumptions (also see pages 13-1, Section 13.1.1, MCR Panels), states that, "For fires within the MCR [Main Control Room] and equipment rooms, the fire is expected to be limited within the panel due to fire detection and suppression by operations personnel." Appendix S of NUREG/CR-6850 provides recommendations regarding fire spreading between adjacent cabinets. Discuss how Assumption 3 bounds these recommendations, including any quantitative basis.

b. Assumption 4 on pages 2 and 2-7 in Section 2.2, Assumptions (also see pages 13-3, section 13.2.1, MCR Abandonment Times), claims conservatism if only half the panels in the MCR Abandonment Times analysis involve multiple cable bundles, because all such fires will start in a single bundle and need to spread to damage additional bundles. Justify this assumption that only half involve multiple cable bundles, including any quantitative basis, or provide the results of a sensitivity analysis, summarizing the calculational details, that considers all panels.

c. On page 13-3, Section 13.2.1, MCR Abandonment Times, two "distinct" areas are identified for the MCR, the operator ("front panel") area and the back panel/equipment area. These are "somewhat isolated" by panels, partitions, doors and walls that restrict interchange of gas or smoke. However, it appears that, despite these two areas being "distinct," the same MCR non-suppression probability (NSP) curve is applied based on sharing of the same HVAC system. During the walk-down as part of the site audit, the staff observed that the MCR "front panel" area appears to effectively be completely isolated from the "back panel" area (for each unit), at least in terms of the ability of the operators (or even staff in the Central Access Station [CAS]) to readily detect fire-related phenomena in the back panel area. Provide the results of a sensitivity evaluation, summarizing the details of the calculation, where an appropriate NSP curve is used for fires in the back panel/equipment area to determine the effect on the fire risks and delta-risks, addressing any ramifications due to changes in these estimates.

d. On page 13-4, Section 13.2.1, MCR Abandonment Times, there appears to be an implicit assumption for selecting an NSP = 0 for MCR abandonment scenarios where the time required for abandonment is "so great that it is implausible that the fire would not be put out before abandonment is required." For such a time to be "so great, etc.," it would have to exceed a minimum of 25 minutes. Discuss this apparent assumption.

e. Some of Tables 13-1 through 13-12, MCR Abandonment Scenarios, contain non-zero NSP values that are <0.001, despite the discussion in Section 13.2.1, MCR Abandonment Times, about not using these. Substitute NSPs = 0.001 for all such cases and recalculate the results as a sensitivity evaluation to determine the effect on the fire risks and delta -risks, addressing any ramifications due to changes in these estimates, or explain the apparent discrepancies.

f. On page 13-17, Section 13.2.1, MCR Abandonment Times, it appears the equation for IGF_{AB} weights equally the abandonment non-suppression probabilities for single and multiple cable bundles, for both the scenarios with and without HVAC. If so, discuss whether this is a consequence of the assumption that half the MCR electrical cabinet fires will involve single bundles and the other half multiple bundles. If so, provide the results of

a sensitivity evaluation where this assumption is relaxed (i.e., all cabinets involve multiple bundle fires) to determine the effect on the fire risks and delta-risks, addressing any ramifications due to changes in these estimates.

g. On page 13-17, Section 13.2.1, MCR Abandonment Times, when calculating the Transient severity factor SF_{AB}, discuss whether there is any contribution included from transient fires in the MCR and Unit 1 and 2 equipment rooms themselves. If not, and in light of the previous observation that transient fires must be postulated in these rooms as well, provide a sensitivity evaluation to include their contribution to determine the effect on the fire risks and delta-risks, addressing any ramifications due to changes in these estimates.

The following two items refer to Attachment 1 of the Calculation.

h. In all cases analyzed in Tables 6-1 through 6-7, Section 6.0, "Model Results," it appears that the limiting (shortest) abandonment time is 3.6 min (Tables 6-4 through 6-6). Discuss how this conclusion was reached and whether the time assumed is for the most limiting case in light of the assumed 15-minute available response time for the Fire Brigade.

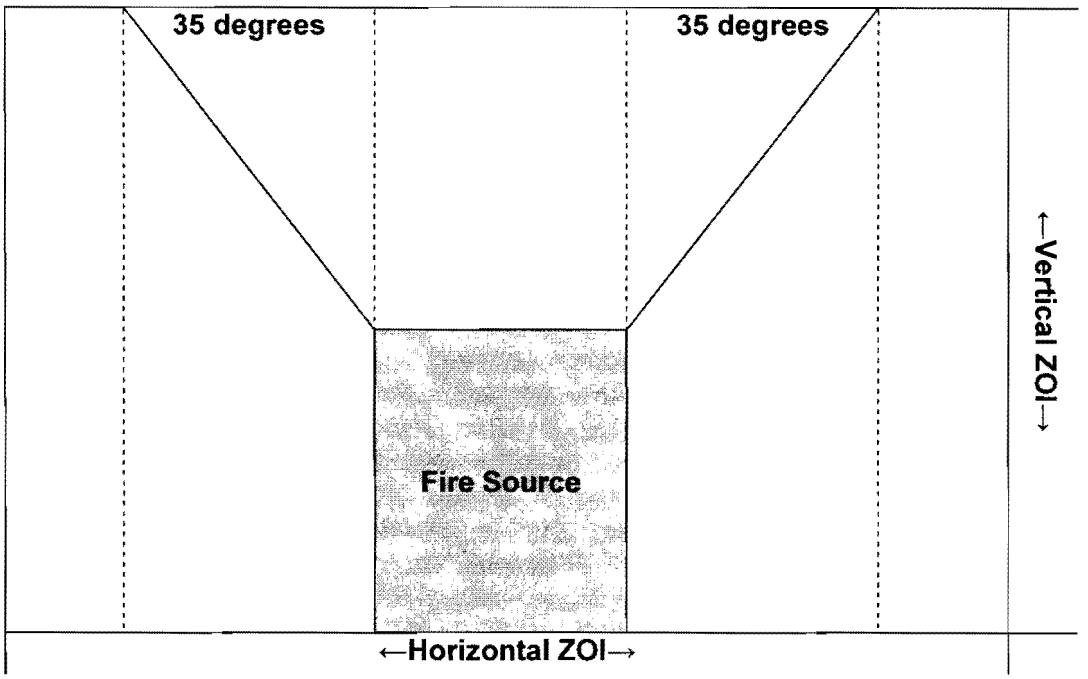
i. On page 47, Section 7.0, "Discussion and Conclusions," the FPRA apparently considers the ability of the operators to don self-contained breathing apparatuses (SCBAs). If this ability is assumed, discuss how it is modeled and credited.

PRA RAI 17

For Calculation PRA-BC-F-11-014, Joseph M. Farley Nuclear Plant, Units 1 & 2, Fire Scenario Development, Version Number 2, dated September 14, 2012, address the following:

a. On pages 14-1 and 14-2, Section 14.0, "Use of Generic Fire Modeling Treatments vs. Detailed Fire Modeling," potential conservatisms present in the Hughes Approach are credited as a basis for not performing detailed fire modeling. There appears to be reliance on information available from the EPRI Fire Events Database, complete only through 2000 and currently being updated, as justification. Discuss the potential impact on this justification in light of the fact that such data may be incomplete and, therefore, non-conservatively adapted outside the consensus approach of NUREG/CR-6850.

b. On page 4-1, Section 4.0, "Damage Criteria," a "cone of damage" based on the zone of influence (ZOI) for an ignition source extended to the ceiling is cited as conservative because it includes additional targets that would be susceptible to damage from fire spread to secondary combustibles and the resulting increased HRR. Discuss whether this extended ZOI associated with the HRR from only the ignition source was assumed to envelope any fire spread beyond the characteristic 35-degree vertical "cone" discussed in Appendix R of NUREG/CR-6850 (see figure below). In addition, explain how, even after considering the flame spread rates recommended in Appendix R of NUREG/CR-6850, the assumed ZOI continues to bound any expansion beyond the cone until the fire is suppressed.



c. For Bin 21 Oil Spills (page 7-2, Table 7-1, "Ignition Source Typical Severity Factor Values;" page 8-8, Figure 8-1, "Event Tree Non-Suppression Probability for Pump Fire"), the split fractions were expected to correspond to the values provided by the NRC in its June 21, 2012, letter from Joseph Giitter, NRC, to Biff Bradley, Nuclear Energy Institute, regarding Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, "Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires," (ADAMS Accession No. ML12171A583). Specifically, 88 percent of the spill fires are limited to the pump; 7 percent produce spills of 10 percent capacity; and 5 percent produce spills of 100 percent capacity. Provide the results of a sensitivity analysis using these updates. (For the event tree in Figure 8-1, with only two branches, when applied specifically to Bin 21, split fractions of 95 percent for non-severe and 5 percent for severe may be used.)

d. With respect to page 10-3, Section 10.5, Potential for Structural Collapse, provide the detailed equation and quantification for the CDF contribution from turbine building collapse, including, at least as a sensitivity evaluation, any needed modifications to address the NRC change in split fraction as discussed above in relation to the June 21, 2012, letter from Joseph Giitter, NRC, to Biff Bradley, Nuclear Energy Institute, regarding Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, "Evaluation of Peak Heat Release Rates in electrical Cabinet Fires," (ADAMS Accession No. ML12171A583).

PRA RAI 18

For Calculation PRA-BC-F-11-015," Joseph M. Farley Nuclear Plant, Units 1 and 2, Hot Gas Layer and Multi-Compartment Analysis for Farley Fire PRA," Version Number 1, dated September 14, 2012, address the following:

a. With respect to page 1, Summary of Conclusions; Section 3.1, "Purpose (MCA)," on page 3-1, discuss whether a hot gas layer (HGL) is the only phenomenon addressed when considering the potential for multi-compartment damage via the spread of fire or fire phenomena. Discuss how barrier breaches via open or degraded penetrations are considered.

b. On page 1, "Purpose of Calculation"; Section 3.2, "Methodology (MCA)," on page 3-1, a damage threshold temperature of 329 degrees Celsius is cited, which is that for thermoset cables. Discuss whether there are other potential targets that could be damaged by an HGL at lower temperatures, such as thermoplastic cables or sensitive electronics. If so, discuss how these are addressed in the HGL/multi-compartment analysis (MCA) evaluation.

c. On page 2-1, Section 2-2, "Methodology; Attachment A, Scenario Data Used in Analysis," three of the 98th percentile HRRs from Table E-1 in NUREG/CR-6850 are cited as being modeled by the Hughes Generic Treatment: 69 kilowatt (kW) (motors), 317 kW (transients) and 464 kW (vertical cabinets with multiple, unqualified cables and closed doors). The highest 98th percentile HRRs for vertical cabinets, 702 kW (multiple, qualified cables) and 1002 kW (multiple, unqualified cables with open doors), were apparently dismissed. Explain the basis. Also, NUREG/CR-6850, Section R.4.2.2, recommends that the first cable tray be ignited not necessarily at 5 min, but at the time to damage/ignition based on a plume temperature correlation. Address whether the 5-min assumption is appropriate on a phenomenological basis.

PRA RAI 19

For Calculation PRA-BC-F-11-016, Joseph M. Farley Nuclear Plant, Units 1 and 2, Human Reliability Analysis for Fire Events, Version Number 1, dated September 13, 2012, addresses the importance analysis (e.g. page 1-2, Section 1.0, "Purpose;" page 2-1, Section 2.1, "Existing FPIE PRA Operator Actions;" and, page 2-23, Section 2.2, "New Post-Fire Safe Shutdown Operator Actions." The use of the computer code FRANC may preclude the ability to perform importance analysis relative to fire risk if, rather than modify the fault trees to include fire-specific basic events, such as modified or even new HFEs, it replaces the existing basic events from the internal events model with fire-specific values via flag files. Discuss whether the technique of adding "-F" to the internal event HFE basic event ensures that an automated importance calculation retains the fire-specific values that are assigned. Consider the statement that "[t]hese actions exist only in the fire PRA; they are not included in the FPIE PRA model," which suggests that the importance calculation may not be representative. Explain how this importance assessment is done to ensure that the importances for the HFEs are evaluated relative to the fire risk, as an automated importance calculation may not necessarily use these replaced values, nor would FRANC necessarily consider the fire-specific initiating frequencies, but rather only the conditional core damage probability relative to a fire ignition frequency of 1.

PRA RAI 20

For Calculation PRA-BC-F-12-004, Joseph M. Farley Nuclear Plant, Units 1 and 2, Fire PRA Sensitivity Analysis, Version Number 0, dated September 21, 2012, address the following:

a. There was a discussion regarding Table 3-1, "Summary of Refined Scenarios." As a spot check on the sensitivity evaluation, Table A-1, "Fire Scenarios by CDF Contribution (Train A)," of PRA-BC-F-11-017, "Fire PRA Summary Report," was reviewed to identify all scenarios with CDF of at least 1.00E-7 for which a non-unity NSP and/or severity factor (SF) was listed. The NRC staff's review indicates a significant difference only for scenario 0163-E1 (ratio ~5), and indicate one scenario where the sensitivity evaluation lowered the result (0335-J2h). (Note also for this scenario that no suppression or SF credit was applied in the "base" case, yet suppression was credited in the "sensitivity" case, which appears contrary to the process where only scenarios which previously had credited the SF were re-evaluated.) Also, there appears to be removal of any credit for the SF, (i.e., SF now set to 1.0, but crediting of suppression for several scenarios where suppression had previously been credited), such that the new product of NSP x SF is essentially the same as before (0.04 x 0.25 = 0.01), e.g., 0229-J1, 0233-I1, 0335-J1 and 0343-F1. Also note that in all cases but that for 0347-F1, the CCDP was the same for both cases. However, it is lower for the sensitivity case in this particular scenario by roughly a factor of 3 (highlighted in yellow). Provide a detailed explanation of how these sensitivity evaluations were performed and why the aggregate indicates only a 20 percent increase in CDF. Include the reason why suppression was credited in both cases for several of these.

b. Assumption 5 on page A-2, cites NUREG/CR-6850 as the source for presuming that fires are likely to be manually detected, in the absence of automatic detection, within 15 minutes of the fire starting. Provide this citation from NUREG/CR-6850 or justify this assumption based on plant-specific fire response testing, both announced and unannounced, and actual fire response history. Note that Section P.2 of NUREG/CR-6850, which assumes a time to delayed detection of 15 minutes in the absence, or failure, of automatic detection, is only an example.

c. Regarding Assumption 7 on page A-2, explain the basis and technique by which the damage times reported as a function of heat flux in Appendix H of NUREG/CR-6850 can be converted to a "percent of damage function." Include a discussion of the sensitivity cases cited to show that any non-conservatism does not significantly impact the calculated NSPs.

PRA RAI 21

For Calculation SE-C051326701-008, "Farley Nuclear Plant, Units 1 and 2, NFPA 805 Fire Risk Evaluations," Version Number 1, dated September 25, 2012, address the following:

a. For the abandonment cases in Tables 2-1a through 2-2b, "Fire PRA Variant Case Results, (Non-)Abandonment Trains A (and B) Alignment," three severity factors are assumed - 5.02E-4, 4.84E-4 and 0.00158. Discuss the bases for these cases. If taken from Figure L-1 in NUREG/CR-6850, where the maximum value of 0.00158 suggests the minimum distance assumed between targets in an MCR panel is 1.0 m for qualified cables or 2.0 m for unqualified cables, discuss the basis for choosing these distances for assignment of severity factors.

c. In Tables 2-1a and 2-1b, "Fire PRA Variant Case Results Train B Alignment," in Attachments – FREs for Unit 1 Fire Area 1-021 and 1-041, several scenarios appear to credit both the 0.25 probability of fire spread beyond an electrical cabinet and a non-suppression probability (NSP) of 0.04. Explain the assignment of these dual factors to the scenarios, addressing any potential for "double-crediting" of suppression.

PRA RAI 22

For Calculation PRA-BC-F-11-002, Joseph M. Farley Nuclear Plant, Units 1 and 2, Component Selection for Farley Fire PRA, Version Number 3, dated June 29, 2012, address the following related to fire area risk evaluations.

a. From page 15, Section 2.6, Inclusion of Potentially High Consequence Related Equipment:"
   i. The screening frequency of 1E-7/yr ensures that no core damage sequence with a frequency > 1E-7/yr will be *a priori* eliminated. However, since the type of spurious operations considered here could also lead directly to a large early release, e.g., via an interfacing systems loss of coolant accident (ISLOCA), this screening value does not ensure that a large early release > 1E-8/yr will be *a priori* retained (assuming the traditional factor of 10 difference between CDF and LERF). Discuss why the screening threshold is not lowered to account for LERF.
   ii. Also, with regard to this same screening criterion, the term "judged" with respect to the numerical value is used. Discuss whether this implies that the screening process is qualitative, or if some at least bounding quantitative estimate is produced before eliminating component failures.

b. From page 19, Section 3.0, "Finalization of Equipment Selection:"
   i. The "N1" and "N3" disposition codes state that the basic event may be ignored. Discuss whether the basic event is still retained in the FPRA model, given that it could still be manifested as a random failure in a sequence induced by a fire.
   ii. Discuss whether all common-cause failures (CCFs) were assigned an "N1" disposition code *a priori*, or only if they could not be affected by a fire. If the former, discuss whether the CCFs dispositioned as N1 in Appendix F could be fire-affected (e.g., motors/pumps fail to start or run). CCFs are all assigned as N1, but have been retained.

c. From Appendix F, "Basic Event Mapping and Disposition:"
   i. Discuss the basis for assigning an "N1" code to items dispositioned as post-initiator human actions, including ones specifically cited as occurring during a fire event.
   ii. Explain why some post-initiator operator actions are dually dispositioned as N1 and YO3, where the second is specifically cited as "during a fire," while others are not (i.e., both are retained as N1). For example (there are others), compare 1HHOA-DOOR-H and ...H-F on page F-104 with 1HHOP88038XFRH and ...H-F on page F-105.
   iii. "Valve rupture" (e.g., 1SWMV506-I-R on p. F-155) is dispositioned as "N1." For any such designation, discuss whether the valve (an MOV) has been checked for IN 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire" protection, such that the rupture could not be fire-induced.

PRA RAI 23

For Calculation PRA-BC-F-11-004, Joseph M. Farley Nuclear Plant, Units 1 and 2, Fire PRA Logic Model for Farley Nuclear Plants, Version Number 2, dated June 29, 2012, address the following:

a. On pages 36-39, Section 2.7, "HVAC Systems;" Table 3, "Summary of Fire Zones for HVAC Modeling; Appendix D, Modeling of HVAC Systems," the Vogtle PRA is cited as a basis for choosing a threshold temperature of 150 degrees Fahrenheit (degrees F) over 24 hours for functional failure of equipment in a room. This is noticeably higher than the typical values, especially when electronics may be involved. Discuss the basis for this selection. Provide an extract from any relevant material directly from the Vogtle PRA reference. Note that Appendix D implies 125 degrees F as the maximum temperature permitted during normal and post-accident operation in the service water intake structure (SWIS) Pump Room, and 104 degrees F in the Battery Charger Room during normal operation, with 114 degrees F the estimated limit after 30 days of accident condition operation. While the FPRA Peer Review issued only a Suggestion (page 54, Table 4-18, "Fire PRA Peer Review Facts and Observations," LTR-RAM-II-12-007 [March 12, 2012]) related to this topic, potentially significant deficiencies were cited, such as solid state relays and certain instruments with environmental design temperatures below 150°F. Further indication is that the basis behind the Vogtle justification lies in generic references.

b. On page 41, Section 2.8, "Incorporate Results of HRAs," it is stated: "[also note that a dependency analysis should be performed to identify cutsets that might be truncated by multiple HFEs. This analysis would be best performed after all the operator actions and fire recoveries are identified and evaluated." Discuss whether an HRA dependency analysis has been performed and if not, explain why not, and discuss how the potential effect on the fire risks and delta-risks is bounded in the FPRA model being applied to NFPA-805 transition.

c. Pages 113-114, Section 2.9.9, Start-up Transformer Cooling, discuss how the PRA models the running of the start-up transformer above its non-emergency rating for extended periods of time without including the self-cooling ability. Discuss whether its failure probability is increased to reflect this "more stressful" operating condition. If not, explain, and discuss the potential effect on the fire risks and delta-risks from this increased failure likelihood and whether it is bounded in the FPRA model being applied to NFPA-805 transition. It is recognized that there may be no detrimental effects on long-term service reliability at nominal conditions; however, discuss whether there could be an increased failure probability for short-term "emergency" operation.

PRA RAI 24

For Calculation PRA-BC-F-11-009, Joseph M. Farley Nuclear Plant, Units 1 and 2, Plant Partitioning and Fire Ignition Frequency for Farley Fire PRA, Version Number 3, dated June 29, 2012, address the following:

a. On page 15, Section 2.2, "Fire PAU Identification," an open space at elevation 189' in the Turbine Bldg (Operating Deck) is designated as a three-hour fire barrier. Discuss the

basis for this designation, including how this "barrier" is credited, if at all, during multi-compartment analysis.

b. On page 75, Section 3.3.2, "Fixed Ignition Source Generic Fire Frequency Allocation," cites a "simplified approach compared to that discussed in NUREG/CR-6850" for assigning weighting factors (i.e., all set = 2) that is used for all locations. Table 6-2 in NUREG/CR- 6850 recommends deriving these factors specifically for each of the different types of plant locations. Discuss how the results via this "simplified approach" compare to those that would be derived using the guidance recommended by NUREG/CR-6850 for each particular location type. Indicate if, as suggested by the sample calculation on page 115 of Section 3.5, "Fire Ignition Frequency Calculations," the "simplified approach" conservatively over-estimates some frequencies.

PRA RAI 25

For the following Fire Areas in Table C-2, "Table 4-3 NFPA-805 Required Fire Protection Systems and Features" of Attachment C to the submittal, discuss whether there should be a risk requirement in one or more of the listed Fire Zones since each area lists both delta-CDF and delta-LERF from VFDRs in Table W-6 of Attachment W to the submittal:

1. 1-014, p. 34
2. 1-030-U2, page 39
3. 1-039, page 41
4. 1-054, page 44
5. 1-055, page 45
6. 1-076-U2, page 46
7. 1-079, page 46
8. 1-081-U2, page 48
9. 1-094, page 50
10. 1-DU-DGSWIS-A-U1, page 51
11. 1-DU-DGSWIS-B-U2, page 52
12. 1-DU-DGVB-A, page 53
13. 1-DU-DGVB-B, page 53
14. 1-S01, page 53
15. 1-S10, page 54
16. 2-014, page 76
17. 2-017, page 77
18. 2-031-U1, page 80
19. 2-039, page 82
20. 2-054, page 86
21. 2-075-U1, page 86
22. 2-076-U1, page 87
23. 2-079, page 88
24. 2-S01, page 94
25. ABRF-U1, page 99
26. ABRF-U2, page 99
27. DU-SWISVB-A-U1, page 100
28. DU-SWISVB-A-U2, page 100

29. DU-SWISVB-B-U1, page 100
30. DU-SWISVB-B-U2, page 101
31. SWWPVB-A-U1, page 101
32. SWWPVB-A-U2, page 102
33. SWWPVB-B-U1, page 102
34. SWWPVB-B-U2, page 102
35. YARD-U1, page 103
36. YARD-U2, page 104

PRA RAI 26

From Table U-1, "Internal Events PRA Peer Review – Facts and Observations," of Attachment U to the submittal address the following:

a. SR IE-A7. Explain whether this is more than a documentation issue, since it appears the PRA model was changed.
b. SR SC-A2. Because fire can induce CCFs of components that are credited for mitigation, but do not fail in the IEPRA, discuss whether the potential for such fire-induced failures "downstream" of the initiator have been considered, such that the effect on the fire CDF could be larger than the estimated 3 percent for the internal CDF.
c. SR SY-A8. Discuss whether these "sub-components" of the diesel generator (DG) are modeled separately, such that fire-induced failures of the DG trigger specific sub-components, vs. any "super-component" consisting of the DG in its entirety. Discuss how this modeling in the IEPRA translates into the FPRA.
d. SR HR-D2. Internal event HFEs that can be affected by fire often experience increases in their HEPs. Given the use of an overly low HEP as a screening value in the IEPRA, discuss whether any HFEs were screened out that might not merit such exclusion in the FPRA due to increased HEP.
e. SR HR-G7. Discuss the extent to which the guidance in NUREG-1921, "Fire HRA Guidelines," is considered in the specific HRA dependency analysis for fire.

PRA RAI 27

With reference to Section V.2.1.2, "Sensitivity on Use of Electrical Cabinet Fire Severity," of Attachment V on page V-4, explain what is meant by the "Unit 1 Train A and B analysis," as there appear to be multiple versions of the FPRA. Note that SR IGN-A9 in Table V-1, Fire PRA Peer Review Facts and Observations of Attachment V to the submittal, implies each train runs ~50 percent of the time. Discuss whether this is correct, such that averaging the results is representative.

PRA RAI 28

The following refer to Table V-1, "Fire PRA Peer Review – Facts and Observations:"

a. SR FQ-D1. Discuss the value (conditional containment failure probability), and how it compares to that for internal events. Provide an explanation if the value is significantly higher.

b. SR FSS-A2. Discuss whether the exclusion of targets outside the fire compartment for full room burnout scenarios was covered by the multi-compartment analysis. If not, explain why not.
c. SR FSS-C2. If more detailed fire modeling were employed, such as the computer codes CFAST or FDS, other options to the 12-minute growth rate could be available. Discuss to what extent limiting the detailed analysis to the Hughes Generic Method affects the potential results that could be obtained from more detailed fire modeling.
d. SR FSS-D7. Discuss whether this is to meet CC-II, for which there is also a requirement that the credited system has not experienced outlier behavior relative to system availability. If not, confirm that only CC-I can be met, at most.
e. SR FSS-D8. Discuss whether fire propagation was assumed and, if not, whether the targets damaged in the fire may be underestimated.
f. SR FSS-E3. With regard to developing statistical representation of uncertainty intervals and mean values for fire modeling parameters, note that NUREG-1824, "Verification and Validation of Selected Fire Models," was available, as well as draft NUREG-1934, "Fire Modeling Analysis Guidelines," to aid in considering uncertainty in fire modeling. Discuss whether these were considered and if not, explain why not. Confirm that this SR is satisfied only at the CC-I level.
g. SR FSS-H5. Discuss whether these "discussions" include a parametric uncertainty evaluation as required by CC-II (e.g., using NUREG-1824 or NUREG-1934). If not, then address whether or not this SR is met and, if so, whether at the CC-I level, at most.
h. SR IGN-A7.
i. Even if not changing the target sets, discuss whether the frequency increase will have a non-negligible effect.
j. Discuss how the suggestion to examine indoor transformers associated with essential lighting, etc., was addressed for bin 23, particularly if any transformers are rated over 45 kVA.
k. Discuss the validity of the bin 15 frequency for each PAU, i.e., 8/(N+16), where N is the number of vertical sections for all cabinets throughout the plant other than the B4 SWGR room cabinet.

PRA RAI 29

As Table V-2 of Attachment V to the submittal, was omitted, confirm that only the following SRs fail to be categorized as at least CC-II (or met, if these in no breakdown of levels): FSS-H5, FSS-E3, FSS-D7, FSS-C2 and FSS-C1. If there are others, identify and discuss why meeting a capability category less than CC-II is acceptable for the NFPA-805 transition for each SR.

PRA RAI 30

Regarding Scenario 0401/E in Table W-3, "Fire Initiating Events Individually Representing at least 1% of Calculated LERF for Unit 1," to Attachment W to the submittal, Table W-6 indicates MCR Abandonment has a total LERF = 3.02E-7, yet this seemingly included scenario, which appears to be a contributor to the total, has a higher value for its LERF than the total itself. Indicate whether it is part of the non-abandonment scenario, with a total LERF = 3.71E-7.

PRA RAI 31

Regarding Scenario 0401/E in Table W-5, "Fire Initiating Events Individually Representing at least 1% of Calculated LERF for Unit 2," in Attachment W to the submittal, Table W-6 indicates MCR Abandonment has a total LERF = 3.47E-7, yet this seemingly included scenario, which appears to be a contributor to the total, has a higher value for its LERF than the total itself. Discuss whether it is part of the non-abandonment scenario, with a total LERF = 3.71E-7. Also, address the fact that there are now three scenarios with a total LERF contribution ~1.2E-6 (add 0471/AT and 0401/AG).

PRA RAI 32

The following refer to Attachment W, Table W-6, FNP Fire Area Risk Summary:

a. CDFs and LERFs (and their delta counterparts) <1E-12 are typically below truncation limits of the PRA quantification software. Discuss whether the values listed are below truncation limits for the fire CDF or LERF (and their delta counterparts), and whether they should be listed as "epsilons" rather than given extremely low quantified values. [Note that the use of a few epsilons suggests truncation limits of 1E-13 (CDF) and 1E-17 (LERF), where the former are conceivable, but the latter are not.]
b. For Fire Area U1 1-040, discuss why CDF equals LERF and delta-CDF equals delta-LERF.
c. For Fire Area U2 2-040, discuss why CDF equals LERF and delta-CDF equals delta-LERF.

PRA RAI 33

For Calculation SE-C051326701-008, Farley Nuclear Plant, Units 1 and 2, NFPA 805 Fire Risk Evaluations, Version Number 1, dated September 25, 2012, pages 31 (Section 2.0, "Risk Evaluation)" and 33 (Section 2.1, "Fire PRA Variant Case Calculation") in Attachment – "Fire Risk Evaluation (FRE) for Unit 2 Fire Area 044," address the following:

a. If an MCR panel is open rather than being separated by double walls and an air gap from an adjacent panel (as indicated during the plant walk-down), discuss the basis for assuming rapid enough detection and manual suppression prior to fire spread into the adjacent cabinet for a fire initiating at the postulated boundary.
b. Discuss whether the abandonment of the MCR due to other than environmental conditions within the MCR itself was considered, e.g., due to loss of functionality from fire damage outside the MCR. If not, explain why not.
c. Provide the quantitative basis for concluding that a CCDP/CLERP = 0.1 bounds all operator actions required for alternate shutdown. In particular, address the reference to Fire Areas 1-021 and 1-041 with regard to this value, where it appears that the maximum CCDP/CLERP for scenarios in these areas is 0.292 (see Tables 2-1a and 2-1b in each). If any values > 0.292 were used, identify these, including the areas to which they were applied.

Monitoring Program RAI 01

Section 4.6 of the LAR Transition Report (Enclosure 1 to letter dated September 25, 2012, ADAMS Accession No. ML12279A235) describes the post transition NFPA 805 monitoring program. Review of the submittal determined that a description of how systems, structures, and components (SSCs) that are already within the scope of the 10 CFR 50.63 (i.e. Maintenance Rule) program will be addressed was not included. Provide a description of how SSCs that are already within the scope of the Maintenance Rule program will be addressed with respect to the NFPA 805 monitoring program.

Programmatic RAI 01

Describe the specific documents that will comprise the post transition NFPA 805 fire protection program licensing basis.

Describe whether documents, analyses, designs, and engineering reviews prepared to support the NFPA 805 fire protection program are managed as controlled documents as described in Section 6, "Document Control" of the Quality Assurance Topical Report, SNC-1.

Programmatic RAI 02

Describe the changes that are planned to the fire protection program (FPP) as a part of the NFPA 805 transition process, including associated training and identification of the positions of any such training necessary to support the program changes.

Programmatic RAI 03

Describe how the training program will be revised to support the NFPA 805 change evaluation process, including positions that will be trained and how the training will be implemented (e.g., classroom, computer-based, reading program).

Programmatic RAI 04

Describe how the combustible loading program will be administered to ensure that FPRA assumptions regarding combustible loading are met.

Programmatic RAI 05

Describe whether you have committed to conducting future NFPA 805 analyses in accordance with each of the requirements of NFPA 805 Section 2.7.3, Compliance with Quality Requirements.

Fire Modeling (FM) RAI 01

Section 2.4.3.3, "Fire Risk Evaluations," of NFPA-805 , states: "[t]he PSA [probabilistic safety assessment] approach, methods, and data shall be acceptable to the AHJ [authority having jurisdiction] ... " The NRC staff noted that fire modeling comprised the following: 1) the

Consolidated Fire Growth and Smoke Transport (CFAST) model was used to calculate control room abandonment times, and 2) the Generic Fire Modeling Treatments approach was used to determine the ZOI in all fire areas throughout plant.

Section 4.5.1.2, "Fire PRA" of the LAR states that fire modeling was performed as part of the FPRA development (Section 4.2.4.2 of NFPA 805). Reference is made to Attachment J, "Fire Modeling V&V," for a discussion of the acceptability of the fire models that were used.

Specifically regarding the acceptability of CFAST for the control room abandonment time study:

a. Provide the basis for the assumption that the fire brigade is expected to arrive within 15 minutes. Describe the uncertainty associated with this assumption, discuss possible adverse effects of not meeting this assumption on the results of the FPRA and explain how possible adverse effects will be mitigated.

b. Provide justification for using transient fire growth rates that are different from those specified in FAQ 08-0052, (ADAMS Accession No. ML092120501 closure memo) and discuss the effect of these deviations on the fire risk and delta-risk. Note that, Table H-1, "NEI 04-02 FAQs" utilized in LAR Submittal," in Attachment H of the submittal credits FAQ 08-0052.

c. Provide justification for using the heat of combustion measured in the oxygen bomb calorimeter for Teflon (6,200 kiloJoule per kilogram (kJ/kg) and Tefzel (12,600 kJ) instead of the effective (or chemical) heat of combustion values (2,000 kJ/kg and 7,300 kJ/kg, respectively) in Tewarson's chapter of the Society of Fire Protection Engineers (SFPE) handbook.

d. During the audit, NRC staff observed several large plastic trash cans against a wall of the Technical Support Center, (Fire Area 044-U1, Unit 1 Control Room Complex and Technical Support Center) along the path to the MCR, and a stack of large plastic containers with personal protective equipment along the back wall of the MCR back panel area. Provide assurance that the fires involving these combustibles are bounded by the fire scenarios in the equipment area that were considered in the CFAST abandonment time analysis and that combustibles are controlled to these bounding limits after transition.

With respect to the sensitivity analysis in Appendix B of calculation PRA-BC-F-11-014, Attachment 1, "Evaluation of Units 1 and 2 Control Room Abandonment Times at the Joseph M. Farley Nuclear Plant, September 18, 2012:

e. Explain how the results of the sensitivity analysis were used in the FPRA.

f. It is stated in Section B.3 that "... poorly ventilated burning conditions could have a small effect on the abandonment times." Provide justification for this statement in light of the fact that in several cases the abandonment time is reduced by more than ten minutes, as shown in Table B-3.

Specifically regarding the acceptability of the Generic Fire Modeling Treatments approach described in Kleinsorg Group Risk Services, LCC, "Generic Fire Modeling Treatments," Revision 0, report:

g. Explain how the modification to the critical heat flux for a target that is immersed in a thermal plume described in Section 2.4 of the Generic Fire Modeling Treatments document was used in the ZOI determination.

h. Provide technical justification to demonstrate that the Generic Fire Modeling Treatments approach as used to determine the ZOI of fires that involve multiple burning items (e.g., an ignition source and an intervening combustible such as a cable tray) is conservative and bounding.

i. Describe how the flame spread and fire propagation in cable trays and the corresponding HRR of cables was determined. Explain how the flame spread, fire propagation and HRR estimates affect the ZOI determination and HGL temperature calculations.

j. Describe how transient combustibles in an actual plant setting are characterized in terms of the three fuel package groupings in the Generic Fire Modeling Treatments, Supplement 3, (Transient Ignition Source Strength). Identify areas, if any, where the NUREG/CR-6850 transient combustible HRR characterization (probability distribution and test data) may not encompass typical plant configurations. Finally, explain if any administrative action will be used to control the type of transient in a fire area.

k. During the audit, NRC staff observed plastic trash cans throughout the areas that were walked down, typically in a corner next to an elevator door on virtually every floor of the plant. Provide assurance that fires involving these trash cans are bounded by the transient fire scenarios that were considered in the fire modeling analysis of the areas where the trash cans are located.

Regarding the acceptability of the PSA approach, methods, and data in general:

l. Address how it was assured that non-cable intervening combustibles were not missed in areas of the plant. Provide information on how intervening combustibles were identified and accounted for in the fire modeling analyses and the FREs.

m. The discussion in Section 6.1 of calculation PRA-BC-F-11-014, "Fire Scenario Development," Joseph M. Farley Nuclear Plant, Units 1 & 2, Version Number 2, dated September 14, 2012, indicates that in some cases a peak HRR for transient combustibles of 69 kW was used instead of 317 kW. Describe the criteria used for selecting the lower HRR over the NUREG/CR-6850 98th percentile for transient combustibles.

n. Section 10.2 of calculation PRA-BC-F-11-014, "Fire Scenario Development," Joseph M. Farley Nuclear Plant, Units 1 & 2, Version Number 2, dated September 14, 2012, discusses cable tray fire propagation. At the end of the section it is stated that "... the analysis is based on the cables being thermoset, Institute of Electrical and Electronics Engineers Standard 383,"Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations" (IEEE-383) qualified. Therefore, vertical

fire propagation of cable trays was not required, however in most cases when the bottom tray of a vertical stack was within the ZOI the fire was assumed to propagate vertically to all trays within that stack."

Explain why vertical fire propagation of trays with thermoset, IEEE-383 qualified cables "was not required". Under what conditions, in cases when the bottom tray was within the ZOI, was fire propagation in a vertical stack of cable trays not assumed.

o. Section 10.3 of calculation PRA-BC-F-11-014, "Fire Scenario Development," Joseph M. Farley Nuclear Plant, Units 1 & 2, Version Number 2, dated September 14, 2012, discusses fire location factor and indicates that "these factors" are documented in Appendix B. However, Appendix B only includes adjusted ZOI dimensions for transient fires within 2 ft of a wall or corner. Explain how wall and corner effects were accounted for in the case of combustible liquid and electrical cabinet fires.

p. During the audit, NRC staff observed transformers filled with approximately 300 gallons of oil in several of the areas. It appears that in the fire scenarios that involve these transformers, they are assumed to be dry transformers and contributed lower HRR than oil filled transformers. Justify this assumption and explain why the transformers were not treated as oil-filled.

Fire Modeling RAI 02

Section 2.5, "Evaluating the Damage Threshold," of NFPA 805, requires damage thresholds be established to support the performance-based approach. Thermal impact(s) must be considered in determining the potential for thermal damage of SSCs. Appropriate temperature and critical heat flux criteria must be used in the analysis.

Section 4.1 of calculation PRA-BC-F-11-014, "Fire Scenario Development," Joseph M. Farley Nuclear Plant, Units 1 & 2, Version Number 2, dated September 14, 2012, states that, "a review of the cable specifications for FNP indicates that both thermoset and thermoplastic cables are found at FNP. However, it has been determined that the amount of thermoplastic cables routed in trays is minimal (6% of total tray routed cables) compared to the amount of thermoset cables. Therefore the cable damage threshold that will be considered in the FPRA will be that of the IEEE-383 thermoset cable."

Provide the following information:

a. Address how the installed cabling in the power block was characterized, specifically with regard to the critical damage threshold temperatures and critical heat flux for thermoset and thermoplastic cables described in NUREG/CR-6850.

b. Section 2.0 of Generic Fire Modeling Treatments provides a discussion of damage criteria for different types of targets. Section 2.1 of the Generic Fire Modeling Treatments document states: "damage to IEEE-383 qualified cables is quantified as either an imposed incident heat flux of 11.4 kW/m^2 (1 $Btu/s\text{-}ft^2$) or an immersion temperature of 329°C (625°F) per Nuclear Regulatory Guidance [NRC, 2005, NUREG 6850, 2005]." Section 2.2 of Generic Fire Modeling Treatments states:

"Damage to non-IEEE-383 qualified cables is quantified as either an imposed incident heat flux of 5.7 kilowatt per meter square ($kW/m^{2)}$ (0.5 $Btu/s\text{-}ft^2$) or an immersion temperature of 204°C (400°F) per Nuclear Regulatory Guidance [NRC, 2005, NUREG 6850, 2005]."
The above statements from Generic Fire Modeling Treatments imply that IEEE-383 qualified cables are assumed to be equivalent in terms of damage thresholds to "thermoset" cables as defined in Table 8-2 of NUREG/CR-6850. In addition, non-IEEE-383 qualified cables are assumed to be equivalent to "thermoplastic" cables as defined in Table 8-2 of NUREG/CR 6850.

For those areas that are assumed to have thermoset damage criteria, confirm that the cables are actually thermoset and not just IEEE-383 qualified.

c. Explain how raceways with a mixture of thermoset and thermoplastic cables were treated in terms of damage thresholds.

d. Explain how the damage thresholds for non-cable components (i.e., pumps, valves, electrical cabinets, etc.) were determined. Identify any non-cable components that were assigned damage thresholds different from those for thermoset and thermoplastic cables.

e. It is stated in Section 4.1 of calculation PRA-BC-F-11-014, "Fire Scenario Development," Joseph M. Farley Nuclear Plant, Units 1 & 2, Version Number 2, dated September 14, 2012, that "if a detailed fire model is employed, and the actual target cables parameters (vendor specification and damage temperature) are known, a cable specific damage threshold temperature is applied." Discuss if there are any fire zones where this was done.

f. It is stated in Section 4.2 of calculation PRA-BC-F-11-014, "Fire Scenario Development," Joseph M. Farley Nuclear Plant, Units 1 & 2, Version Number 2, dated September 14, 2012 that "NUREG/CR-6850 recommends failure criteria for solid-state control components of 3 kW/m^2 (versus 11 kW/m^2 for IEEE-383 qualified cables and 6 kW/m^2 for non-IEEE-383 qualified cables) be used for screening purposes. However, given that the enclosure would provide protection to the sensitive internal contents from external fire effects, it is reasonable to apply the same zone of influence established for cable damage. The omission of the credit for the enclosure is judged to offset the non-conservatism of the damage threshold." Discuss the technical justification for using cable damage thresholds for temperature sensitive equipment inside cabinets.

g. Describe the basis for the assumption that all sensitive electronics are contained within "sealed" cabinets, and therefore presumably immune to smoke damage when the smoke originates from outside the cabinet.

Fire Modeling RAI 03

Section 2.7.3.2, "Verification and Validation," of NFPA 805, states: "each calculational model or numerical method used shall be verified and validated through comparison to test results or comparison to other acceptable models."

Section 4.5.1.2, "Fire PRA" of the LAR states that "fire modeling was performed as part of the Fire PRA development (NFPA 805 Section 4.2.4.2)." The LAR further states that "The acceptability of the use of these fire models is included in Attachment J."

Furthermore Section 4.7.3 "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805" of the LAR states that "Calculational models and numerical methods used in support of compliance with 10 CFR 50.48(c) were verified and validated as required by Section 2.7.3.2 of NFPA 805."

Regarding the V&V of fire models:

a. It is stated on page J-2 of Attachment J that "CFAST does not use a fire diameter, therefore, it is possible to specify a fire that falls within the range of Froude numbers considered in the NUREG-1824 validation documentation." Provide clarification/confirmation whether the Froude number based on the HRR and diameter of the fire being modeled is within the validated range for the CFAST model calculations used, or justify why CFAST could be used for Froude numbers outside the validated range.

b. It is stated on page J-2 of Attachment J that "[the] flame length ratio is normally met, but in the case of the largest fire sizes postulated, the flame height may reach or exceed the ceiling height. Because sprinkler actuation and thermal radiation to targets are not computed with the CFAST model, this parameter is not an applicable metric." Provide additional justification for using CFAST to model fires with flames that impinge on the ceiling.

Fire Modeling RAI 04

Section 2.7.3.3, "Limitations of Use," of NFPA 805, states "acceptable engineering methods and numerical models shall only be used for applications to the extent these methods have been subject to verification and validation. These engineering methods shall only be applied within the scope, limitations, and assumptions prescribed for that method".

Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," of the LAR states that "engineering methods and numerical models used in support of compliance with 10 CFR 50.48(c) were applied appropriately as required by Section 2.7.3.3 of NFPA 805."

Identify uses, if any, of the Generic Fire Modeling Treatments (including the supplements) outside the limits of applicability of the method and for those cases explain the analysis that was used or why the use of the Generic Fire Modeling Treatments approach was justified.

Fire Modeling RAI 05

Section 2.7.3.4, "Qualification of Users," of NFPA 805, states "cognizant personnel who use and apply engineering analysis and numerical models (e.g., fire modeling techniques) shall be competent in that field and experienced in the application of these methods as they relate to nuclear power plants, nuclear power plant fire protection, and power plant operations."

Section 4.5.1.2, "Fire PRA" of the LAR states that fire modeling was performed as part of the Fire PRA development (Section 4.2.4.2 of NFPA 805). This requires that qualified fire modeling and PRA personnel work together. Furthermore, Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," of the LAR states:

> Cognizant personnel who use and apply engineering analysis and numerical methods in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by Section 2.7.3.4 of NFPA 805.
>
> During the transition to 10 CFR 50.48(c), work was performed in accordance with the quality requirements of Section 2.7.3 of NFPA 805. Personnel who used and applied engineering analysis and numerical methods (e.g. fire modeling) in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by NFPA 805 Section 2.7.3.4.
>
> Post-transition, for personnel performing fire modeling or Fire PRA development and evaluation, SNC will develop and maintain qualification requirements for individuals assigned various tasks. Position Specific Guides will be developed to identify and document required training and mentoring to ensure individuals are appropriately qualified per the requirements of NFPA 805 Section 2.7.3.4 to perform assigned work. See Implementation Item in Table S-3 of Attachment S.

Regarding qualifications of users of engineering analyses and numerical models:

a. Describe what constitutes the appropriate qualifications for the staff and consulting engineers to use and apply the methods and fire modeling tools included in the engineering analyses and numerical models.

b. Describe the process/procedures for ensuring the adequacy of the appropriate qualifications of the engineers/personnel performing the fire analyses and modeling activities.

c. Provide the position and qualifications of the personnel who performed the walkdowns for the MCR (abandonment based on damage and inhabitability) and the remaining fire areas in the plant. Address whether the same people who performed walkdowns conduct the fire modeling analysis.

d. Explain the communication process between the fire modeling analysts and PRA personnel to exchange the necessary information and any measures taken to assure the fire modeling was performed adequately and will continue to be performed adequately during post-transition.

e. Explain the communication process between the consulting engineers and plant and corporate personnel to exchange the necessary information. Describe measures taken to assure the fire modeling was performed adequately and will continue to be performed adequately during post-transition.

Fire Modeling RAI 06

NFPA 805, Section 2.7.3.5, "Uncertainty Analysis," states: "An uncertainty analysis shall be performed to provide reasonable assurance that the performance criteria have been met."

Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," of the Transition Report states that "Uncertainty analyses were performed as required by 2.7.3.5 of NFPA 805 and the results were considered in the context of the application. This is of particular interest in fire modeling and Fire PRA development."

Regarding the uncertainty analysis for fire modeling:

a. Describe how the uncertainty associated with the fire model input parameters was accounted for in the fire modeling analyses.

b. Describe how the "model" uncertainty was accounted for in the fire modeling analyses.

c. Describe how the "completeness" uncertainty was accounted for in the fire modeling analyses.

Radiation Release RAI 01

Section 4.4 "Radioactive Release Performance Criteria" of Enclosure 1, "LAR Transition Report" to letter dated September 25, 2012, appears to sometimes refer to meeting limits for instantaneous release of radioactive effluents specified in the Technical Specifications (TS) and sometimes refers to meeting 10 CFR 20 limits (e.g., Attachment E, Conclusion where the criteria are based on meeting NFPA 805, 2001 edition). Address which standard is intended to be used.

Radiation Release RAI 02

Qualitative Assessment – Section 10.0 "Conclusion," to Radioactive Release Calculation SM -CO51326701-010, states that a qualitative review has been chosen versus a quantitative review in meeting the Radioactive Release criteria. However, additional detail is needed:

a. For fire areas with engineering controls that normally provide containment to mitigate effluent releases, provide a qualitative description of the source term (e.g., relative extent of plant contamination, likely effect of fire suppression activation on the contamination, the relative magnitude of potential release rates given the engineered structures, and potential offsite dose consequences).

b. For fire areas where there are no engineered barriers to provide containment of effluent releases, provide a quantitative description addressing the same type of information indicated above.

c. When radiation monitoring by radiation protection professionals is performed and a release is detected, discuss the types of administrative controls that will be used as mitigating actions to minimize a release (e.g., scrubbing, filtration, dikes, storm drain covers).

Should you have any questions, please feel free to contact me at (301) 415-2315.

Sincerely,

/RA/

Eva A. Brown, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Portal Document Request
2. RAI

cc w/enclosures: Listserv

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