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~~Withhold from public disclosure per 10 CFR 2.390.~~

SOUTHERN COMPANY

October 30, 2013

Docket Nos.: 50-348
50-364

NL-13-2039

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

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Regarding License Amendment Request for Transition to 10 CFR 50.48(c) – NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants

Ladies and Gentlemen:

By letter dated September 25, 2012, the Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) for Joseph M. Farley Units 1 and 2 (Ref. TAC NOS. ME9741 and ME9742). 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 (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.*

By letter dated December 12, 2012, the Nuclear Regulatory Commission (NRC) Staff requested supplemental information regarding the acceptance of the license amendment (Adams Accession No. ML12345A398). SNC provided the requested information by letter dated December 20, 2012. The NRC staff subsequently completed the acceptance review by letter dated January 24, 2013, (Adams Accession No. ML13022A158).

By Letter dated July 8, 2013, the NRC Staff formally transmitted a request for additional information (RAI) related to the referenced license amendment to be provided within 120 days. To facilitate NRC review, SNC provided a response to selected RAIs at 60 days, approximately 90 days, and the remainder by 120 days. By letter dated September 16, 2013, SNC provided the 60 day responses. The enclosures provide currently available responses to several remaining NRC RAIs. Responses to the remaining RAIs will be provided in accordance with the agreed upon completion dates. The attachments also contain a description of changes to the LAR that SNC has identified in connection with the RAI responses.

The No Significant Hazards Consideration determination provided in the original submittal is not altered by the RAI responses provided herein.

A006
NRR

This letter contains no new NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Mr. C. R. Pierce states he is Regulatory Affairs Director of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true and correct.

Respectfully submitted,

C. R. Pierce

C. R. Pierce
Regulatory Affairs Director

CRP/jkb/lac

Sworn to and subscribed before me this 30 *day of* October, *2013.*

Laura L. Crumpton
Notary Public

My commission expires: 10/8/2017

Enclosures:
1. Response to Fire Protection Engineering RAIs
2. Response to Safe Shutdown Analysis RAIs
3. Response to Probabilistic Risk Assessment RAIs
4. Response to Monitoring Program RAI
5. Response to Programmatic RAIs
6. Response to Fire Modeling RAIs
7. Response to Radiation Release RAIs

Attachments: Revisions to the Transition Report Main Body
Revisions to the Transition Report Attachment L
Revisions to Modifications and Implementation Items Attachment S

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. T. A. Lynch, Vice President – Farley
Mr. B. L. Ivey, Vice President – Regulatory Affairs
Mr. B. J. Adams, Vice President – Fleet Operations
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U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Regional Administrator
Ms. M. Barillas, NRR Project Manager - Farley
Mr. P. K. Niebaum, Senior Resident - Farley
Mr. J. R. Sowa, Senior Resident – Farley

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Regarding License Amendment Request for Transition to 10 CFR 50.48(c) – NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants

Enclosures and Attachments

Enclosure 1: Response to Fire Protection Engineering RAIs

Enclosure 2: Response to Safe Shutdown Analysis RAIs

Enclosure 3: Response to Probabilistic Risk Assessment RAIs

Enclosure 4: Response to Monitoring Program RAI

Enclosure 5: Response to Programmatic RAIs

Enclosure 6: Response to Fire Modeling RAIs

Enclosure 7: Response to Radiation Release RAIs

Attachment 1: Revisions to the Transition Report Main Body-Section 4.2.1.2

Attachment 2: No revisions

Attachment A: No Revisions

Attachment B: No Revisions

Attachment C: No Revisions

Attachment D: No Revisions

Attachment E: No Revisions

Attachment F: No Revisions

Attachment G: No Revisions

Attachment H: No Revisions

Attachment I: No Revisions

Attachment J: No Revisions

Attachment K: No Revisions

Attachment L: Revisions to Transition Report Attachment L – NFPA 805 Chapter 3 Requirements for Approval (10 CFR 50.48(c)(2)(vii)

Attachment M: No Revisions

Attachment N: No Revisions

Attachment O: No Revisions

Attachment P: No Revisions

Attachment Q: No Revisions

Attachment R: No Revisions

Attachment S: Modification and Implementation Items

Attachment T: No Revisions

Attachment U: No Revisions

Attachment V: No Revisions

Attachment W No Revisions

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information (60 Day Response)
Regarding License Amendment Request for Transition to 10 CFR 50.48(c)
NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants

Enclosure 1
Response to Fire Protection Engineering RAIs

Farley RAI FPE 01

The compliance strategy for Attachment A, Table B-1, Section 3.3.8 "Bulk Storage of Flammable and Combustible Liquids", to the License Amendment Request (LAR) dated September 25, 2012 (Agencywide Document Access and Management System (ADAMS) Accession Nos. ML12279A235, ML12279A009, and ML12279A012) is identified as "N/A" or not applicable. However, a compliance basis is provided. Discuss the compliance strategy.

SNC RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013

Farley RAI FPE 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).

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013

Farley RAI FPE 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 NFPA 805, Section 3.3.5.2, for non-embedded configurations are met. Also, provide the compliance basis for future installations of conduit embedded in concrete.

SNC RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013

Farley RAI FPE 04

LAR Attachment A, Table B-1, 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.

SNC RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013

Farley RAI FPE 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.

SNC RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FPE 06

Fire protection systems and features that require NFPA code compliance are reflected in Chapter 3 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.

SNC RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FPE 07

The compliance basis for LAR Attachment A, Table B-1, Section 3.3.5.3, "Electrical Cable Flame Propagation Limits," 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.

SNC RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FPE 08

The compliance basis for LAR Attachment A, Table B-1, Section 3.11.2, "Fire Barriers," 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, are met.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FPE 09

The compliance basis for LAR Attachment A, Table B-1, Section 3.3.1.3.4, "Control of Ignition Sources on Portable Heaters," states that procedures will be revised to "restrict portable fuel fired heaters from plant areas containing equipment important to nuclear safety", NFPA 805, Section 3.3.1.3.4, 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FPE 10

Various sections of LAR Attachment A, Table B-1 (e.g. 3.3.1.1, 3.3.7.1, 3.3.8, and 3.4.1) state that code compliance reviews have been performed to ensure the requirements of appropriate NFPA codes are met. Describe how any non-compliance identified during these reviews was addressed. If any non-compliances are still outstanding, describe how these will be addressed prior the completion of NFPA 805 implementation.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Regarding License Amendment Request for Transition to 10 CFR 50.48(c) – NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants

Enclosure 2
Response to Safe Shutdown Analysis RAIs

Farley Safe Shutdown Analysis (SSA) RAI – 01 Databases and Software

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.

RESPONSE:

The above range of requirements applicable to the use of integration databases and software overall used to perform and maintain the analyses that support the NFPA 805 Licensing Bases have the following configuration control and quality assurance processes applied:

a. 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. Checklists in the NMP-ES-035 series of procedures for reviewing proposed changes for impact on the Fire Protection Program (FPP) include (or will include) questions regarding potential impacts to the various databases used in the analysis. These include fundamental FPP elements from Chapter 3 of NFPA 805, NSCA and NPO analyses, Radioactive Release analysis, and Fire PRA analyses and supporting data. In this way, impacts to the various databases and analyses can be identified and updated as proposed changes to the plant are being developed. A new LAR Attachment S Table S-3 Implementation Item for revising the NMP-ES-035 series to ensure the above listed analyses are subject to these control processes is being added. This new Implementation Item was included in the September 16, 2013 SNC transmittal letter NL-13-1503, Attachment S, Table S-3 Update, Item 32, in response to Programmatic RAI 5.

b. 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 will use existing database configuration control procedures/processes. These are fleet level control processes that recognize the needs of controlled design inputs and verification. In LAR Section 4.7.3, it was stated, "During the transition to 10 CFR 50.48(c), FNP has and will continue to perform work in accordance with the quality requirements of Section 2.7.3 of NFPA 805." The PDMS, ARCPlus, and PRA databases are subject to the control procedures used to develop the NFPA 805 LAR. The software is subject to software QA requirements appropriate to the end use.

c. Processes and procedures will be used to ensure that integration database and software analyses are conducted and updated by persons properly trained and experienced in its use. LAR Section 4.7.1 discusses the training and qualification of current and potential future staff performing the analyses in support of the NFPA 805 Licensing Basis. Attachment S, Table S-3 item 29 of the LAR commits to creating the Position Specific Guides for personnel performing these analyses.

d. Processes and procedures will be used to ensure that integration database and software analyses comply with NFPA 805 modeling, content, and quality control requirements. In LAR Section 4.7.3, it was stated, "During the transition to 10 CFR 50.48(c), FNP has and will continue to perform work in accordance with the quality requirements of Section 2.7.3 of NFPA 805." The calculations performed for the NFPA 805 License Amendment Request were performed as augmented quality documents, which require independent review per SNC procedure. See response to RAI PROG 05.

Farley SSA RAI 02 – This RAI response will be provided with supplemental correspondence.

Farley SSA RAI 03 – This RAI response will be provided with supplemental correspondence.

Farley 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley SSA RAI 08 – This RAI response will be provided with supplemental correspondence.

Farley 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.

RESPONSE:

Approval Request 1 in Attachment L has been reworded as shown below. In specific response to the basis for request regarding "eliminating cables with potential shorts", this basis was not stated in the correct context in Approval Request 1. This statement was intended to indicate that power cables were routed in conduit in the Auxiliary Building corridors. This position is clarified in the reworded Approval Request.

There are no modifications associated with this Approval Request. There is an implementation item associated with Attachment A, Table B-1, Section 3.3.5.1. Implementation Item #6 states "Plant documentation will be revised to incorporate the requirements for electrical wiring above suspended ceiling limitations for all future installations."

In addition, although not requested in the Farley RAIs, the Approval Request was reworded to remove the discussion of IEEE-383 cable being equivalent to NFPA 262 requirements for plenum rated cable. This change is based upon RAIs asked on this topic from other licensees.

Reworded Approval Request 1:

Approval Request 1

NFPA 805 Section 3.3.5.1

NFPA 805 Section 3.3.5.1 states:

> *"Wiring above suspended ceiling shall be kept to a minimum. Where installed, electrical wiring shall be listed for plenum use, routed in armored cable, routed in metallic conduit, or routed in cable trays with solid metal top and bottom covers."*

Wiring above suspended ceilings at FNP has been kept to a minimum, but the wiring present may not comply with the other requirements of this code section.

The areas at FNP currently with suspended ceilings inside the NFPA 805 defined power block include:

- Auxiliary Building corridors and office areas
- Control Room/associated offices and the Technical Support Center
- Computer Room, Unit 1 and 2

A review of design drawings confirmed that there are cables above the ceiling in the Auxiliary Building corridors and office areas, the Control Room/associated offices and the Technical Support Center (TSC), and the Computer Rooms. The Computer Room area is not risk significant.

The Auxiliary Building corridors and office areas have pre-action systems installed above and below the suspended ceilings in areas where power cables are installed in trays without solid

metal covering. Areas that are not protected with a pre-action system above and below the ceiling contain only low voltage control and instrumentation cables, or power cables routed in metallic conduit. Thus, the potential for a cable short developing and becoming an ignition source is low.

The Control Room/associated offices and the TSC have power and control cabling installed above the suspended ceilings. The Control Room area is a continually staffed area, and the potential of an incipient fire developing is low. Only low power level circuits and circuits required to support Control Room systems are routed in the Control Room. This design, in conjunction with the absence of other significant sources of ignition, further reduces the potential of a Control Room fire. The areas outside the Control Room, such as the kitchen as well as surrounding rooms and the TSC, contain minimal power cables above the suspended ceiling, but due to the proximity to the occupied Control Room spaces, it is unlikely that a fire could develop undetected.

The Computer Rooms have total flooding Halon systems installed in the areas with suspended ceilings. The majority of cables installed in these areas are control and instrumentation cables, with a few 120V power cables installed. Thus, the potential for a cable short developing and becoming an ignition source is low. In addition, the presence of the Halon system ensures that the area is protected in the unlikely event that a fire develops.

Basis for Request:

The basis for the approval request of this deviation is:

- The wiring above ceilings in the spaces discussed above do not pose a hazard:
  - Low voltage wiring is not susceptible to shorts causing a fire.
  - The Main Control Room area is continually staffed, and the potential of an incipient fire reaching the smoldering stage is very low.
  - Auxiliary Building areas with pre-action systems above and below the suspended ceilings have additional protection against the threat of a fire. Areas without the pre-actions system contain only low voltage cables and compliant power cables.
  - Computer Room areas contain low voltage cables. In addition, the areas with Halon suppression systems have additional protection against the threat of a fire.

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The location of wiring above suspended ceilings does not affect nuclear safety. The wiring, while it may not be in armored cable, in metallic conduit, or plenum rated, is low voltage cable not susceptible to shorts that would result in a fire. Therefore there is no impact on the nuclear safety performance criteria.

The location of cables above suspended ceilings has no impact on the radiological release performance criteria. The radiological review was performed based on the potential location of radiological concerns and is not dependent on the type of cables or locations of suspended ceilings. The cables do not change the results of the radiological release evaluation performed that concluded that potentially contaminated water is contained and smoke is monitored. The cables do not add additional radiological materials to the area or challenge systems boundaries.

Safety Margin and Defense-in-Depth:

The amount of non-rated and non-enclosed wiring above the ceilings is minor and does not pose a significant fire hazard. Wiring, while it may not be in armored cable, in metallic conduit, or plenum rated, is low voltage cable not susceptible to shorts that would result in a fire. These

areas and the cables have been analyzed in their current configuration. Therefore, the inherent safety margin and conservatisms in these analyses remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The prior introduction of non-listed cables routed above suspended ceilings does not impact fire protection defense-in-depth. Echelon 1 is maintained by the cable installation procedures documenting the requirements of NFPA 805 Section 3.3.5.1. The introduction of cables above suspended ceilings does not affect echelons 2 and 3. The cables routed above suspended ceilings does not directly result in compromising automatic fire suppression functions, manual fire suppression functions, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the use of non-listed cables routed above suspended ceilings.

The engineering analysis performed determined that the performance-based approach utilized to evaluate a variance from the requirements of NFPA 805 Chapter 3:

(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
(B) Maintains safety margins; and
(C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

Farley 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.

RESPONSE:

Response to Item a.: No recovery actions are omitted in Attachment. G. The correlation of recovery action to VFDR was provided in SNC response to NRC Question 2, page E2-2, in the LAR Supplement, dated December 20, 2012.

Response to Item b.: All recovery actions and activities occurring at a primary control station (which are not recovery actions by definition) supporting the alternate shutdown fire areas are documented in Attachment G. There are no VFDRs related to activities occurring at a primary control station, since they are considered compliant. These activities at the primary control station, however, are included in Attachment G of the LAR and the December 20, 2012 LAR Supplement. The Fire Risk Evaluation (SE-C051326701-008) provides the source detail.

Farley 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley 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.

RESPONSE:

General

Consistent with NFPA 805 and supplemental guidance, Farley Nuclear Plant (FNP) Calculation SE-C051326701-003, "FNP NSEL and Safe and Stable Fault Trees," defines safe and stable conditions as the ability to maintain $K_{eff} < 0.99$ with a reactor coolant temperature at or below the requirements for hot standby. Demonstration of the Nuclear Safety Performance Criteria (NSPC) for safe and stable conditions was performed in two analyses:

- At-Power Analysis, which includes Modes 1-2 (This analysis is discussed in Section 4.2.4 of the LAR). The ability to achieve and maintain safe and stable conditions for the "At-Power Analysis" is demonstrated by the Nuclear Safety Capability Assessment (NSCA). The NSCA includes systems and equipment necessary to establish and maintain hot standby (Mode 3) or hot shutdown (Mode 4) conditions for an indefinite period of time.
- Non-Power Analysis, which includes Modes 3 and below (This analysis is discussed in Section 4.3 of the LAR). The ability to achieve and maintain safe and stable conditions for the "Non-Power Analysis" is demonstrated by the Non-Power Operations (NPO) Analysis. The NPO Analysis was performed in accordance with NFPA 805 and FAQ 07-0040, Revision 4. Consistent with this FAQ, the NPO Analysis includes systems and equipment necessary to transition from decay heat removal via a steam generator to decay heat removal via the residual heat removal (RHR) system, and all lower operating modes. The NPO analysis encompasses the following Plant Operating States (POSs), as defined by FAQ 07-0040:

  POS 1: This POS starts when the RHR system is put into service (Mode 3 to Mode 4). The Reactor Coolant System (RCS) is closed such that a steam generator could be used for decay heat removal, if secondary side of a steam generator is filled. The RCS may have a bubble in the pressurizer. This POS ends when the RCS is vented such that the steam generators cannot sustain core heat removal. This POS typically includes Mode 4 (hot shutdown) and portions of Mode 5 (cold shutdown).

POS 2: This POS starts when the RCS is vented such that: (1) the steam generators cannot sustain core heat removal and (2) a sufficient vent path exists for feed and bleed. This POS includes portions of Mode 5 (cold shutdown) and Mode 6 (refueling). Reduced inventory operations and mid-loop operations with a vented RCS are subsets of this POS.

POS 3: This POS represents the shutdown condition when the refueling cavity water level is at or above the minimum level required for movement of irradiated fuel assemblies within containment as defined by Technical Specifications. This POS occurs during Mode 6.

The FNP NSCA is documented in Calculation SE-C051326701-003, "FNP NSEL and Safe and Stable Fault Trees" and Calculation SE-C051326701-010, "NFPA 805 NSCA Transition Table B-3 – Fire Area Review." The analysis was developed in accordance with the NFPA 805 requirements and applicable FAQs. Section 1.5.1 of NFPA 805 identifies the pertinent Nuclear Safety Performance Criteria that are to be satisfied in order to "provide reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition." The criteria are as follows:

- Reactivity Control
- Inventory and Pressure Control
- Decay Heat Removal
- Vital Auxiliaries
- Process Monitoring

Attachment C of the FNP LAR contains results of the NSCA analysis as documented in Calculation SE-C051326701-010, "NFPA 805 NSCA Transition Table B-3 – Fire Area Review." These results document how each performance criteria is satisfied on a fire area basis. When applicable, VFDRs are identified for each performance goal in the B-3 table and provided with a disposition as appropriate.

Sustaining hot standby conditions (once achieved) for a prolonged time frame is accomplished by (1) ensuring a continuous source of water to at least one steam generator in support of natural circulation decay heat removal, (2) ensuring a source of inventory for makeup to the RCS, (3) ensuring positive RCS pressure control, and (4) ensuring continuous operation of at least one emergency diesel generator or availability of off-site power to supply AC power to the electrical distribution system. Capacity limits, resources, and actions supporting these functions are discussed in Parts a through c below.

Part a

The following systems/equipment require replenishment to support long-term hot standby conditions:

1. Condensate Storage Tank (CST) – The volume of the CST is adequate to sustain decay heat removal using one or more steam generators for approximately 24 hours, at which time additional inventory must be added to ensure a continuous source of water to the auxiliary feedwater (AFW) system. However, the NSCA also includes equipment required to directly supply water to the suction side of both the motor-driven AFW pump and turbine-driven AFW pump. The service water system takes suction from the pond, which is designed for 30 days of operation without makeup.

2. Refueling Water Storage Tank (RWST) – The RWST has sufficient capacity to maintain adequate primary inventory for an extended period of time, including cooldown to the transition point for RHR initiation. Inventory replacement is not anticipated to be necessary for over a week.

3. Emergency Diesel Generator (EDG) Fuel – Each EDG is equipped with its own day tank and is connected to a shared fuel oil storage and transfer system. This shared fuel oil storage and transfer system consists of five (5) underground Fuel Oil Storage Tanks (FOSTs) interconnected with piping, valves and redundant capacity fuel transfer pumps. The total delivered capacity from 4 tanks is sufficient to operate the required EDGs for a period of seven (7) days. There is a common header with the capability to connect any one of the five (5) storage tank pump discharges, whether it be the auto or manual transfer pump discharge, to any EDG day tank. The day tank provides over 3 hours of fuel without makeup. Each EDG has an associated FOST with an automatically controlled transfer pump that is normally aligned to the EDG's day tank. The automatically controlled transfer pump is powered from a MCC supplied by the associated diesel, while the manually operated pump is powered from a MCC associated with another diesel. The NSCA includes EDG fuel makeup to the day tank via the associated FOST's automatically controlled transfer pump. The associated underground FOST has sufficient inventory to maintain EDG operation for greater than three (3) days prior to needing replenishment.

4. RCS Pressure Control: In developing the NSCA, additions were made to the original Appendix R safe shutdown equipment list to ensure adequate pressure control during long-term hot standby conditions, including:
   - The ability to reduce pressure using auxiliary spray or a pressurizer PORV
   - The ability to increase pressure using one set of EDG-backed pressurizer heaters

   Neither of these functions has an inherent capacity limitation with respect to long-term considerations.

5. No stored capacity air systems are relied upon to support long-term hot standby. Where air systems are credited, availability of the associated air compressor has been confirmed.

Part b

Resources and actions (including time critical actions) required to achieve and maintain safe and stable conditions for the "At-Power Analysis" are identified in Attachment G of the LAR. The recovery actions contained in Attachment G have been evaluated for feasibility and reliability in accordance with FAQ 07-0030, as documented in FNP Calculation SE-C051326701-11, "Recovery Action Feasibility Evaluation."

With respect to long-term hot standby considerations, beyond those actions identified in LAR Attachment G there are no time-critical actions (actions required within 24 hours) needed to ensure the credited functionality, as described previously. Refer to Part c below for a detailed discussion of post-fire actions required to support long-term hot standby conditions. These actions are proceduralized and thus operators are familiar with the steps to accomplish the action. However, the procedures are not frequently exercised and are therefore described here in response to the RAI.

Part c

Depending on specific post-fire plant conditions, the following actions might be necessary to support long-term hot standby conditions:

1. Replenish CST inventory within 24 hours – CST inventory is monitored and corrective action for a low level is addressed by Procedure FNP-1/2-ARP-1.9. Although the NSCA includes the service water backup supply to the AFW pumps, this lineup is undesirable because it injects raw water into the steam generators. Depending on the specific conditions operators would first try to refill the CST via the water treatment plant (normal) or demineralized water supply (backup) in accordance with Procedure FNP-1/2-SOP-5.0. Additionally, Procedure NMP-EP-402 (B5b procedure) provides two additional methods for refilling the CST from the fire main or circulating water canal.

2. Replenish RWST inventory within approximately seven (7) days – RWST inventory is monitored and corrective action for a low level is addressed by Procedures FNP-1/2-ARP-1.3. If normal makeup functions are available the RWST will be refilled in accordance with Procedure FNP-1/2-SOP-2.3. Additionally, Procedure NMP-EP-402 (B5b procedure) provides two additional methods for refilling the RWST from the fire main or circulating water canal.

3. Replenish the EDG storage tank within approximately three (3) days – The EDG day tank is automatically replenished from its associated underground FOST. The automatic fuel oil transfer from the FOST to the associated EDG's day tank is included in the NSCA. After greater than (3) days of continuous operation, it may become necessary to replenish the FOST or provide makeup to the EDG day tank via another FOST. This action is accomplished in accordance with FNP-0-SOP-42.0. Replenishment of an FOST can be accomplished by transferring fuel oil from another FOST, from the auxiliary boiler fuel oil storage tank (AFOST), or directly from a tanker truck. The AFOST is replenished via tanker truck in accordance with Procedure FNP-0-SOP-55.

None of the long-term actions described above are time critical, and hence could be accomplished by on-shift personnel if necessary. However, it is likely that additional qualified support staff will be available to assist with these long-term, proceduralized actions.

Farley SSA RAI 13 - NPO Pinch Points

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 (MOV's) 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley SSA RAI 14 – This RAI response will be provided with supplemental correspondence.

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Regarding License Amendment Request for Transition to 10 CFR 50.48(c) – NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants

Enclosure 3
Response to Probabilistic Risk Assessment RAIs

Farley RAI PRA 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 01(a)

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 01(b) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 01(c) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 01(d) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 01(e) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 01(f) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 01(g) – This RAI response will be provided with supplemental correspondence.

FARLEY RAI PRA 01(h) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 04

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 05 – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 06(a) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 06(b) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 06(c) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 07(a) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 07 (b-a) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 07 (b-b) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 07 (b-c) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 07(b-d) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 08(a) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 08(b) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 09a

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.

RESPONSE:

The methodology for evaluating the ΔCDF and ΔLERF for the variance from the deterministic requirements (VFDRs) is described in Section 5.3 of Fire Risk Evaluation (FRE) Report (SE-C051326701-008, Version 1.0, September 25, 2012), and summarized below.

For each area, the variant case represents the post transition Fire PRA model. The variant case provides the point of reference for the FRE, from which the difference in risk between full compliance with NFPA 805 deterministic requirements is measured.

If the total area risk was less than the screening criteria, then the Fire Area CDF/LERF was typically considered as a surrogate for the delta CDF/LERF. For fire areas with higher total area risk, additional analysis was performed as described below.

For each area, targets that were failed in the Fire PRA model were reviewed, along with scenarios where the VFDR targets(s) were damaged. If any scenarios contained the VFDR targets, then next step was followed. If no scenarios for a given Fire Area resulted in damage to VFDR targets, then delta risks are zero and the variant case became the post-transition baseline case. The remaining analysis steps were not needed for this case.

For scenarios that were determined to damage the VFDR targets, the variant case was modified to reflect a deterministically compliant case for each scenario that impacted the listed VFDRs. This 'compliant' case provided the fire risk if the plant configuration was modified to reroute, or protect all of the associated VFDR components and cables. The quantification of this case was defined by setting the basic events in the Fire PRA model that are associated with the VFDRs to their nominal, no fire, random failure probability.

If ΔCDF and ΔLERF for a fire area were greater than the acceptance criteria, considerations for risk reduction were pursued such as model refinements, recovery actions, plant modifications or combinations thereof.

There were no methods used in the delta risk calculations that were not included in the Fire PRA Peer Review. The proposed plant modifications were included in the variant CDF/LERF and the compliant CDF/LERF calculation, representing a conservative estimation of the change in risk associated with transition to NFPA 805. Recovery actions were considered in the compliant case by setting the HEP to 0, representing the operator action being successful 100% of the time.

Farley RAI PRA 09b

b. Step 3 of Section 5.3, Performance of Fire Risk Evaluation, on page 11of 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 09c

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.

RESPONSE:

See response to Farley RAI PRA 09a for details on how the delta risks have been calculated.

The conclusion stated in section V.2.3 is cited against the relative increase in risk.

Farley RAI PRA 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 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.

RESPONSE:

There are no VFDRs involving performance-based evaluations of cables protected by Electrical Raceway Fire Barrier Systems (ERFBS) or embedded cables. The NSCA did not take credit for the cables being protected by ERFBS. The cable would have been assumed damaged in any fire area that the cable is in and if a VFDR was needed, it would have been generated. VFDRs were not generated due to inadequacy of the ERFBS. The cables that are embedded have been determined to be free of fire damage by an engineering evaluation. No VFDRs were generated to address embedded cable configurations.

There are plant modifications that are installing ERFBS for trays or conduits as discussed in Table S-2 of Attachment S. These modifications are credited for the variant and compliant cases in the Fire Risk Evaluations.

Promat H board cable protection in Stairwell No. 2 Fire Areas 1-S02 and 2-S02 in Unit 1 and Unit 2, respectively is credited in the variant and compliant cases in the Fire Risk Evaluations.

Farley RAI PRA 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).

RESPONSE:

The procedures and processes used to perform and maintain the analyses that support the NFPA 805 Licensing Bases will have post-transition change evaluation guidance that addresses key uncertainties, assumptions, sensitivity analyses and peer review Facts and Observations (F&Os). As discussed in the response to RAI-SSA-01, 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. Checklists in the NMP-ES-035 series of procedures for reviewing proposed changes for impact on the Fire Protection Program (FPP) include (or will include) questions regarding potential impacts to the various databases used in the analysis. These include fundamental FPP elements from Chapter 3 of NFPA 805, NSCA and NPO analyses, Radioactive Release analysis, and Fire PRA uncertainties, assumptions, sensitivity analyses and peer review Facts and Observations (F&Os) and supporting data. In this way, impacts to the various databases and analyses can be identified and updated as proposed changes to the plant are being developed. A new LAR Attachment S Table S-3 Implementation Item for revising the NMP-ES-035 series to ensure the above listed analyses are subject to these control processes is being added. This new Implementation Item was included in the September 16, 2013 SNC transmittal letter NL-13-1503, Attachment S, Table S-3 Update, Item 32, in response to Programmatic RAI 5.

The existing database, analysis, and document configuration control procedures/processes will be utilized. These are fleet level control processes that recognize the needs of controlled design inputs and verification. In LAR Section 4.7.3, it was stated, "During the transition to 10 CFR 50.48(c), FNP has and will continue to perform work in accordance with the quality requirements of Section 2.7.3 of NFPA 805." The PDMS, ARCPlus, and PRA databases are subject to the control procedures used to develop the NFPA 805 LAR. The software is subject to software QA requirements appropriate to the end use. The processes and procedures that will be used will ensure that integration database and software analyses comply with NFPA 805 modeling, content, and quality control requirements. Also, see response to RAI PROG-05 on Quality Requirements

Farley RAI PRA 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 15 - Deterministic seismic IPEEE basis question

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 16a

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.

RESPONSE:

The scenarios developed in the MCR and equipment rooms at Farley were defined using Appendix S of NUREG/CR-6850. The panel fires were limited to the source panel itself if there were no openings to other panels and if there was a double wall with an air gap. If there were openings or no barrier between panels then the adjoining panels would be considered failed in one scenario.

Assumption 3 will be revised to clearly state the assumption that was used in the analysis:

"For fires within the main control room and equipment rooms, the fire is expected to be limited within the panel, *if there is a double wall with an air gap to the adjacent panels with no other openings. This is* due to fire detection and suppression by operations personnel."

Farley RAI PRA 16(b) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 16(c) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 16(d) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 16(e) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 16(f) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 16(g) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 16(h) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 16(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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 17(a) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 17(b) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 17(c) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 17(d) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 18(a)

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:

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA (18b) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 18c Use of 5 Minute Cable Tray Ignition Time Rev A

For Calculation PRA-BC-F-11-015, "Joseph M. 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:

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.

RESPONSE:

The 702 kW electrical panel fires (multiple bundles of IEEE-383 qualified cable) were evaluated at Farley. In Attachment A of PRA-BC-F-11-015 "Joseph M. Nuclear Plant, Units 1 and 2, Hot Gas Layer and Multi-Compartment Analysis for Farley Fire PRA", beginning on page A-1 of the referenced calculation, entries in the column with the column descriptor "Heat Release Rate Qualified" equal to 702 indicate an evaluation of a 702 kW fire was performed. Attachment A ends on page A-31. Note that every entry in the far right column under the column descriptor "Cable Type" contains an entry equal to 383 indicating IEEE-383 qualified cable (see response to RAI FM-02a). As such, the 1,002 kW HRR fire assigned to represent multiple unqualified (Non-IEEE-383) cables in an open cabinet configuration fire was not evaluated. The text referenced on page 2-1 of referenced calculation will be revised to reflect this and to avoid further confusion.

The original approach for addressing secondary combustibles is described in Hughes Associates, "Supplemental Generic Fire Model Treatments: Hot Gas Layer Tables," Supplement 2, Revision G. Supplement 2 to the Generic Fire Modeling Treatments provides tables that document the time to reach threshold hot gas layer (HGL) temperatures for electrical panel ignition sources that ignite two adjacent 0.61 m (24 in) wide cable trays. The cable trays ignite at a plane section through the cable trays five minutes after the ignition source ignites and propagate fire laterally in two directions in a manner consistent with the NUREG/CR-6850, Appendix R guidance for thermoset cables. The configuration is considered to be an average representation of an ignition source – cable tray configuration and consequently may be over conservative in some situations and under-conservative in other situations when compared against the FLASH-CAT methodology provided in NUREG/CR-7010, Volume 1.

The treatment for ignition source – cable tray configurations has been updated in Reports 0005-0030-003-002, Rev. 0 and 0005-0030-0030-003, Rev. 0 to explicitly account for the heat release rate contribution from the cable trays on both the ZOI dimensions and the hot gas layer temperatures. The method used to calculate the vertical and lateral cable tray propagation in Reports 0005-0030-003-002, Rev. 0 and 0005-0030-0030-003, Rev. 0 follow the guidance provided in NUREG/CR-6850, Appendix R and NUREG/CR-7010, Volume 1 (FLASH-CAT calculation method) and is thus considered reasonable and conservative. In particular, the ignition time for the first cable tray in a stack is one minute after the ignition source ignition time. This time corresponds to the damage time listed in NUREG/CR-6850, Appendix H for thermoset cables applicable to temperature exposure conditions over 370°C (700°F) and heat flux exposure conditions greater than 20 kW/m^2 (1.75 Btu/s-ft^2) and is therefore bounding. The propagation time between the first tier cable tray and the second tier cable tray is four minutes as recommended in Appendix R of NUREG/CR-6850. The third tier, where present, ignites three minutes after the second tier, the fourth tier ignites two minutes after the third tier, and each additional tray beyond the fourth tier ignites one minute after the next lower cable tray in accordance with NUREG/CR-6850, Appendix R guidance.

A more detailed discussion of this treatment is provided in the responses to RAI FM-01h (Cable Tray ZOI) and FM-01i (Cable Tray Flame Propagation). In addition, Report PRA-BC-11-015 has been updated to reflect the heat release rate calculation method, which includes the initial cable tray ignition time as described in Reports 0005-0030-003-002, Rev. 0 and 0005-0030-0030-003, Rev. 0.

Farley RAI PRA 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, sec 1.0, "Purpose; page 2-1, Section 2.1, "Existing FPIE PRA Operator Actions;" and, page 2-23, Section 2.2, "New Post-Fire SSD 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 20(a) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 20b

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:

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.

RESPONSE:

The credit of a 15 minute detection time in the absence of automatic detection comes from Appendix P of NUREG/CR-6850 Section P.2. The 15 minute detection time is also consistent with Inspection Manual Chapter 0609, Appendix F, Attachment 8, Task 2.7.1 – Fire Detection Analysis. This attachment provides guidance for detection times associated with different types of detection, one of which is detection by general plant personnel. Page F8-3, bullet 2 under *Detection by General Plant Personnel* states, "In the absence of any other means of detection, a maximum fire detection time of 15 minutes will be used."

Farley RAI PRA 20(c) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 21(a)

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

FNP RAI PRA 21(b)

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:

b. 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 22(a)

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013

Farley RAI PRA 22(b)

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.

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 22(c)

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.

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 23(a)

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 23b

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:

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.

RESPONSE:

In the Farley fire PRA, fire-specific multiple human errors were identified in the same sequences or cutsets and their dependencies were evaluated to develop Joint Human Error probabilities as addressed in Section 4.0 and Attachment 4 of Calculation PRA-BC-F-11-016, Human Reliability Analysis for Fire Events, Version 1, September 13, 2012.

It should be noted that the dependency analysis was performed after the HFE identification process was complete such that it addressed all of the HFEs included in the FPRA model.

Farley RAI 23c SUT Cooling Modeling

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.

RESPONSE:

To clarify, the Farley Fire PRA credits self-cooling for the Start-up Transformers (SUTs), but does not model forced transformer cooling. Based on review and discussions among transformer design engineering, system engineering, operations, and the Farley Fire PRA team, it was concluded that forced cooling is not needed to meet PRA success criteria during the PRA mission time. These reviews and discussions included review of core documentation.

There are three basic ratings for the SUTs identified in U1864644, "Instruction Starting Station Service Transformer 1A & 1B" depending on the cooling mode of operation. These are shown below:

Cooling Mode	Transformer Rating
Self-cooling (oil-insulated, air-cooled)	29120 KVA
Forced-Cooling Stage 1 (Forced-Oil, Forced Air)	38752 KVA
Forced-Cooling Stage 1&2 (Forced-Oil, Forced Air)	48384 KVA

The following table shows the worst case plant loading as identified in SE-94-0470, "Unit 1 As-Built Load Study" compared to self-cooling mode rating of the transformer:

Transformer	X winding	Y winding	Total	Percent Load/ 65 deg. C Rated
1A	14706.6	15013.9	29720.5	102%
1B	17523.2	14243.2	31766.4	109%

As you can see in the self-cooling mode of operation, there is a possibility of overloading up to 9%, but that either forced-cooling mode would be more than adequate given that the forced-cooling value is much greater than the anticipated worse case of 9% overloading. While it is not expected that the self-cooling rating would be exceeded for very long even in the most extreme conditions (normally inconsequential loads would trip off), but if it does exceed these conditions it would be only at 9% which for the mission time of the Fire PRA (24 hours), is not expected to challenge the SUT given operating experience and practice as mentioned in IEEE C57.119 below. The failure probability is not increased because the self-cooling (oil-insulated air-cooled) mode relies on natural circulation and is not expected to fail. In order for the self-cooling mode to fail, some other initiating event would have to be introduced (i.e., some kind of puncture causing a leak). Since the SUT is not expected to be challenged in forced-cooling mode and the failure mode would rely on an outside initiating event, cooling of the SUT is not modeled and the probability of SUT failure is not increased.

The point of transition between a mechanical concern and thermal concern cannot be precisely determined, but mechanical effects tend to have a more prominent role in larger kilovolt ampere ratings, because the mechanical stresses are higher. For the range of discussion, it is expected that this results in long term degradation should there be any impact for such a short duration and minimal increase over nameplate rating, but not immediate failure.

Note that operating experience and practice has brought many insights about what transformers can actually withstand as illustrated below:

"Over the years, there has been a marked increase in the practice of loading transformers beyond their nameplate rating. In the past, many transformers were loaded beyond nameplate rating only during short time emergencies. Today, many users have established loading practices which subject transformers to loads beyond nameplate rating on a planned basis during periods of seasonal or daily peak loads, in addition to unexpected loads occurring during short or long time emergencies." Reference IEEE C57.119.

Farley RAI PRA 24(a)

(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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 24b

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:

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.

RESPONSE:

In the Farley fire PRA, a simplified location weighting factor (WF_L) of 2.0 was applied for the fixed and transient ignition sources. The justification of the simplified approach is discussed with two types of plant locations, shared locations and non-shared (or partially shared) locations.

Shared Plant Locations

Since Unit 1 and Unit 2 share the following plant locations, their WF_L should be "2" per NUREG/CR-6850. This means that the "simplified approach" applied to these locations is exactly the same as the approach recommended in NUREG/CR-6850.

- Main Control Room
- Diesel Generator Building
- Transformer Yard
- Plant-wide Components

Non-Shared or Partially-Shared Plant Locations

The other plant locations where the simplified approach was applied are Battery Room, Containment, Control/Auxiliary Building and Turbine Building discussed below.

- Battery Room: Batteries are the only applicable fixed ignition sources in the Battery Room location. Since the total number of Battery Rooms and the count of batteries in each Battery Room are the same in Unit 1 and Unit 2, the fire frequencies evaluated with the simplified approach are the same as the fire frequencies calculated with the WF_L of "1" per NUREG/CR-6850.
- Containment: Reactor Coolant Pumps (RCPs) are applicable fixed ignition sources in the Containment location. Since there is one Containment and three RCPs in each unit, the fixed ignition source fire

frequencies evaluated with the simplified approach are the same as the fire frequencies calculated with the WF_L of "1" per NUREG/CR-6850.

- Turbine Building: The applicable fixed ignition sources are boilers, Main Feedwater Pumps, Exciters, Turbine Generator Hydrogen and Oil. Similar to the Battery Room and Containment above, both approaches should generate the same fire frequencies for each bin of ignition sources since there are symmetries between Units 1 and 2.
- Transient Fires applicable to Containment, Control/Auxiliary Building and Turbine Building: The human activity influence factors for the transient fires were applied identically to the symmetric fire zones between units. However, since there are some asymmetric fire zones between units and cable influence factors were based on cable loading of each fire zone, the resultant fire frequencies per NUREG/CR-6850 would be slightly different from those estimated with the simplified approach of WF_L. The following comparisons explain the differences.

In the example of Fire Zone 0122 presented in Section 3.5 of Calculation PRA-BC-F-11-009, since the fixed ignition source, pump, is a Plant-Wide Component, the WF_L estimated per NUREG/CR-6850 is 2 which is the same as the simplified WF_L. Therefore, the fire frequency estimated with the simplified WF_L is the same as the fire frequency estimated with the approach addressed in NUREG/CR-6850.

For the transient ignition sources, both approaches, simplified and NUREG/CR-6850, are compared for Bins 6 and 7 and the following table shows the results of fire frequencies.

Fire Zone	Parameters	Bin 6		Bin 7	
		Simplified Approach	NUREG/CR-6850	Simplified Approach	NUREG/CR-6850
0122	Transient Source Weighting Factor	1.46E-03[1)]	2.79E-03[2)]	1.57E-03[1)]	3.06E-03[2)]
	Location Weighting Factor (WF_L)	2	1	2	1
	Bayesian Updated Ignition Frequency	2.43E-03		4.75E-03	
	Fire Frequency	7.10E-06	6.78E-06	1.49E-05	1.45E-05
2122	Transient Source Weighting Factor	1.46E-03[1)]	3.07E-03[3)]	1.57E-03[1)]	3.23E-03[3)]
	Location Weighting Factor (WF_L)	2	1	2	1
	Bayesian Updated Ignition Frequency	2.43E-03		4.75E-03	
	Fire Frequency	7.10E-06	7.46E-06	1.49E-05	1.53E-05

1) From Table 3.4-2 of Calculation PRA-BC-F-11-009

2) Based on human activity influence factors of Unit 1 and shared fire zones of Control/Auxiliary Building in Table 3.4-2 of Calculation PRA-BC-F-11-009

3) Based on human activity influence factors of Unit 2 and shared fire zones of Control/Auxiliary Building in Table 3.4-2 of Calculation PRA-BC-F-11-009

As shown in the table, Unit 1 fire frequencies estimated with the NUREG/CR-6850 location weighting factor are less than the current fire frequencies calculated with the simplified location weighting factor, while Unit 2 fire frequencies are larger than the current fire frequencies. The reasons for the differences between units are related to the difference of total counting of influence factors due to the asymmetric fire zones between Unit 1 and Unit 2, as discussed above.

In conclusion, only transient fire frequencies are impacted by the approaches of location weighting factors and it is expected that the impact to the fire risk is minor and will not skew the risk insights significantly.

Farley RAI PRA 25

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

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley PRA RAI 26(a)

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

c. SR IE-A7. Explain whether this is more than a documentation issue, since it appears the PRA model was changed.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 26(b)

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 26(c)

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

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.RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 26(d)

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

> 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 26e

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

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.

RESPONSE:

A dependency analysis accounting for the Farley fire-specific human actions was performed in the fire HRA as documented in Section 4.0 of Calculation PRA-BC-F-11-016, "Human Reliability Analysis for Fire Events", Version 1, September 13, 2012, and P0293100006-3621, "Human Reliability Analysis Attachment C – Dependency Analysis", Revision. 0, September 2010. The latter reference is an attachment to the internal events HRA notebook.

In general, the process used to perform the dependency analysis is consistent with the NUREG-1921 methodology; however, some clarifications and enhancements were made to the nodal definitions of the EPRI HRA Calculator's version of NUREG-1921 Figure 6-1. In the fire PRA, the floor value of 1.0E-05 was applied to any JHEP combinations with values less than 1.0E-05.

Farley RAI PRA 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 28(a)

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

> 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

FNP RAI PRA 28(b)

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

> 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 28c

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

> 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 28d

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

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.

RESPONSE:

The inclusion of the testing and inspection procedures in Section 8.1.1 of PRA-BC-F-11-014 is intended to meet CC-II for SR FSS-D7. These procedures show that the system is installed and maintained in accordance with applicable codes and standards. The implementation of the testing and maintenance procedures, in addition to the plant modification identified in Attachment S of the LAR, ensures that the system has been in a fully operable state during plant condition and there has also been no outlier behaviors identified.

Farley RAI PRA 28e

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

e. SR FSS-D8. Discuss whether fire propagation was assumed and, if not, whether the targets damaged in the fire may be underestimated.

RESPONSE:

The Fire PRA was first developed without credit for suppression or detection, and the target set for a given scenario was based on the ignition source type. Further in the analysis, existing detection and suppression systems were credited, for some cases, and these were based on plant modifications. For these cases where the credit was taken the target set was not changed based on the time to suppression or distance to target. Instead a conservative approach was taken to leave the original target set, which accounts for fire propagation, included in the Fire PRA along with the failure rate of the suppression system, therefore not requiring a review of damage time vs. suppression time. This is found in section 8.0 of PRA-BC-F-11-014.

There are additional walk downs being completed in response to other RAIs that address the addition of secondary combustibles and flame spread for a given scenario. These RAIs are PRA 17 and FMOD 01. In addition to these walk downs, a non-suppression probability may be applied to the scenarios that will take into account the distance to the nearest secondary combustible to better represent the time at which the secondary combustible target will actually be damaged.

Farley RAI PRA 28(f) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 28(g) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 28(hi) – This RAI response will be provided with supplemental correspondence.

Farley RAI PRA 28(j)

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

h. SR IGN-A7.

   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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 28(k)

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

h. SR IGN-A7.

   k. Discuss the validity of the bin 15 frequency for each physical analysis unit (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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 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).

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

FNP RAI PRA 32(a)

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

> 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.]

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 32(b)

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

For Fire Area U1 1-040, discuss why CDF equals LERF and delta-CDF equals delta-LERF.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 32(c)

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

For Fire Area U2 2-040, discuss why CDF equals LERF and delta-CDF equals delta-LERF.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 33a

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.

RESPONSE:

The Main Control Board panels are physically open to each other, as such there is no credit for any separation between the different panel sections. The fires postulated at these sources are propagated to adjoining cabinets regardless of panel designation. In the case of other non-Main Control Board panels located in the MCR, these were reviewed on a case by case basis to determine if these were open to each other or should be considered separated by a double wall with an air gap.

The response to Farley RAI PRA 16a also address the methodology used in the MCR fire scenario development.

Farley RAI PRA 33(b)

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:

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PRA 33(c) – This RAI response will be provided with supplemental correspondence.

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Regarding License Amendment Request for Transition to 10 CFR 50.48(c) – NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants

Enclosure 4
Response to Monitoring Program RAI

Farley RAI MPV 01 Monitoring Program

Section 4.6 of the LAR Transition Report (Enclosure 1 to letter dated September 25, 2012, ADAMS Accession No. ML 12279A235) 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Regarding License Amendment Request for Transition to 10 CFR 50.48(c) – NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants

Enclosure 5
Response to Programmatic RAIs

Farley RAI PROG 01 Program Documentation

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PROG 02 FP Program Changes

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PROG 03 FP Program Training

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).

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PROG – 04 Combustible Control Program

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

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI PROG 05 Quality Requirements

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Regarding License Amendment Request for Transition to 10 CFR 50.48(c) – NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants

Enclosure 6
Response to Fire Modeling RAIs

Farley RAI FMOD-01a

Section 2.4.3.3, "FREs," of NFPA-805, states: "[t]he PSA 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 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.

RESPONSE:

The assumption of a fifteen minute fire brigade response time is based on a reasonable response time for the control room. There is no specific data of fire brigade drill time response to the control room available ; however, a review of the fire brigade drill times for drills conducted between November 22, 2011 and September 5, 2013 for the fire brigade arrival to various plant areas is summarized in Table FMOD-01a-1. The plant areas listed in Table FMOD-01a-1 include spaces near the control room as well as outlier areas such as the Service Water Intake Structure (SWIS) and provide an indication of the likely control room fire brigade response. The times shown in Table FMOD-01a-1 represent the time interval between the fire brigade page and the arrival of the last required fire brigade member. The average response time is 14.2 minutes with a minimum time of eleven minutes and a maximum time of nineteen minutes. The key aspect of the response with respect to the control room abandonment calculation is the potential for the ventilation conditions to change via an open door; as such, the times shown in Table FMOD-01a-1 are conservative insofar as they are based on the arrival of the last team member.

Farley RAI FMOD 01(b)

Section 2.4.3.3, "FREs," of NFPA-805, states: "[t]he PSA 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 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:

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 01(c)

Section 2.4.3.3, "FREs," of NFPA-805, states: "[t]he PSA 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 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:

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 01(d)

Section 2.4.3.3, "FREs," of NFPA-805, states: "[t]he PSA 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 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:

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 01(e)

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 01(f)

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:

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 01(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 at FNP.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 01(h) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 01(i) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 01 (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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 01(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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 01(l) – This RAI response will be provided with supplemental correspondence.

Farley FMOD RAI 01(m) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 01(n) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 01(o) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 01(p) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 02(a)

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 thermo set and thermoplastic cables described in NUREG/CR-6850.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 02(b) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 02(c)

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:

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

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 02(d) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 02(e)

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:

(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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 02(f) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 02(g) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 03(a)

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 03(b)

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:

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 04

NFPA 805, Section 2.7.3.3, and "Limitations of Use," 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 Transition Report states that "Engineering methods and numerical models used in support of compliance with 10 CFR 50.48(c) was applied appropriately as required by Section 2.7.3.3 of NFPA 805."

Regarding the limitations of use:

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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 05(a)

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 FNP staff and consulting engineers to use and apply the methods and fire modeling tools included in the engineering analyses and numerical models.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 05b

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

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FM 05c

Provide the position and qualifications of the personnel who performed the walkdowns for the Main Control Room (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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 05(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.

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 05(e)

Explain the communication process between the consulting engineers and FNP and SNC 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI FMOD 06(a) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 06(b) – This RAI response will be provided with supplemental correspondence.

Farley RAI FMOD 06(c) – This RAI response will be provided with supplemental correspondence.

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Enclosure 7
Response to Radiation Release RAIs

Farley RAI Radioactive Release 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.

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

Farley RAI Radioactive Release 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).

RESPONSE:

Response provided by SNC letter NL-13-1503, dated September 16, 2013.

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NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants

Attachment 1
Revisions to the Transition Report Main Body – Section 4.2.1.2

4.2.1.2 Safe and Stable Conditions for the Plant

Overview of Process

The nuclear safety goals, objectives and performance criteria of NFPA 805 allow more flexibility than the previous deterministic programs based on 10 CFR 50 Appendix R and NUREG 0800, Section 9.5-1 (and NEI 00-01, Chapter 3) since NFPA 805 only requires the licensee to maintain the fuel in a safe and stable condition rather than achieve and maintain cold shutdown.

NFPA 805, Section 1.6.56, defines Safe and Stable Conditions as follows

> *"For fuel in the reactor vessel, head on and tensioned, safe and stable conditions are defined as the ability to maintain Keff < 0.99, with a reactor coolant temperature at or below the requirements for hot shutdown for a boiling water reactor and hot standby for a pressurized water reactor. For all other configurations, safe and stable conditions are defined as maintaining Keff < 0.99 and fuel coolant temperature below boiling."*

The nuclear safety goal of NFPA 805 requires "*...reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition*" without a specific reference to a mission time or event coping duration.

For the plant to be in a safe and stable condition, it may not be necessary to perform a transition to cold shutdown as currently required under 10 CFR 50, Appendix R. Therefore, the unit may remain at or below the temperature defined by a hot standby/hot shutdown plant operating state for the event.

Results

Based on FNP Calculation SE-C051326701-003, "FNP NSEL and Safe and Stable Fault Trees," the NFPA 805 licensing basis for FNP is to establish and maintain safe and stable conditions, which is defined as the ability to maintain $K_{eff} < 0.99$ with a reactor coolant temperature at or below the requirements for hot standby.

Demonstration of the Nuclear Safety Performance Criteria for safe and stable conditions was performed in two analyses:

RAI SSA-12

- At-Power analysis, Modes 1-2. This analysis is discussed in Section 4.2.4.
- Non-Power analysis, which includes Modes 3 and below. This analysis is discussed in Section 4.3.

Following the reactor trip, the plant will be placed in a known safe and stable condition.

With the plant safe and stable in hot standby (Mode 3), decay heat removal is accomplished via one or more steam generator under natural circulation. The auxiliary feedwater (AFW) system supplies water to one or more steam generator being used to remove decay heat by venting steam to atmosphere. Reactor coolant system (RCS) inventory is maintained by one or more charging pumps aligned to the refueling water storage tank (RWST). Charging continues as needed to account for RCS shrinkage due to cooldown and expected losses. The RWST utilizes borated water to maintain

reactivity shutdown margins. Pressure is controlled via operation of the pressurizer auxiliary spray valve, one or more of the pressurizer Power Operated Relief Valves (PORVs), and one of two sets of emergency diesel generator backed pressurizer heater banks.

When plant temperature is less than approximately 350°F and pressure is less than approximately 425 psig, the Residual Heat Removal (RHR) system can be placed in service. However, the At-Power analysis confirms the ability to remain in hot standby conditions for an extended.

The ability to maintain safe and stable conditions for an extended period of time while in hot standby might require replenishment of diesel fuel oil, borated injection water for RCS makeup, and/or condensate for steam generator makeup. These actions can be accomplished by either on-shift operators or the Site's Emergency Response Organization, which would be manned for a significant fire event. See Implementation Item in Table S-3 of Attachment S.

~~Based on FNP calculation entitled "FNP NSEL and Safe and Stable Fault Trees", the NFPA 805 licensing basis for Safe and Stable as used in this analysis assumes the plant to be taken subcritical and maintained in any one of the modes of hot standby, hot shutdown, cold shutdown, or refueling conditions as defined in the FNP Technical Specifications.~~

~~Demonstration of the Nuclear Safety Performance Criteria for safe and stable conditions was performed based upon the plant being at 100% full power operation at the onset of the fire as a bounding condition. The At-Power analysis, Modes 1-2, discusses this in Section 4.2.4. The Non-Power analysis, which includes Modes 3 and below is discussed in Section 4.3.~~ RAI SSA-12

~~Following the reactor trip, the plant will be placed in a known safe and stable condition.~~

~~With the plant safe and stable in hot standby, a natural circulation cooldown resulting from a loss of offsite power will be initiated to transition to the next safe and stable mode of hot shutdown. Assuming complete loss of offsite power is regarded as the most conservative method and limits the scope of analysis to those SSCs powered from the diesel generators and batteries. At this point in time, the analysis for safe and stable in non-power operational modes also begins and is enveloped by the cooldown from at power. Emergency feedwater operation continues and steam is released from one or more steam generators to remove decay heat. Charging continues to account for RCS shrinkage and expected losses and utilizes borated water to maintain reactivity shutdown margins. Pressure is reduced via operation of the pressurizer auxiliary spray valve or the pressurizer Power Operated Relief Valves (PORVs).~~

~~When plant temperature is less than approximately 350°F and pressure is less than approximately 425 psig, the Residual Heat Removal (RHR) system will be placed in service. This utilizes a RHR pump to circulate RCS water to the RHR heat exchanger, where the decay heat is transferred to the component cooling water system. The plant will continue to cool down and the component cooling water system transfers the heat to~~

~~the service water system. In this manner, plant temperature will be reduced below 200°F and cold shutdown.~~

~~Depending upon the location and extent of the fire, the unit may be maintained in any one of the safe and stable modes described above for an extended period of time until the readiness of the systems necessary for the next safe and stable mode on the cooldown timeline can be verified to be operational. The ability to maintain safe and stable conditions at a particular mode for extended periods (generally regarded as greater than 24 hours) may require additional actions such as replenishment of diesel fuel oil, replenishment of borated injection water, or replenishment of condensate supplies and can be performed by the Site's Emergency Response Organization. See Implementation Item in Table S-3 of Attachment S.~~

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Attachment L
NFPA 805 Chapter 3
Requirements for Approval

Approval Request 1

Replace Approval Request 1 in its entirety with the Approval Request working provided in the response to RAI SSA 09

~~**NFPA 805 Section 3.3.5.1**~~

~~NFPA 805 Section 3.3.5.1 states:~~

> ~~*"Wiring above suspended ceiling shall be kept to a minimum. Where installed, electrical wiring shall be listed for plenum use, routed in armored cable, routed in metallic conduit, or routed in cable trays with solid metal top and bottom covers."*~~

~~Wiring above suspended ceilings at FNP has been kept to a minimum, but that wiring may not comply with the other requirements of this code section.~~

~~The areas at FNP currently with suspended ceilings inside the NFPA 805 defined power block include:~~

- ~~Auxiliary Building corridors and office areas~~
- ~~Control Room/associated offices and the Technical Support Center~~
- ~~Computer Room, Unit 1 and 2~~

~~Current FNP cable design and procurement requirements specify IEEE 383 cables for power and control cables. Original power and control cables may not have been IEEE-383 or equivalent. A review of design drawings confirmed that there are power/control cables above the ceiling in the Auxiliary Building corridors and office areas, the Control Room/associated offices and the Technical Support Center (TSC), and the Computer Rooms. The Computer Room area is not risk significant.~~

~~The Auxiliary Building corridors and office areas have pre-action systems installed above and below the suspended ceilings in areas where power cables are installed in trays without solid metal covering. Areas that are not protected with a pre-action system above and below the ceiling contain only low voltage control and instrumentation cables, or power cables routed in metallic conduit. Thus, the potential for a cable short developing and providing an ignition source is low.~~

~~The Control Room/associated offices and the TSC have power and control cabling installed above the suspended ceilings. The Control Room area is a continually staffed area, and the potential of an incipient fire developing is low. Only low power level circuits and circuits required to support Control Room systems are routed in the Control Room. Heavy power circuits are routed entirely in rigid conduit. This design, in conjunction with the absence of other significant sources of ignition, further reduces the potential of a Control Room fire. The areas outside the Control Room, such as the kitchen as well as surrounding rooms and the TSC, contain minimal power cables above the suspended ceiling, but due to the proximity to the occupied Control Room spaces, it is unlikely that a fire could develop undetected.~~

~~The Computer Rooms have total flooding Halon systems installed in the areas with suspended ceilings. The majority of cables installed in these areas are control and instrumentation cables, with a few 120V power cables installed. The presence of the Halon system ensures that the area is protected in the unlikely event that a fire develops.~~

NFPA 805 FAQ 06-0022 identified acceptable electrical cable construction tests. ~~Plenum rated cable is tested to NFPA 262. The FAQ concluded that the NFPA 262 test is equivalent to the IEEE 383 test. Therefore, IEEE cable is inherently equivalent to plenum rated cable and acceptable to be routed above suspended ceilings. Current FNP cable design and procurement requirements specify IEEE 383 cables for power and control cables.~~

~~**Basis for Request:**~~

~~The basis for the approval request of this deviation is:~~

- ~~The wiring above ceilings in the spaces discussed above do not pose a hazard:~~
  - ~~Low voltage wiring is not susceptible to shorts causing a fire.~~
  - ~~Power and control cables procured to IEEE 383 requirements are protected~~
  - ~~(plenum rated equivalent) to NFPA 262 requirements.~~
  - ~~By eliminating cables with potential shorts, this eliminates ignition sources and therefore the jacketing of cable is not relevant.~~
  - ~~The Main Control Room area is continually staffed, and the potential of an incipient fire reaching the smoldering stage is very low.~~
  - ~~Areas with pre-action systems above and below the suspended ceilings have additional protection against the threat of a fire.~~
  - ~~Areas containing Halon suppression systems have additional protection against the threat of a fire.~~

~~**Acceptance Criteria Evaluation:**~~

~~**Nuclear Safety and Radiological Release Performance Criteria:**~~

~~The location of wiring above suspended ceilings does not affect nuclear safety. Power and control cables comply, or comply with the intent of this section. Other wiring, while it may not be in armored cable, in metallic conduit, or plenum rated, is low voltage cable not susceptible to shorts that would result in a fire. Therefore there is no impact on the nuclear safety performance criteria.~~

~~The location of cables above suspended ceilings has no impact on the radiological release performance criteria. The radiological review was performed based on the potential location of radiological concerns and is not dependent on the type of cables or locations of suspended ceilings. The cables do not change the results of the radiological release evaluation performed that concluded that potentially contaminated water is contained and smoke is monitored. The cables do not add additional radiological materials to the area or challenge systems boundaries.~~

~~**Safety Margin and Defense-in-Depth:**~~
~~Power and control cables meet the requirements or the intent of this requirement. Other wiring, while it may not be in armored cable, in metallic conduit, or plenum rated, is low voltage cable not susceptible to shorts that would result in a fire. These areas and the cables have been analyzed in their current configuration. Therefore, the inherent safety margin and conservatisms in these analysis remain unchanged.~~

~~The three echelons of defense in depth are 1) to prevent fires from starting (combustible/hot work controls), 2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre fire plans), and 3) provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The prior introduction of non-listed cables routed above suspended ceilings does not impact fire protection defense in depth. Echelon 1 is maintained by the cable installation procedures documenting the requirements of NFPA 805 Section 3.3.5.1. The introduction of cables above suspended ceilings does not affect echelons 2 and 3. The cables routed above suspended ceilings does not directly result in compromising automatic fire suppression functions, manual fire suppression functions, or post fire safe shutdown capability.~~

~~Conclusion:~~

~~NRC approval is requested for the use of non-listed cables routed above suspended ceilings.~~

~~The engineering analysis performed determined that the performance based approach utilized to evaluate a variance from the requirements of NFPA 805 Chapter 3:~~

- ~~(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;~~
- ~~(B) Maintains safety margins; and~~
- ~~(C) Maintains fire protection defense in depth (fire prevention, fire detection, fire suppression, mitigation, and post fire safe shutdown capability).~~