Attachments C and S contain security-related information and should be withheld under 10 CFR 2.390. Upon removal of Attachments C and S, this correspondence is de-controlled.

Southern Nuclear

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Edwin I. Hatch Nuclear Plant Units – 1 and 2
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 April 4, 2018, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) for Edwin I. Hatch Units 1 and 2 (Agencywide Documents Access and Management System Package Accession No. ML18096A936; EPID L-2018-LLA-0107). 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 email dated March 29, 2019, the NRC Staff formally transmitted a request for additional information (RAI) related to the referenced license amendment. The enclosure provides currently available responses to the NRC RAIs. Please note that the SNC response does not address PRA RAIs 03 and 18. SNC would like to request a clarification call with the NRC to discuss the response to the PRA RAIs applicable to PRA RAI 03 and 18 at the NRC's earliest convenience. SNC will respond to PRA RAIs 03 and 18 within 90 days of this clarification call. In addition, SNC identified the need to provide a supplement to the original LAR. This supplement will also be provided within 90 days of the clarification call. The attachments 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.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

A006
NRR

I declare under penalty of perjury that the forgoing is true and correct. Executed on the 28th day of May 2019.

Respectfully submitted,

Michael D. Meier
Vice President of Regulatory Affairs

MDM/RMJ

Enclosures:
1. Response to Fire Modeling (FM) RAIs
2. Response to Safe Shutdown Analysis (SSD) RAIs
3. Response to Health Physics (HP) RAIs
4. Response to Probabilistic Risk Assessment (PRA) RAIs
5. Response to Fire Protection Engineering (FPE) RAIs

Attachments:
1. Attachment to NFPA 805 PRA RAI 08
A. Revisions to Transition Report Attachment A
C. Revisions to Transition Report Attachment C (Security-related information)
J. Revisions to Transition Report Attachment J
S. Revisions to Transition Report Attachment S (Security-related information)

Cc: Regional Administrator, Region II
NRR Project Manager – Hatch
Senior Resident Inspector – Hatch
RTYPE: CHA02.004

Edwin I. Hatch Nuclear Plant Units – 1 and 2
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 1
Response to Fire Modeling (FM) RAIs

NFPA 805 Fire Modeling (FM) Request for Additional Information (RAI) 01

Section 2.4.3.3, "Fire Risk Evaluations," of National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition (NFPA 805) states: "[t]he PSA [probabilistic safety assessment] approach, methods, and data shall be acceptable to the AHJ [authority having jurisdiction]... "

Based on information provided by the licensee, the Nuclear Regulatory Commission (NRC) staff determined that FM is comprised of the following:

- A plant-specific Fire Modeling Workbook (FMWB) that was developed in lieu of using NUREG-1805, "Fire Dynamics Tools (FDT^{S}) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program" (ADAMS Accession No. ML043290075), or Electric Power Research Institute (EPRI) Fire Induced Vulnerability Evaluation Methodology, Revision 1 (FIVE Rev. 1), to determine the Zone of Influence (ZOI) for ignition sources and the time to Hot Gas Layer (HGL) conditions in all fire areas throughout the plant.

- Heskestad's plume temperature correlation which was used to determine the vertical separation distance based on temperature to a target in order to determine the vertical extent of the ZOI.

- The Consolidated Fire Growth and Smoke Transport (CFAST) model which was used to assess main control room (MCR) abandonment time and multi-compartment analysis (MCA).

- FLASH-CAT which was used for calculating fire propagation in stacks of horizontal cable trays.

- Heat soak method which was used for evaluating time to cable damage.

License Amendment Request (LAR) Section 4.5.1.2, "Fire PRA," states that FM was performed as part of the Fire PRA (FPRA) development (NFPA 805-Section 4.2.4.2) and reference is made to LAR Attachment J, "Fire Modeling Verification and Validation," for a discussion of the acceptability of fire models that were used to develop the FPRA. Based on the information in the LAR, the NRC staff was unable to fully evaluate the FM performed as part of the FPRA and requests that the licensee:

(a) Regarding fires in the proximity of a corner or walls, explain how the FM approach was applied. Explain how wall and corner affects the ZOI and HGL timing calculations were accounted for, or provide technical justification if these effects were not considered. Explain how transient fires against a wall or in a corner were considered in the MCR abandonment calculations.

(b) The NRC staff finds that typically, during maintenance or measurement activities in the plant, electrical cabinet doors remain open for a certain period of time. Describe whether there are any administrative controls in place to minimize the likelihood of fires involving such a cabinet, and describe how cabinets with temporarily open doors are treated.

(c) Describe and provide technical justification for the approach that was used in the FLASH-CAT model to determine the time to ignition, the heat release rate per unit area (HRRPUA), and the flame spread rate for cable trays that contain a mixture of thermoplastic and thermoset cables.

(d) Describe the "Heat Soak Method" that was used to convert the damage times in Appendix H of NUREG/CR-6850 to a percent of damage function for targets exposed to a time-varying heat flux.

(e) Describe how non-cable secondary combustibles were identified and accounted for.

(f) Explain how the model assumptions in terms of location and HRR of transient combustibles in a fire area or zone will not be violated during and post-transition.

(g) Describe how high energy arcing fault (HEAF) initiated fires are treated in the HGL development timing. Regarding HEAF generated fires, describe the criteria used to decide whether a cable tray in the vicinity of an electrical cabinet will ignite following a HEAF event in the cabinet. Explain how the ignited area was determined and subsequent fire propagation calculated. Describe the effect of cable tray covers and fire-resistant wraps on HEAF induced cable tray ignition and subsequent fire propagation.

(h) Facts and Observations (F&O) 20-19, Structural Steel Scenario Selection," was generated and dispositioned with the following:

The F&O identifies a statement in report H-RIEFIREPRA- U00-008D, "Specific ignition sources proximate to or with direct impingement on exposed structural steel were not evaluated." The purpose of this statement is to distinguish that detailed analytical heat transfer and structural analysis of steel members was not performed.

As stated earlier in the report, the following criteria are used to develop structural steel scenarios, which makes the statement identified in the F&O unnecessary:

a) Exposed structural steel is present and,
b) A high-hazard fire source is present.

Following these criteria, the scenarios developed are inherently more conservative than an analysis that relies on detailed fire modeling and analytical heat transfer modeling of individual structural steel members.

The F&O has been resolved by deleting the statement in question.

Describe and provide technical justification for the approach that make structural steel scenarios more conservative.

(i) Regarding the Main Control Abandonment (MCA), describe how the size of the opening between the exposing and exposed compartments assumed in the CFAST HGL calculations was determined, and explain to what extent these vent sizes are representative of conditions in the plant.

(j) Regarding the acceptability of CFAST for the control room abandonment time study, describe whether the volumes of the main control boards (MCBs), electrical panels, raised platforms, ductwork in the interstitial space above the egg-crate ceiling, and other obstructions are excluded from the effective control room volume used in the CFAST calculations.

(k) Because the Main Control Room (MCR) abandonment calculations are based on the assumption that all doors would normally remain closed, describe if any natural leakage vents were assumed in the analysis.

SNC Response

(a) For ignition sources located within two feet of a wall or corner, the heat release rate was multiplied by a modification factor of two for fires located near a wall and a factor of four for fires located near a corner. The modification factor was applied in the Fire Modeling Workbook, which was used to perform the Hatch Fire PRA detailed fire modeling calculations. The approach is consistent with the discussion presented in Appendix F of Attachment 3 to the NRC Inspection Manual.

The ZOI calculations and HGL timing calculations for fires located close to a wall or corner were accounted for in the Fire Modeling Workbook by increasing the heat release rates. An increase in heat release rate causes an increase in ZOI in both the horizontal and vertical directions. If the expanded ZOI causes the ignition of additional, secondary combustibles, then the contribution of these burning secondary combustibles was incorporated into the heat release rate curves produced using the FLASH-CAT model and incorporated into the calculation of the hot gas layer temperature versus time.

In the Main Control Room abandonment calculations, transient fuel packages in the open, wall, and corner locations were evaluated using CFAST. The CFAST model reflects the condition wherein fires located near a wall boundary or near a corner would experience a reduced air entrainment and a force imbalance on the plume.

(b) The fire modeling analysis assumes that electrical cabinets with normally closed doors are maintained closed. This assumption is based on existing plant procedures and administrative controls that require compliance with applicable industrial/electrical safety requirements, prevention of foreign material from entering the maintenance area, and proper housekeeping practices related to fire prevention and protection of the equipment. For the brief period's cabinet doors may be temporarily open, these administrative procedures provide reasonable assurance that the likelihood of a fire in these cabinets is minimal.

Additionally, electrical cabinet doors are unlikely to be left open when maintenance or measurement activities are not in progress. To ensure this assumption remains valid, the condition of cabinet doors will be included in the monitoring program. LAR Attachment S, Table S-3, describes the Implementation Items that will be completed prior to the implementation of the new NFPA 805 Fire Protection Program. The following implementation item has been added to Table S-3.

Table S-3 Implementation Item IMP-22: Verification of the condition of electrical cabinet doors to meet fire modeling assumptions will be included in the monitoring program.

(c) The method for evaluating secondary combustible cable tray heat release rate profiles used the FLASH-CAT model provided in NUREG-7010. The Hatch Fire PRA calculated the time to ignition of the secondary combustibles in one of the following two ways: (1) the time to ignition was equal to the time at which the flame height reached the first unprotected cable tray, or (2) a conservative time of 1 minute was used, based on the minimum ignition time for cables as described in Appendix H of NUREG/CR-6850.

For all Unit 1 Fire Zones and fire zones that are shared between Unit 1 and Unit 2 (i.e., common fire areas/zones), the FLASH-CAT model used the heat release rate per unit area and horizontal flame spread rates for thermoplastic cables. For Unit 2 Fire Zones in which FLASH-CAT was used, the model used the heat release rate per unit area and horizontal flame spread rates for thermoset cables. Per RBA-16-002-H, 97.39% of the cable insulation in Unit 2 could be classified as thermoset. Employing a weighted average approach, as recommended in NUREG-7010, the heat release rate per unit area would be 152 kW/m2 (an increase of 2 kW/m2) and the flame spread rate would be 1.15 m/hour (an increase of 0.05 m/hr.) Results from sensitivity cases using the weighted average values for the FLASH-CAT analysis showed negligible changes in the results. Therefore, the values for thermoset are appropriate for Unit 2.

(d) Technical Procedure TECH-FRA-02 describes the methodology employed in the Hatch Fire PRA for evaluating the time-to-damage for generic cables exposed to a time-dependent temperature of heat flux. The damage integral heat soak method was employed in the same manner as documented in Prairie Island Response to NFPA 805 LAR, PRA RAI 21.01, Available via NRC ADAMS: https://www.nrc.gov/docs/ML1706/ML17065A339.pdf

(e) The Hatch Fire PRA identifies areas where significant foam insulation was observed (via walkdowns) and documents the analysis of those scenarios in which the ignition of these non-cable secondary combustibles could result in an increase in the scenario heat release rate and target impacts of that scenario. Specifically, if the foam insulation was ignited and resulted in an increased heat release rate that was greater than 15% of the primary ignition source, the heat release rate profile for the scenario was revised to include the contribution of the burning foam. Due to uncertainty in the heat release rate values for the foam material, combustible foam sources with estimated heat release rates less than 15% of the primary ignition source were screened from this analysis. With this larger heat release rate value, the revised (i.e., larger) zone of influence for the ignition source scenario was determined. If these additional targets resulted in an increase in the fire risk for that scenario, this increase was noted and incorporated into the model results.

Currently no above ground high density polyethylene (HDPE) piping is present at plant Hatch. In the future, if any plant change involves installation of HDPE material or other non-cable secondary combustibles, as part of the 805 process, the change will be reviewed and the impact on the Fire PRA will be captured and addressed accordingly.

(f) All of the open floor areas in the fire zone were included in the transient zone areas. This ensures that all possible locations of transient combustibles were considered in the Hatch Fire PRA. Further, the size of a defined transient zone was larger than the size of a "zone of influence" as calculated using fire modeling tools (i.e., as described in Appendix F of NUREG/CR-6850). This approach accounted for uncertainties in the specific location of a transient source within a transient zone and ensured that these specific model assumptions would not be violated during or post-transition.

Walkdowns were performed to validate the appropriateness of the heat release rate chosen for modeling transient fires in that fire zone. In most cases, the 98th percentile peak heat release rate value of 317 kW was used. For those fire zones with posted combustible controls, a reduced peak heat release rate of 65 kW was used. Control of combustible materials is governed in accordance with Procedure NMP-ES-035-014, Fleet Transient Combustible Controls. This procedure requires immediate removal of waste, debris, scrap, packing materials, and other combustibles from an area immediately following the completion of work or at the end of the shift, whichever comes first (NFPA 805, Ch.3, Section 3.3.1.2(3). Further, combustible storage and staging areas are designated, and limits are established on the types and quantities of storage materials (NFPA 805, Ch. 3, Section 3.3.1.2(5). Additionally, general housekeeping is maintained in accordance with Procedure NMP-MA-054 General Plant Housekeeping.

Applicable procedures are as follows:

- Procedure NMP-ES-035-014, Fleet Transient Combustible Controls,
- Procedure NMP-MA-054, General Plant Housekeeping, and
- Procedure NMP-ES-035-003, Fleet Hot Work Procedure

(g) HEAF scenarios were analyzed following the empirical approach described in Appendix M of NUREG/CR-6850. HEAFs are events with two distinct phases: an initial energetic phase (the arcing explosion), and an ensuing fire phase. The heat release rates in high energy arcing fault scenarios were determined as follows:

1. A peak heat release rate value was assigned to the switchgear or load center where the high energy arcing fault was postulated following the guidance described in Section 4.3 of Report H-RIE-FIREPRA-U00-008A. It was assumed that the peak heat release rate was achieved at the time of the explosion (i.e., no fire growth phase was credited).

2. The combustibles within the zone of influence of the fault were determined and heat release rates were assigned to those combustibles. No growth profile was credited to combustibles in the zone of influence.

3. Heat release rates of exposed cable secondary combustibles were assigned using the FLASH-CAT model.

Tray covers and fire-resistant wraps that were within the ZOI of the HEAF were assumed to be damaged during initial fault event and were not credited to reduce area of the cable trays ignited by the fire. Therefore, the heat release rate curve incorporated into the CFAST model included (1) the peak heat release rate value of the switchgear or load center, (2) the immediate involvement of trays within the HEAF ZOI, and (3) flame spread through the exposed trays modeled in accordance with FLASH-CAT.

(h) Consistent with the ASME/ANS Fire PRA Standard, the risk-relevant scenarios in the structural steel analysis were based on the following conditions only: 1) exposed structural steel and 2) a high-hazard fire source was present. Since no detailed structural analysis was performed, this screening analysis assumed that the failure of only one exposed structural steel was sufficient to justify further evaluation of the scenario. A high-hazard fire refers to a large and long-duration fire consistent with the clarification note to Table 4-2.6-7 of the ASME/ANS Fire PRA Standard [Reference: Part 4 of the ASME/ANS Combined PRA Standard, RA-Sa-2009]. Examples of these fires include 1) ignited oil running down to lower elevations of the Turbine Building or 2) fires involving ignition of the entire content of oil reservoirs. Evaluation of the scenarios included an assessment of target damage that ranged from damage to targets outside the PAU where the fire originates to collapse of the building in question.

Distance to exposed structural steel (which could be used in a determination of flame impingement or in a heat transfer analysis) was not refined. Rather, the screening analysis assumed that any high-hazard fire was capable of damaging exposed structural steel if both were found in the same physical analysis unit.

Consistent with the conservative nature of this study, the quantified structural steel scenarios did not credit manual suppression actions that might have been taken during the fire and assumed failure of the structural steel immediately upon failure of any available automatic suppression system. The time required to heat steel up to its structural failure temperature was not considered.

Therefore, the approach employed in the Structural Steel Scenario Selection Analysis Report should be considered bounding because it results in the inclusion of all fire scenarios for which a structural collapse is conceivable, without requiring a detailed structural and/or heat transfer analysis.

(i) In the Hatch Fire PRA Multi-Compartment Analysis, the horizontal vent between the exposing and exposed fire compartments in the CFAST model was a standard double-doorway (Width = 1.83 m, Height = 2.12 m). The double doorway was centered on the wall shared by the two compartments and was entirely open. Fire effects flowed freely to the exposed compartment once the depth of the hot gas layer exceeded the distance between the ceiling of the exposing compartment and the height of the doorway. The open doorway was conservative for this multi-compartment application as follows:

1. The open double-doorway provided a less restrictive flow path than the small horizontal vents between the compartments and the "outside", which were added in the CFAST model to account for typical leakage in compartment construction. This ensured that the majority of the fire effects would migrate through the open door rather than to the outside via the leakage vents.

2. The open double-doorway was a bounding configuration compared to typical plant conditions, because it allowed fire effects to migrate from one compartment to another. Other more typical openings between compartments are usually provided with protective features, such as dampers, closed fire doors, or penetrations seals, and would not allow fire effects to pass between compartments.

(j) The Main Control Room abandonment report documents how volume obstructions were subtracted from the gross area volume for each space included in the CFAST. The volume obstructions include electrical enclosures and structural components (e.g. columns). In addition, the volume of the nonstructural obstructions (e.g., ductwork, smaller utilities, and conduits) in the interstitial space was conservatively assumed to be ten percent of the gross volume of the interstitial space, which was based on field observations and a review of the major ductwork in the areas. The volume of the contents in each restroom (e.g. storage cabinets) was conservatively assumed to be five percent of the gross restroom volume. The main control boards are not enclosed (i.e., have open backs), and therefore were not included as a volume obstruction.

(k) The MCR abandonment calculations assumed that doors between the control room, surrounding spaces, between the electrical equipment rooms, and surrounding spaces remained closed. However, exfiltration and boundary leakage occurred via small gaps and openings in the enclosure boundary. The effect of this configuration was to maximize the severity of the fire conditions within the control room by limiting leakage of fire effects to locations outside of those rooms.

The boundary leakage was determined from leakage fractions and the internal boundary area of the enclosure. The baseline assumption in the abandonment calculation was that the boundary leakage corresponds to tight construction as defined in the SFPE Handbook of Fire Protection Engineering. The analysis evaluated the model sensitivity to uncertainty in the boundary leakage fraction and showed that the baseline results were conservative or were not significantly sensitive to large uncertainties in this parameter. Because the boundary leakage was assumed to be uniformly distributed, it was defined as a single vent having a height equal to the enclosure height and a width that was determined from the leakage area and vent height.

Adequate airflow to support full development of the postulated fire was provided via mechanical ventilation or via natural ventilation. Validation studies confirmed that the scenarios evaluated fall within the equivalence ratio validation range. This ensures that the fires analyzed would not be limited by ventilation conditions.

NFPA 805 FM RAI 02

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

American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) Standard RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications," Part 4, indicates that damage thresholds be established to support the FPRA. The standard further indicates that thermal impact(s) must be considered in determining the potential for thermal damage of SSCs and appropriate temperature and critical heat flux criteria must be used in the analysis.

Unit 1 uses thermoplastic and thermoset cables, but due to insufficient cable material data, the FPRA assumes all cables are thermoplastic material, unless a definitive determination could be made that a cable's insulation is thermoset or equivalent to thermoset based on cable material codes contained in the Plant Data Management System (PDMS) or in vendor supplied data. Unit 2 uses all thermoset cables.

Provide the following information:

(a) For Unit 1, explain how raceways with a mixture of thermoplastic and thermoset cables are treated in terms of damage thresholds.

(b) For Unit 2 assumed to have thermoset damage criteria, confirm that the cables are actually thermoset and not just qualified by the Institute of Electrical and Electronics

Engineers (IEEE) Standard 383, "IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities."

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

SNC Response

(a) The Hatch Nuclear Plant Unit 1 Fire PRA model used the assumption that all cables within Unit 1 were of thermoplastic type. Cables in fire zones shared by Unit 1 and Unit 2 and raceways in these zones that may contain a mixture of cables types were conservatively assumed to be and modeled as thermoplastic cables. Therefore, thermoplastic damage criteria was used for all Unit 1 cables. Since thermoplastic cables have lower damage thresholds than thermoset cables, assuming all cables are thermoplastic for HNP Unit 1 is conservative.

(b) A Risk-Based Analysis (RBA) was conducted to determine the percentage of thermoset cable in Unit 2. The total lengths of Unit 2 cables were extracted from the equipment, raceway and cables database. The cable code associated with each cable was translated to an insulation or jacket type using plant records. The insulation or jacket type was than categorized as either thermoplastic and thermoset. Some of the Unit 2 cables had unknown insulation or jacket materials and were not used in the analysis. Of the Unit 2 cables that were identified as thermoplastic or thermoset, 97.39% was determined to be thermoset, given the thermoset classifications listed in the SNC specification document for cables used at Hatch. Based on this result it was concluded that the thermoset damage criteria for cables is appropriate for Unit 2.

(c) In accordance with Appendix H.2 of NUREG/CR-6850, for major components such as motors, valves, etc., the fire vulnerability was assumed to be limited by the vulnerability of the power, control, or instrument cables supporting the component. All non-cable components used the damage threshold of the cable type that the associated unit is composed of (i.e., thermoplastic for Unit 1, thermoset for Unit 2).

Damage to sensitive electronics is addressed in the response to PRA RAI 07.

Instrument air lines and pipes, verified by walkdowns, were determined to be carbon steel, stainless steel, or copper with Swagelok and mechanical connections. Given that there are no brazed connections or soldered copper joints, instrument air loss is not postulated due to fire exposure to the air lines.

NFPA 805 FM RAI 03

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

LAR Section 4.5.1.2, "Fire PRA" states that FM was performed as part of the FPRA 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.

LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805" 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.

Based on the information provided in the LAR, the NRC staff was unable to confirm whether each calculational model or numerical method was properly verified and validated, therefore, the NRC staff requests that the licensee:

(a) Describe how the Fire Modeling Workbook was verified, and describe how it was ensured that the empirical equations/correlations were coded correctly and that the solutions are identical to those that would be obtained with the corresponding chapters in the NUREG-1805 (FDTs) or FIVE-Rev.1.

(b) For any FM tool or method that was used in the development of the LAR, provide the Verification and Validation (V&V) basis if it is not already explicitly provided in the LAR Attachment J. Further, identify any applications of FM tools or methods used in the development of the LAR that are not discussed in LAR Attachment J.

(c) LAR Attachment J states that the smoke detection actuation correlation (Method of Heskestad and Delichatsios) has been applied within the validated range reported in NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications" (ADAMS Accession No. ML071650546). However, the latter reports a validation range only for Alpert's ceiling jet temperatures correlation. Provide technical details to demonstrate that the temperature to smoke density correlation has been applied within the validated range, or justify the application of the correlation outside the validated range reported in the V&V basis documents.

SNC Response

(a) Verification of the Fire Modeling Workbook is documented in the "ERIN Fire Modeling Methodology, Verification of the ERIN Fire Modeling Workbook". This report outlines the process used to verify the equations implemented in the Fire Modeling Workbook. Per the report, a set of representative input parameters of the verification calculation were entered into the Fire Modeling Workbook. These parameters define the physical analysis unit, ignition source heat release rate and location, secondary ignition source ignition, failure criteria, and detection and suppression attributes.

Solutions produced by the Fire Modeling Workbook are documented in the report. These results are compared against the hand-calculated solutions of the empirical the correlations obtained from NUREG-1805. Compared calculations included, but were not limited to, effective fire diameter, mean flame height, convective heat release rate, and total fire heat release rate. Any delta in the results was evaluated and corrected such that the Fire Modeling Workbook accurately produced the results of the hand calculations. On this basis, it was ensured that the empirical equations/correlations in the Fire Modeling Workbook were coded correctly and that the solutions are identical to those that would be obtained within the corresponding chapters in NUREG-1805 (FDTs) or FIVE-Rev. 1.

(b) The characterization in the LAR on the use of the Method of Heskestad and Delichatsios is not accurate and has been updated. An optical density method was used for the calculation of the time to actuation of an automatic smoke detection system. This optical density method was used because it more appropriately matches the physics of how automatic smoke detectors detect fires compared to the method of temperature rise as described by Heskestad and Delichatsios. The validation of time to automatic smoke detection using the smoke optical density is presented in the response to FM RAI 03(c).

(c) The optical density smoke detector activation method was used to determine the time to activation of automatic smoke detectors. The method is explained in detail in the Fire Modeling Workbook. A summary of the development of the method, the validation of the method, and the demonstration that the method was used within the validation range are provided in this RAI response.

The optical density method was developed because it more accurately matches the physics of how automatic smoke detectors operate, such as, using smoke concentration, rather than temperature increase as the trigger for detection. The method was developed by starting with the Alpert's ceiling jet temperature correlation, making a series of substitutions (using recommended values from fire industry journals and handbooks), and ending with the optical density method to determine time to smoke detection, based on optical density, rather than temperature. The optical density method involves the following calculations:

- Ceiling jet temperature. As presented in NUREG-1805, Chapter 11.5.1, the Alpert's Ceiling Jet correlation provides the ceiling jet temperature, as a function of the total heat release rate of the fire, the height of the ceiling above the fuel source, and the radial distance from the plume center line to the detector.

- Ceiling jet density. The ceiling jet temperature (Alpert's correlation) is converted to density, using the ideal gas law.

- Dilution factor. The dilution factor is a measure of how much entrainment (i.e., dilution) has occurred in the fire plume and ceiling jet in order to achieve the temperature rise at the detector, as calculated by Alpert's ceiling jet correlation. Using the First Law of Thermodynamics, the temperature increase of the ceiling jet above ambient conditions at the radius of the smoke detector can be shown to be inversely proportional to the amount of air entrained (i.e. more entrainment – lower temperature and vice versa).

- Soot density. Using the values of ceiling jet density and the dilution factor already calculated, and the soot yield in the fuel that is burning, the soot density in the ceiling jet is calculated.

- Optical density. The soot density is converted to optical density by multiplying by the specific light extinction coefficient.

- Arrival time. The arrival time is calculated, incorporating the delay time for smoke arrival at the detector following the guidance presented in NUREG-1805 (Chapter 11).

Validation of the optical density smoke detection method was performed using results from home smoke alarm performance test results summarized in NIST Technical Note 1455-1. NIST Technical Note 1455-1 contains data from a series of smoke alarm tests that were used to estimate the performance of smoke detectors in residences. These test series were also used in the validation of smoke detector activation times summarized in Supplement 1 to NUREG-1824. Tests were performed in a manufactured home using various ignition sources (upholstered chair and mattress) and detector locations. For this validation, only the results of activation times for three detectors (ionization and photoelectric) were appropriate to use. The other detector locations did not meet the requirement of Alpert's Correlation that ceiling jet be unimpeded. Results show a positive bias (later detection time when compared to the experimental results) in the optical density smoke detector activation method of up to +24% and a standard deviation of up to 50%. These validation results demonstrate that the optical density smoke detection method results in an average over-prediction for modeling the activation times for smoke detectors.

To show that the optical density method was used within the applicable validation range, the appropriate normalized parameter (Ceiling Jet Distance Ratio: the ratio of the ceiling jet radial distance to the vertical distance from the base of the fire) was calculated for each scenario in which the method was applied. Even though the optical density method is not specifically included in Supplement 1 to NUREG-1824, the validation range of 0 to 8.3 for the Ceiling Jet Distance Ratio is applicable to the optical density method, since the method is based on the Alpert's correlation. In all cases the normalized parameters for the scenarios that used the optical density method were within the validation range.

Note that the optical density method was not used in those areas where obstructions could impede the flow of smoke and delay automatic detection timing. In those cases, the time to automatic detection was determined using a CFAST analysis, in which the time to automatic detection was the time at which the simulated upper layer reached a depth greater than the bottom of the ceiling beams and reached the optical density threshold.

Attachment J, Fire Modeling V&V, of the Transition Report has been revised to reflect the use of the optical density smoke detector activation method

References:

1. Bukowski, R., et al. "Performance of home smoke alarms." NIST Tech. Note 1455-1 (2008).
2. NUREG-1824, Supplement 1, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," EPRI 3002002182, Electric Power Research Institute, Palo Alto, CA and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES) Rockville, MD, May 2015
3. Overholt, K. Verification and Validation of Commonly Used Empirical Correlations for Fire Scenarios, NIST Special Publication 1169, National Institute of Standards and Technology, March 2014
4. McGrattan, K. B., et al., "Fire Dynamics Simulator, Technical Reference Guide, Volume 3: Validation," National Institute of Standards and Technology and VTT Technical Research Centre of Finland, Sixth Edition, November 2019.

NFPA 805 FM RAI 04

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

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

Based on the information provided in the LAR, the NRC staff was unable to determine whether the engineering methods used were applied within the scope, limitations, and assumptions prescribed for those methods, therefore, the NRC staff requests that the licensee identify uses, if any, of the FMWBs outside the limits of applicability of the method and for those cases outside of the limits, explain the analysis that was used or why the use of the FMWBs was justified.

SNC Response

Supplement 1 to NUREG-1824 (Table 3-3) lists various normalized parameters that can be used to determine whether a modeled fire scenario fits with the range of the experimental parameters. The normalized parameters for the fire scenarios modeled in the Fire Modeling Workbook were compared against the validation applicability ranges provided in Supplement 1 to NUREG-1824 (Table 7-1).

The normalized parameters for Froude Number and Flame Length Ratio fell within the acceptable validation limits provided in NUREG-1824 for all scenarios evaluated using the Fire Modeling Workbook.

The Ceiling Jet Ratio, which is calculated external to the Fire Modeling Workbook, was found to be within the validation limits provided in Supplement 1 to NUREG-1824 for scenarios where the Fire Modeling Workbook was used to estimate the time to automatic detection.

For the Radial Distance Ratio, a small fraction of scenarios did not fall within the validation limits provided in the Supplement 1 to NUREG-1824 at low heat release rates; however, the results are conservative, based on the following justification. The Radial Distance Ratio is characterized as the radial distance over the fire diameter. In the Fire Modeling Workbook the fire diameter is computed using a user prescribed upper and lower limiting Fire Froude number. Per NUREG-1934, a typical Froude number for an accidental fire is around 1. The upper and lower limit Fire Froude numbers used in this analysis are 0.2 and 2.5 representing reasonable bounds for buoyancy-driven and momentum-driven fire plumes respectively. Using these bounds and the scenario specific radial distance, an estimation of the scenario HRRs required to meet the validation limits may be determined. For each scenario the calculated range of validation limit acceptable HRRs ultimately exceeded the peak scenario source HRR. This suggests that target damage occurred prior to reaching a HRR that would have created a corresponding fire diameter needed to reach the validation limits. While outside the validation limits provided in in the Supplement 1 to NUREG-1824, these results are conservative as they suggest damage occurred at lower HRR percentiles.

The Equivalence Ratio and Compartment Aspect Ratio do not apply to the Fire Modeling Workbook Analysis. Instead, the validation applicability ranges for these two parameters apply to the CFAST fire models. The normalized parameters for Equivalence Ratio and Compartment Aspect Ratio fell within the acceptable validation limits provided in NUREG-1824 for all single compartment scenarios evaluated using CFAST. In the Multi-Compartment Analysis, the exposing compartment for one of the calculations has a compartment aspect ratio of 0.38, which is below the lower limit of 0.6. With a single fixed height shared among each volumetric group used for the simulation compartments, the lower ratio for this simulation results from a smaller compartment volume to heat up and produces a conservative analysis.

Additionally, a sensitivity was performed for the cases where the probability of non-suppression (PNS) for these scenarios was changed to 1.0 – essentially removing the contribution of the time to damage calculation from the analysis. The removal of the PNS changed individual scenario risk contribution from values on the order of 1E-11 to 1E-10. The increased risk contribution of these scenarios was negligible if the benefit of performing fire modeling was removed.

NFPA 805 FM RAI 05

Section 2.7.3.4, "Qualification of Users," of NFPA 805, states, "cognizant personnel who use and apply engineering analysis and numerical models (e.g., FM 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."

LAR Section 4.5.1.2, "Fire PRA" states that FM was performed as part of the FPRA development (Section 4.2.4.2 of NFPA 805). This requires that qualified FM and PRA personnel work together. Furthermore, LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," 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. FM) 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 FM or FPRA 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 Attachment S, Table S-3, Implementation Item IMP-8.

Based on the information provided in the LAR, the NRC staff was unable to determine how the licensee complies with NFPA 805, Section 2.7.3.4, therefore, the NRC staff requests that the licensee:

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

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

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

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

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

SNC Response

(a) As a baseline, the appropriate qualifications for the staff and consulting engineers to use and apply the fire modeling methods and tools are consistent with Regulatory Guide 1.189, Section 1.6.1.a and Section 2.2 of NEI 07-12. NEI 07-12, Fire PRA Peer Review Process Guidelines, includes qualifications for each Peer Review.

Regulatory Guide 1.189 states that Fire Protection Staff or Consultants should be a graduate of an engineering curriculum of accepted standings and satisfies the eligibility requirements as Member grade (or Professional Member grade) in the Society of Fire Protection Engineers. Section 2.2 of NEI 07-12 requires that members of the Peer Review Team have at least a bachelor's degree in Engineering/Science/Mathematics, and at least five years of nuclear plant PRA experience which should include performance of management of one previous Fire PRA project.

The requirements of SNC procedure S-ES-FIREPRA2, Job Performance Requirement (JPR), are given in the response to RAI FM 05(b). By satisfying these requirements, the Fire PRA Engineer is qualified for selected Fire PRA tasks including Scenario Development and Fire Modeling.

Prior to the selection of JENSEN HUGHES as the Fire PRA vendor, SNC reviewed the fire modelers' credentials. Credentials included education in fire protection engineering and fire modeling, and extensive experience performing fire modeling studies. SNC reviewed the vendor's credentials of the analysts performing the fire modeling tasks and ensured that each task was performed by analysts with appropriate training in the fire modeling area being performed.

(b) 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 Attachment S, Table S-3, Implementation Item IMP-18.

SNC procedure S-ES-FIREPRA2, Job Performance Requirement (JPR), qualifies a Fire PRA Engineer for selected Fire PRA Tasks including Scenario Development and Fire Modeling and Sensitivity and Uncertainty Analysis. The procedure contains requirements for trainees and mentors to ensure that engineers performing fire analyses and modeling activities are qualified to perform the Fire PRA tasks described NUREG/CR-6850 (Task 8: Scoping Fire Modeling, Task 11: Detailed Fire Modeling, and Task 15: Uncertainty and Sensitivity Analyses).

(c) The analyst who prepared the Main Control Room Abandonment Calculation (H-RIE-FIREPRA-U00-008B) is a Senior Fire Protection Engineer at JENSEN HUGHES and a graduate of an accepted engineering curriculum (Worcester Polytechnic Institute) with a B.S. degree in Civil Engineering, an M.S. degree in Fire Protection Engineering, and a member grade in the Society of Fire Protection Engineers (SFPE).

With support of Principle Engineers at SNC, the Technical Lead of the Fire Modeling Tasks led the walkdowns of the MCR and remaining fire areas in the plant. This analyst is a Principle Engineer at JENSEN HUGHES and a graduate of an accepted engineering curriculum (University of Maryland) with a B.S. and an M.S. degree in Fire Protection Engineering. Walkdown information – including pictures, video, notes, etc. – were developed and discussed with all members who performed fire modeling tasks, including those associated with the MCR.

(d) During the development phase of the Fire PRA, the Fire Modeling analysts and the PRA personnel maintained frequent communications. The Fire Modeling analysts populated the workbooks, spreadsheets, and fire models from which the target sets and scenario frequencies were calculated, while the PRA engineers were primarily responsible for the PRA model and its quantification. Cooperation between the fire modeling analysts and the PRA engineers was achieved through weekly teleconferences, periodic in-person meetings, and cutset review meetings during the development of the Fire PRA. These meetings assured that the inputs and refinements necessary were effectively communicated.

Post transition, SNC will maintain Position Specific Guides that 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. The training will address the interactions between the fire modeling analysis and the PRA model and will emphasize the importance of communication between fire modeling analysts and Fire PRA engineers.

(e) The communication process between the consulting engineers and the HNP staff, held during the course of the NFPA 805 Fire PRA model transition, consisted of project meetings, both onsite and by phone, and plant reviews of draft deliverables. The meetings and reviews included consideration of the technical adequacy of the fire modeling as applied at HNP. The Fire PRA team remained in close contact with the plant throughout the development of the Hatch Fire PRA. Walkdowns were often performed with both SNC personnel and consulting staff together for scenario development. Consulting engineers, SNC corporate staff, and Site Operations met face to face during critical stages of Fire PRA and LAR development. Interactions between SNC personnel and consulting engineers were encouraged by SNC management and was recognized as an opportunity for knowledge transfer.

The measures taken to assure that fire modeling as adequately performed during the development of the Fire PRA (i.e., prior to the NFPA 805 transition) are described in Part b of FM RAI 06.

Once the NFPA 805 transition process is completed, fire modeling calculations will be performed by Fire Protection or PRA Engineers who meets the qualification requirements of Section 2.7.3.4 of NFPA 805. This will be ensured through qualification requirements and training that will be developed as described in Table S-3, Implementation Item 18. An analyst will be required to complete this qualification before modeling in support of the Fire PRA or a qualified person will need to review and sign off the prepared material before its use within the Fire PRA. Qualifications will be tracked through SNC's training program and will be procedurally required to be checked prior to completing the task that requires fire modeling.

NFPA 805 FM RAI 06

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

LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," states that "Uncertainty analyses were performed as required by 2.7.3.5 of NFPA 805 and the results were considered in the context of the application. This is of particular interest in FM and FPRA development."

Based on the information provided in the LAR, the NRC staff was unable to determine how the licensee complies with NFPA 805, Section 2.7.3.5; therefore, the NRC staff requests for the uncertainty analysis for FM, that the licensee:

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

(b) Describe how the "model" uncertainty was accounted for in the FM analyses.

(c) Describe how the "completeness" uncertainty was accounted for in the FM analyses.

SNC Response

(a) To account for the uncertainty in the fire modeling input parameters, the input parameters to the HNP Fire PRA were chosen from the range of industry accepted values such that conservatism was preserved, as described below.

1. The input parameter that has the most impact on results is the heat release rate (HRR). The uncertainty associated with this parameter is accounted for as follows:

    a. The gamma distributions of the heat release rate values for most fixed ignition-source types are taken from Appendix E of NUREG/CR-6850 and Section 4 of NUREG-2178. The mean and representative percentile values for each ignition source type are presented in Table E-1 of NUREG/CR-6850 and Tables 4-1 and 4-2 of NUREG-2178. Use of these industry guidance documents ensures that the heat release rates used in the analysis align with test data and accepted practices. [Reference: H-RIE-FIREPRA-U00-008A]

b. In the case of oil fires, the mean value is calculated using the equation for heat release described in Appendix G of NUREG/CR-6850 and Appendix C.3 Report H-RIE-FIREPRA-U00-008A. The uncertainty associated with the heat release rate intensity and duration of oil fires is mostly governed by the postulated size of the spill and the amount of oil. The Fire PRA assumes bounding heat release rates for oil spills to account for the uncertainty associated with the amount of oil in the ignition source (i.e. pumps). At the same time, the frequency of oil fires is calculated using two probabilistic parameters: 1) the probability that the ignition source fire is associated with oil (spilt fraction from Table 6-1 NUREG/CR-6850) and 2) the severity factor for small and large oil fires described in Appendix E of NUREG/CR-6850. These inputs are treated as uncertain parameters in the risk equation.

c. In the Main Control Room Scenarios, the fire models include a 98% transient heat release rate and the open back Main Control Board heat release rate. These bounding heat release rates ensure that the calculated abandonment time is conservative. [Reference: H-RIE-FIREPRA-U00-011E]

2. The material properties recommended in NUREG-7010 or NUREG/CR-6850 are used in the HNP FLASH-CAT models. [Reference: H-RIE-FIREPRA-U00-008A] These empirically derived material properties include the heat of combustion per unit weight of cable insulation, the heat release rate per unit area of cable insulation, the horizontal and vertical flame spread rates, mass per unit length of cable, the plastic yield, and the char yield. Specifically:

   a. NUREG-7010 reports that the effective values of heat release rate per unit area (150 kW/m^2 for thermoset cables, 250 kW/m^2 for thermoplastic cables) yielded predictions of the total heat release that were comparable or greater than the experimentally measured values. [Reference: H-RIE-FIREPRA-U00-008A]

   b. Although the fire in many of the multiple tray experiments documented in NUREG-7010 did not spread beyond the V-shaped region formed by the initial upward spread of the fire, the HNP FLASH-CAT model implements the horizontal flame spread rates from NUREG/CR-6850 as recommended by NUREG-7010. [Reference: H-RIE-FIREPRA-U00-008A]

   c. The timing sequence for the upward spread of fire in a vertical array of horizontal cables reported in NUREG/CR-6850 predicts reasonably well the vertical spread of the fire. This empirically derived timing is used in the HNP FLASH-CAT model. [Reference: H-RIE-FIREPRA-U00-008A]

   d. The mass per unit length of cable, char yield, plastic yield, and the number of cables in the tray directly impact the burning duration. The input values chosen for these parameters yield a burning duration for a single 1 sq. m. cable tray segment of greater than 50 minutes. Therefore, the segment ignited by the fire will continue burning for nearly 1-hour after ignition. Therefore, burnout is not expected until after a damaging hot gas layer has already formed. [Reference: H-RIE-FIREPRA-U00-008A]

3. With respective to fire growth, fire scenarios as currently modeled begin with a t^2 (t-squared) fire growth profile start at time, t = 0. This simplification ignores the incipient phase of the fire. This approach is consistent with NUREG/CR-6850. [Reference: H-RIE-FIREPRA-U00-008A]

4. The time to damage for a cable is determined from the data presented in Table H-5 through H-8 in NUREG/CR-6850. See response to RAI FM 01.e regarding the use of this cable damage data. When exposure is expected to be direct flame spread due to high heat flux or temperature exposure, the assumed damage timing was considered one minute as the most conservative value recommended in Tables H-5 thru H-8 in NUREG/CR-6850.

The following input parameters are not a significant source of uncertainty. Rather, each of these inputs reflect plant-specific conditions documented in plant drawings or observed during detailed fire modeling walkdowns.

1. The shortest distance between the source and the target was determined by a thorough review of plant drawings and verification of those values through detailed fire modeling walkdowns.

2. The cable tray arrangements (i.e., tray width, separation distance between vertically arrayed trays, the count of trays in the vertical array) used in the HNP FLASH-CAT Models reflect plant conditions. These values were determined by a thorough review of plant drawings or through detailed fire modeling walkdowns.

3. Compartment geometry and ventilation characteristics are obtained from plant drawings. In CFAST, these attributes are sometimes simplified, or a representative value is chosen. Namely, the compartment geometry is converted into a parallelepiped for which volume is conserved. The screening methodology employed in the multi-compartment analysis assumes that the two compartments are located on the same elevation and are separated by a common wall. However, there are some multi-compartment combinations in which the exposed compartment is above the exposing compartment. Therefore, a sensitivity case was performed to determine if the results of the screening analysis would change if the CFAST model placed the exposed compartment above the exposing compartment (i.e., the common boundary was the floor/ceiling assembly). The method and results of this evaluation are presented in Report H-RIE-FIREPRA-U00-008C. The results show a small difference of about 10°C in either direction (higher and lower temperatures). This difference is not considered a substantial difference given the number of unique iterations performed for each simulation group which are intended to ensure a conservative analysis was achieved.

(b) The model uncertainties associated with fire scenario development and detailed fire modeling for the HNP Fire PRA includes uncertainties related to the selection of transient zones, fire location, fire growth and propagation, activation and function of the detection and suppression systems, the selection of damage criterion, conduit routing, selection of fire models, and the inputs to the chosen fire models. A summary of the uncertainty sources and treatments associated with the fire scenario development and detailed fire modeling in the HNP Fire PRA is provided as follows:

1. Uncertainty Associated with the Scenario Development Process

   a. Selection of transient zones and scenarios: Transient fires are postulated so that all open floor areas are captured in the analysis. However, there is uncertainty in the selected sizes and boundaries of the transient zones. As a conservative practice, and when practically possible, the transient zones have been selected large enough to capture target damage beyond a typical zone of influence of an ignition source. In addition, a target overlap between transient zones has been incorporated in the analysis to account for targets near the boundaries. That is, the targets located nearby the boundaries between transient zones have been mapped to all transient zones adjacent to that boundary. When applicable, fire propagation between transient zones is included in the scenario progression.

   b. Fire Location (All Fire Zones except the Main Control Room): The location of a fire is unavoidably a source of uncertainty as all fires in the Fire PRA are assumed to occur at a specific spatial point within the fire zone. This fire location impacts the heat process of nearby targets. The source of uncertainty is addressed in the Fire PRA in a consistent approach. That is, the guidance for assigning fire location has been consistently applied throughout the analysis. The guidance is based on a conservative practice of assigning fixed and transient fire locations consistently for all scenarios. For fixed ignition sources, the fires have been postulated at the elevation of the ignition source. In the specific cases of potentially risk-significant electrical panels, the fires are conservatively postulated at the top of the cabinet. In the case of general transients and transients due to hotwork, the base of the fire has been postulated 2 ft. above the floor. Since the transient fires are due to items brought into the fire zone on a temporary basis, this 2 ft. elevation is a practical assumption to account for combustibles that may not be located a floor level. This fire height is considered representative of typical equipment carts, for example. Oil spill fires are postulated at floor level.

2. Uncertainties Associated with Detection and Suppression

   a. Activation Time: The activation times for detection and suppression systems is a source of uncertainty in all Fire PRA's primarily because of the complexity of the configurations encountered in the scenarios postulated across multiple different fire zones. An example of a complex configuration includes devices mounted at or near obstructed ceilings. As a conservative practice, activation times have been calculated using bounding values. For example, bounding vertical and horizontal distances from the ignition source to the device are used in an attempt to bound complex configurations.

   b. Suppression Time: The ability to control or completely suppress the fire as a function of time using the different suppression means available in the given scenario is a source of uncertainty. In the HNP Fire PRA, detection and suppression is treated with an event tree approach where both outcomes, i.e., successful and unsuccessful suppression, are considered. In addition, suppression times are conservatively addressed based on selected input parameters to the fire models and the use of validated models.

c. Damage Criterion: The damage criterion is a source of uncertainty that is considered in the analysis. In Unit 1, thermoplastic damage criteria from NUREG/CR-6850, Appendix H is applied. Thermoset damage criteria are applied to Unit 2 fire zones only after a review of cable insulation material was completed. Targets in any fire zone shared between the two units is conservatively assigned the thermoplastic damage criteria.

d. Conduit Routing: Conduit routing is a source of uncertainty that is considered in the analysis. Non-appendix R conduits are not dimensioned in plant drawings and conduits are only labeled in at each end. Consequently, routing of conduits has been evaluated on an individual basis based on their location within the route string of a cable. Conduits containing important cables were walked down and their exact location identified. Other conduit locations represented in the Fire PRA bounds this uncertainty by mapping the cable to all the "logical" places in which the conduit might be located.

e. Fire Modeling Selection: For the HNP Fire PRA, CFAST and hand calculations have been utilized. These models have been subjected to verification and validation studies for selected scenarios as described in NUREG-1824. An analysis was performed to determine whether models were used within their validated range. If the models were found to be used outside of the range, then the input parameters were varied in a conservative direction (i.e., more challenging fire conditions) and the revised model results were used as input to the Fire PRA.

3. Uncertainty of Parameters and Risk Quantification

Consistent with the guidance, uncertainties associated with the fire modeling parameters are reflected in the risk quantification as follows:

a. Severity factor: In addition to the conservative determination of critical heat release rate values (i.e., through selection of the shortest distance and of the fire location), the uncertainty associated with the severity factory is explicitly modeled in the uncertainty task of the Fire PRA.

b. Non-suppression probabilities: These values are calculated using the time to damage resulting from fire modeling.

c. Conditional core damage probability/conditional large early release probability: These probabilities are calculated based on the targets associated with each fire scenario. Mapping of targets of the different fire scenarios follows a conservative process to ensure that the resulting probabilities are bounding.

(c) "Completeness" uncertainties refers to the fact that a model may not be a complete description of the phenomena it is designed to predict. Completeness uncertainty was addressed by the same process used to address the model uncertainty. The responses to part (b) for "model" uncertainties and part (a) for parameter uncertainties should be considered integral to accounting for "completeness" uncertainty.

In PAUs with a large number of ignition sources and targets, such as working floors of the Reactor Building, Turbine Building, and Control Building, there are practical limitations on the amount and detail of fire scenarios that can be analyzed. Due to this limitation, the potential impact of ignition sources on PRA targets is conservatively characterized. Specific conservative treatments related to the "completeness" uncertainty include Scoping Fire Modeling and Assumed Target Routing.

For the scenarios where scoping fire modeling is used, the severity factor is calculated based on the closest target or intervening combustible. The ignition source is then conservatively assumed to either damage all targets in the fire zone or a reduced number of targets based on the results of a walkdown of that ignition source. The basis for this approach is:

1. Scoping fire modeling is a simplified approach that can be refined where needed based on risk quantification results.

2. In some cases, scenario refinement provides little or no value because:

    a. Detection or suppression does not actuate for the source

    b. Detection or suppression actuates after most targets are damaged. This includes large fires with a fast growth rate (e.g., oil fire scenarios)

3. Due to cable routing limitations, there are a significant number of assumed targets in the zone. Consistent with the documented methodology, these targets will fail regardless of heat release rate or detection/suppression. Therefore, any refinements would do little to reduce the targets impacted by the scenario.

Once scoping fire modeling is performed, the risk contribution of the scenario may be acceptable and the application of the non-suppression probability my not be necessary.

There are a subset of PRA targets whose location is unknown due to the lack of labeling, obstructed views, or other limitations. These targets are assumed to be vulnerable to damage by any ignition source in the fire zone. If possible, photos or raceway drawings were used to identify the path of travel or general location of an assumed target in order to limit the number of fixed and transient ignition sources in a given PAU that would damage the target. During the fire modeling and quantification process, an iterative approach was used to minimize the number of high-risk targets whose route and proximity to ignition sources was assumed. This was done by locating the highest risk targets whose location was being assumed. The remaining assumed targets is a large source of conservatism and accounts for the potential for the targets to be damaged by any fire where the target may reasonably exist.

Edwin I. Hatch Nuclear Plant Units – 1 and 2
<u>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</u>

Enclosure 2
Response to Safe Shutdown Analysis (SSD) RAIs

NFPA 805 Safe Shutdown (SSD) RAI 01

NFPA 805, Section 2.4.2, "Nuclear Safety Capability Assessment," requires licensees to perform a nuclear safety capability assessment (NSCA). RG 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, (ADAMS Accession No. ML092730314) endorsed the guidance in [*sic*] NEI 00-01, Chapter 3, as one acceptable approach to perform an NSCA. Nuclear Energy Institute, (NEI) 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," (ADAMS Accession No. ML091770265) Section 3.5.2, indicates that with respect to the electrical distribution system, the issue of breaker coordination must also be addressed.

The licensee's cable selection and circuit failure analysis indicates that some devices may not be coordinated or coordination may be undetermined but will be addressed through procedures. Since the probabilistic risk assessment (PRA) treats all credited power supplies as having proper electrical coordination:

(a) Discuss whether a comprehensive electrical coordination study for the credited power supplies has been completed and whether all issues have been identified and resolved. If not, provide a proposed path forward to resolve the outstanding issues.

(b) Discuss any outstanding issues which should be considered for inclusion in LAR Attachment S, as modifications or implementation items as necessary.

SNC Response

(a) Electrical coordination studies for credited power supplies have been completed per HNP LAR, Attachment B, Sections 3.3.1.1.4, 3.3.1.1.5, 3.3.2, 3.5.2.4 and 3.5.2.5. All breaker coordination issues have been resolved except for the associated plant modifications listed in HNP LAR Table S-2, Items #6 and #7. One remaining issue, Multiple High Impedance Faults (MHIF), will be addressed using the guidance in NUREG/CR-7150 Volume 3 and documented per the HNP LAR Table S-3, Item IMP-9.

(b) All outstanding issues have been included in LAR, Attachment S. Specifically, LAR Table S-2, Items #6 and #7 and LAR Table S-3, Item IMP-9.

NFPA 805 SSD RAI 02

NFPA 805 Section 3.11.3, "Fire Barrier Penetrations," indicates that penetrations in fire barriers shall be provided with listed fire-rated door assemblies or listed rated fire dampers having a fire resistance rating consistent with the designated fire resistance rating of the barrier as determined by the performance requirements established by NFPA 805 Chapter 4, and that passive fire protection devices such as doors and dampers shall conform with NFPA 80, "Standard for Fire Doors and Fire Windows."

In order to meet the requirements of NFPA 80, the licensee has proposed a plant modification to relocate or install fusible links on certain sliding fire doors, as described in LAR Attachment S, Table S-2, Modification Item 1. The licensee indicates that this modification also addresses the potential of water intrusion into Switchgear 1R23S004 located in Fire Area 1017 from fire suppression activities in Fire Zone 0014K. For a postulated fire in Fire Zone 0014, discuss:

(a) The actuation time of the fusible link and the suppression system, and whether the fusible link actuated sliding fire door will close in a timely manner as to preclude potential water damage to Switchgear 1R23S004 from fire suppression activities.

(b) Since sliding fire doors are not watertight doors, discuss the impact, if any, on Switchgear 1R23S004 due to water migration into Fire Area 1017.

Describe how this modification ensures Switchgear 1R23S004 will be protected.

SNC Response

(a) In addition to addressing the NFPA 80 requirements for the number and location of sliding door fusible links, the Table S-2, Item 1 modification will also replace the existing 180° angle nozzle located outside of door 1L48C31 (Component ID 1Z43N434AR on wet-pipe sprinkler system 1Z43130W01) with a 65° angle nozzle. The new nozzle will eliminate the potential for water intrusion into Switchgear 1R23S004 during sprinkler discharge. As a result, actuation of the sliding door fusible link is not required prior to suppression system actuation to preclude potential water damage to Switchgear 1R23S004 during sprinkler discharge. The proposed modification for Table S-2, Item 1 is revised as shown in Attachment S to this letter.

(b) Any water that passes through the door or gaps between the door and wall will not have an impact on Switchgear 1R23S004. Until the door is closed, the new 65° angle nozzle will eliminate the potential for direct water spray onto the switchgear and will minimize the amount of water spray into Fire Area 1017. Additionally, an existing 1.5-inch curb at the entrance to Fire Area 1017 serves a dual-purpose to protect Switchgear 1R23S004 from fire suppression activities. First, it will limit water build-up within Fire Area 1017 to 1.5 inches and allow water to flow out of the room prior to damaging the switchgear, which sits on a 3-inch pad. Second, this curb along with the multiple floor drains that are provided within Fire Zone 0014K will prevent any water build-up in Fire Zone 0014K from flowing into Fire Area 1017.

NFPA 805 SSD RAI 03

NFPA 805 Section 4.2.1 requires one success path necessary to achieve and maintain the nuclear safety performance criteria (NSPC) shall be maintained free of fire damage by a single fire, and that the effects of fire suppression activities on the ability to achieve the NSPC shall be evaluated.

In LAR Attachment C, Table C-1, the discussion of fire suppression effects in many fire areas includes "water from some deluge or sprinkler systems and from hose streams might temporarily exceed the capacity of the drain system in some areas. However, safety related equipment is elevated above the floor level by pads or pedestals, such that equipment is protected from flooding."

Based on the information provided in the LAR, the NRC staff was unable to determine whether the effects of fire suppression activities on the ability to achieve the NSPC have been properly evaluated for areas where flooding is a concern and pads and pedestals are credited to protect SSD equipment.

Provide a summary of the internal flooding analysis that demonstrates the raised pads and pedestals are adequate in height.

SNC Response

Quantitative internal flooding analysis calculations were not credited within the suppression effects analysis. The statement was included generically for all 46 fire areas that contain floor drains and NSCA equipment. However, insight from SNC calculations SMNH-03-002, "Circulating Water System Flooding Analysis"; SMNH-02-010, "Circulating Water Flood Level in the Turbine and Control Buildings"; and Document of Engineering Judgment DOEJ-HR2102089601-M001, "Evaluation of the Impact of Fire Sprinkler Actuation in the Hatch Cable Spreading Rooms," collectively support the conclusion that the drainage systems in the control and turbine buildings are capable of handling a worst case flood in those buildings. The analyses in SMNH-03-002 and SMNH-02-010 for circulating water system flooding bound a worst-case fire system flooding scenario, including up to volumes equal to an entire fire water tank. The analysis in DOEJ-HR2102089601-M001 is a typical evaluation for a fire suppression system actuation, which demonstrates the adequacy of the floor drainage in the area of the suppression system actuation.

The HNP-2-FSAR, Subsection 9.3.3 "Equipment and Floor Drainage System" provides bases for the equipment and floor drainage systems within the reactor buildings. The reactor building diagonal rooms, the high-pressure coolant injection (HPCI) room, and the torus chamber room are each equipped with instrumented floor drain sumps. These sumps gravity drain to the reactor building floor drain sumps located in the southeast and southwest diagonal rooms. Flooding of a diagonal room or the HPCI room due to a line break in the room can be confined to that room alone by means of the remotely operated isolation valves in the drainage system. Should the capacity of the sump pumps be exceeded and the sumps overflow, the diagonal rooms and the HPCI room will isolate from each other and from the torus area to confine the water to that particular room, thus preventing damage to any redundant equipment. Redundancy of essential equipment and physical separation of the diagonal rooms, coupled with the remote isolation capability of the drainage system, ensures the protection of the ECCS against common flooding events. As discussed above for the control and turbine building flooding scenarios, these features in the reactor building will protect against both fire pipe breaks scenarios, as well as for any suppression system actuation in a reactor building area.

In response to this RAI, the suppression effects analysis has been refined to remove the statement noted above for 29 fire areas in which credit for raised pads and pedestals is not necessary (e.g., because the fire area does not contain a suppression system and is adequately separated from other fire areas). For the remaining 17 fire areas, area-specific bases were developed that justify the drainage of the area based on the documents identified above. Attachment C, Fire Area Transition, of the Transition Report has been revised to reflect the changes made to the suppression effects analysis.

Edwin I. Hatch Nuclear Plant Units – 1 and 2
<u>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</u>

Enclosure 3
Response to Health Physics (HP) RAIs

NFPA 805 Health Physics (HP) RAI 01

LAR Attachment E, "Radioactive Release Transition" states that the potential release of contaminated effluents resulting from a fire involving radioactive contents in the Bounded Areas compartment is bounded by Vendor Document S77684. In addition, Attachment E states, "This calculation demonstrates that releases are below 10 CFR 20 limits and satisfies the acceptance criteria of FAQ 09-0056." For Vendor Document S77684, please provide a summary of the assumptions, methodology, input parameters, resulting doses and conclusions.

SNC Response

The overall NFPA 805 radioactive release review task assessed the potential for radioactive release due to fire suppression activities when utilizing the plant's pre-fire plans, Fire Brigade training materials and engineering controls. The focus of Vendor Document S77684 is to bound the potential radioactive release due to fire suppression activities from contaminated liquid and gaseous effluents resulting from a fire involving radioactive contents in screened-in locations where engineering controls are deemed insufficient to prevent radioactive release beyond plant boundaries, not having monitored, fixed drainage and HVAC systems.

Vendor Document S77684 provides a bounding dose analysis for the Plant Hatch site on the release of radioactivity contained within a maximally loaded low-specific activity (LSA) container due to a fire and/or firefighting activity. The analysis demonstrates that the release of radioactivity to any unrestricted area from airborne and/or liquid pathway releases due to the direct effects of fire suppression activities (but not involving fuel damage) is as low as reasonably achievable and will not exceed the applicable 10 CFR Part 20 instantaneous dose limit of 2 mrem in any hour; and that the limitations on the instantaneous release of gaseous and liquid effluents specified in the Plant Hatch Technical Specifications is not exceeded.

Methodology:

The methodology includes using the NRC computer software RASCAL to calculate the airborne pathway dose consequences and a spreadsheet to calculate the liquid pathway doses.

The radioactivity contained in the standard container is calculated using the RADMAN radwaste computer program. The RADMAN program calculated the radioactivity in the standard container when it is maximally loaded with radioactive waste using the dimensional characteristics of the standard 20-foot intermodal container (SeaLand container), the Plant Hatch radwaste source term, and the maximum permitted dose rate of 10 millirem/hour (mrem/hr) at two meters from the surface of the standard container.

For the airborne effluent pathway, the quantity of radioactivity released by the fire is calculated using the default fire release fractions described in the RASCAL 4.3 User's Guide for all radioactivity contained in the standard container. For the liquid effluent pathway, all of the radioactivity contained in the standard container is released by the fire.

The key assumptions and input parameters:

- The release occurs over the course of 1 hour. This very conservatively assumes the Fire

Brigade continues discharging the firefighting hose stream for the entire period in calculating liquid pathway doses.

- The standard container used for low-specific activity waste shipments at the plant is completely consumed by a fire with the resulting release of all contained radioactivity over a period of one hour.
- For the airborne pathway effluent, for the entire radioactivity contained in the standard container, the default fire release fractions described in the RASCAL 4.3 User's Guide is released from the container to the atmosphere.
- For the liquid effluent pathway, the entire radioactivity contained in the standard container is released by the fire runoff. No activity is released via smoke to the atmosphere. All is available for liquid release in site runoff to the Altamaha River.
- The one-hour duration is based on the use of a container that does not support combustion and the use of administrative controls which limit the amount of time a container is open; and prevent the storage of materials that could affect container integrity or serve as an ignition source.
- Dose conversion factors from ICRP Reports 26 and 30 contained in Federal Guidance Report (FGR) Reports No. 11 and No. 12 are used for the calculations.
- The waste weight is at the limit for the container maximum gross weight. The 20-foot non-combustible SeaLand container is the limiting source term, with its proportionally higher effective density.
- The most limiting atmospheric stability condition is used (atmospheric stability class G and wind speed <4 mph). This maximizes the effective dose received.
- The fire protection hose stream flow rate from firefighting activities is assumed to be 250 gpm for one hour (Ordinary Hazard occupancy). That results in a total volume of water discharged of 15,000 gallons.
- The average site dilution flow rate for releases is 10,000 gpm. The Altamaha River discharge as measured by the nearest USGS river gauge near Baxley, GA is 10,000 CFS averaged over the three-year period May 2013 to May 2016.
- Per the Plant Hatch Offsite Dose Calculation Manual, there is no drinking water or irrigated garden pathway for the Hatch site.
- The lower limit of detection (LLD) values for H-3, C-14, I-129 and Tc-99 as provided in the RADMAN source term were conservatively assumed to be present at their LLD values.

Resulting Doses and Conclusions

Plant Hatch Airborne Pathway Dose

Location		**RASCAL Software**	**HOTSPOT Software**
		Gaseous Pathway Dose (mrem)	Gaseous Pathway Dose (mrem)
WSTSF	0.3 miles	1.3E-3	5.1E-3
SSF	0.5 miles	8.0E-4	1.7E-3

Plant Hatch Liquid Pathway Dose

Location	Liquid Pathway Dose (mrem)
Altamaha River	5.9E-3

By ensuring that the highest source term shipping container used at the Hatch Site is kept within the dose limits established for transportation, a fire that consumes the contents will not cause an instantaneous dose at the site boundary in excess of the 10 CFR 20 limit of 2 mrem during any one hour for either airborne or liquid pathway releases.

NFPA 805 HP RAI 02

To meet the radioactive release performance criteria for NFPA 805, licensees must demonstrate that radiation released to any unrestricted area due to the direct effects of fire suppression activities remains as low as is reasonably achievable (ALARA), not to exceed the limits in 10 CFR Part 20.

The NRC staff noted that the licensee has performed a bounding analyses to demonstrate that the doses from the airborne and liquid pathways resulting from fire suppression activities will not exceed the limits of 10 CFR Part 20. In the licensee's analysis, the calculated bounding doses are provided in terms of total effective dose equivalent (TEDE), which is consistent with the limits specified in 10 CFR Part 20. The limits in 10 CFR Part 20 are specified in terms of TEDE because the regulations in 10 CFR Part 20 are based on the International Commission on Radiation Protection's (ICRP) recommendations in ICRP Reports 26 and 30. However, when using Radiological Assessment System for Consequence Analysis (RASCAL) to perform the bounding analysis, the licensee chose to use ICRP 60/72 inhalation dose coefficients. In addition, when HotSpot Version 3.0.2 was used, the licensee selected the Federal Guidance Report (FGR) No. 13 dose conversion factors (DCFs). As a result, both the RASCAL and HotSpot calculations provided doses in terms of total effective dose (TED). Likewise, the use of the ICRP 60/119 DCFs for the liquid pathway calculations also resulted in doses in terms of TED. Nevertheless, the results provided in the licensee's conclusions were provided in terms of TEDE. While both TEDE and TED calculate dose for external and internal exposure, the underlying dosimetry models used to develop the DCFs are not the same. The DCFs selected for the gaseous and liquid bounding analyses results in the use of dosimetry models and DCFs that differ from those used in ICRP Reports 26 and 30. Dose conversion factors acceptable to the NRC staff are derived from data and methodologies provide in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" and can be found in FGR No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," and FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," for exposure to radionuclides in air, water, and soil.

Please provide a summary explaining why the use of the TED DCFs is acceptable, even though the dose limits in 10 CFR Part 20 are specified in terms of TEDE.

SNC Response

The bounding analyses to demonstrate that the doses from the airborne and liquid pathways resulting from fire suppression activities will not exceed the limits of 10 CFR Part 20 were revised to use the dose conversion factors from ICRP Reports 26 and 30 contained in FGR Reports No. 11 and No. 12. All doses are reported in terms of Total Effective Dose Equivalent (TEDE), and results demonstrate doses remain a small fraction of the applicable 10CFR20 limit. See response to RAI-HP-01 for summary of results.

Edwin I. Hatch Nuclear Plant Units – 1 and 2

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 Probabilistic Risk Assessment (PRA) RAIs

NFPA 805 Probabilistic Risk Assessment (PRA) RAI 01 – Fire PRA F&O Closure Review Process

NFPA 805 Section 2.4.3.3 states that the PRA approach, methods, and data shall be acceptable to the NRC. Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, (ADAMS Accession No. ML090410014), describes a peer review process utilizing an associated American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard (currently ASME/ANS-RA-Sa-2009, "Addenda to ASME/ANS RA-S- 2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Februrary 2, 2009) as one acceptable approach for determining the technical adequacy of the PRA once acceptable consensus approaches or models have been established. The primary results of a peer review are the Facts and Observations (F&Os) recorded by the peer review and the subsequent resolution of these F&Os. In a letter dated May 3, 2017 (ADAMS Accession No. ML17079A427) the NRC staff has accepted, with conditions, a final version of Appendix X to Nuclear Energy Institute (NEI) 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," and NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," (ADAMS Accession No. ML17086A431), which defines a review process for closing finding-level F&Os.

LAR Attachment U and LAR Attachment V state that an F&O independent assessment (IA) was performed on the FPRA peer review results to close finding-level F&Os using the process documented in Appendix X to NEI 07-12.

Based on the information provided in the LAR, the NRC staff was unable to determine if the F&O closure reviews were performed fully consistent with the NRC accepted process described above, therefore, the NRC staff requests that the licensee:

(a) Describe the process used to determine whether a change to the PRA is maintenance or an upgrade. Describe the actions taken or internal processes applied to ensure the robustness of your determination.

(b) Confirm, for each FPRA F&O resolved, whether the resolution was determined to be a PRA upgrade or maintenance update. Include discussion of how the guidance in Appendix 1-A of ASME/ANS RA-Sa-2009 was used in the basis of each determination. Discuss any changes made to your initial assignment of maintenance or upgrade through deliberations with the IA team.

(c) If the request in part (b) above cannot be confirmed based on the current F&O closure review documentation, then provide for each finding-level F&O an indication of whether the resolution was determined to be a PRA upgrade or maintenance update along with the specific bases for those determinations as reviewed by the independent assessment (IA) team.

SNC Response

A full scope peer review for the Fire PRA was performed in May 2016. A total of 61 Finding level F&Os were issued.

An Independent Assessment (IA) Team reviewed disposition of 61 Finding level F&O in October 2017. The IA Team was provided with a characterization of each Finding level F&O resolution as a PRA upgrade or maintenance. The characterization was performed using guidance provided in RIE Department Procedure RIE-001 "Generation and Maintenance of Probabilistic Risk Assessment Models and Associated Updates". Section 2.16.h of the RIE procedure defines what constitutes a model maintenance or model upgrade. This is consistent with the ASME/ANS guidance and used by the RIE department for several years. The RIE procedure states:

"A change defined as a PRA Upgrade by ASME/ANS PRA standard definitions section. Changes classified as PRA Upgrades require peer review of the portions of the standard impacted are:

1. *Incorporation of a new calculation or evaluation methods Not previously used in the specific baseline or application model.*
2. *A change in model scope that impacts the significant accident sequences (sequences that are in the top 90% of CDF or that individually contribute greater than 1% of overall risk) or the significant accident progression sequences (sequences that are in the top 90% LERF or that individually contribute greater that n 1% of overall risk).*
3. *A change in model capability that impacts the significant accident sequences or the significant accident progress sequences. "*

This process has been used several times by Southern Nuclear Company for various models changes and self-assessment of internally dispositioned Finding level F&Os prior to the Independent Assessment (IA) for Appendix "X" closure team review of dispositioned F&Os. Feedback and lesson learned are incorporated by revising the said procedure.

The IA Team determined that disposition of all 61 Finding level F&Os was satisfactory; therefore, all findings were closed out. The IA Team also determined that resolution of one Finding level F&O (201-16) constituted PRA Upgrade. As a result, a concurrent focused scope peer review was performed to review a method that calculated time to cable damage due to exposure of a fire environment. The IA Team determined that the method was technically sound and provided a reasonable and realistic method for estimating time to cable damage due to exposure of a fire environment. No additional F&O were issued because of the focused scope peer review.

NFPA 805 PRA RAI 02 – Incorporation of Internal Events PRA Updates into the FPRA

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. It appears to the NRC staff that a number of internal event (IEPRA) model additions and revisions were made to resolve F&Os that were then closed in the April 2017 IE F&O closure review. The NRC staff notes that the F&O closure review for the IEPRA was followed closely by a FPRA F&O closure review in October/November 2017. It appears to the NRC staff that a number of updates were made to the IE PRA to resolve F&Os that are relevant to the FPRA's underlying plant response model.

Therefore, the NRC staff requests that the licensee:

(a) Confirm that applicable IE PRA model updates that were performed to resolve finding- level F&Os ahead of the IE F&O closure review were also performed for the FPRA model used to determine the fire risk estimates for the NFPA 805 LAR.

(b) If the IEPRA model updates that were performed to resolve finding-level F&Os were not also performed for the FPRA model used to determine the fire risk estimates for the NFPA 805 LAR, then justify that these model updates have no impact on NFPA 805 LAR application. Alternatively, perform these updates for the integrated analysis provided in response PRA RAI 03.

SNC Response

(a) With the exception of Internal Flooding F&Os, applicable IEPRA model updates that were performed to resolve finding-level F&Os before the IEPRA F&O closure review were also performed for the Fire PRA model used to determine the fire risk estimates for the NFPA 805 LAR. Therefore, the Fire PRA model includes applicable IEPRA model updates resulting from the finding-level IEPRA F&Os.

Four finding-level IEPRA F&Os were considered open or partially closed per the NEI Appendix X process. They have been assessed to have no impact on the NFPA 805 application and mentioned in the LAR.

(b) N/A

<u>NFPA 805 PRA RAI 03 – Integrated Analysis – This RAI response will be provided with supplemental correspondence.</u>

NFPA 805 PRA RAI 04 - Use of Unacceptable Methods

LAR Attachment V states that the Hatch full-scope-scope FPRA peer review identified "0 unreviewed analysis methods (UAMs)". Though UAMs, as evaluated by the EPRI/NRC panel were not used, this does not preclude the possibility that methods may have been used in the FPRA that deviate from guidance in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (ADAMS Accession Nos. ML052580075, ML052580118, and ML103090242), or other acceptable guidance (e.g., frequently asked questions (FAQs), NUREGs, or interim guidance documents). Based on the information provided in the LAR, the NRC staff could not determine whether any methods that deviate from NUREG/CR-6850 or other acceptable guidance were used, therefore the NRC staff requests that the licensee:

(a) Identify methods used in the FPRA that deviate from guidance in NUREG/CR-6850 or other acceptable guidance.

(b) If such deviations exist, then justify their use in the FPRA. Alternatively, replace those methods with a method acceptable to NRC in the integrated analysis performed in response to PRA RAI 03. Include a description of the replacement method along with justification that it is consistent with NRC accepted guidance.

SNC Response

(a) There were no methods used in the development of the Hatch Fire PRA that deviated from the guidance provided in NUREG/CR-6850 or other NRC acceptable guidance.

(b) N/A

NFPA 805 PRA RAI 05 – Implementation Item to Update Fire PRA When Modification are Complete

LAR Attachment S, Table S-3, presents an implementation item (i.e., IMP-19) to update the FPRA after all plant modifications have been implemented to reflect the as-built, as-operated plant.

This implementation item does not indicate HNP's plan in the event that the updated FPRA results do not meet RG 1.174, Revision 2, risk acceptance guidelines. Also, implementation item IMP-19 does not indicate that updates to the FPRA should include adjustments needed to reflect completion of other implementation items such as update of fire response procedures.

Revise implementation item IMP-19 to include an action to update the FPRA following completion of modifications and implementation items and include a plan of action should the updated as-built as-operated FPRA results risk estimates exceed RG 1.174, Revision 2, risk acceptance guidelines (e.g., this plan could include refining the analytic risk estimates or performing additional modifications to the plant).

SNC Response

As stated in LAR Attachment S, Table S-3, implementation item 19 was identified to address the need for updating the Fire PRA once all plant modifications have been installed as described in LAR Attachment S, Table S-2. If, through the incorporation of the plant modifications and updating the model to the as-built as-operated condition, the Fire PRA is no longer within the RG 1.174, Revision 2, risk acceptance guidelines, measures will be taken to reduce the Fire PRA risk. Examples of these measures may include: Additional model refinements or new plant modifications. If any of the measures taken include the use of new PRA methods, that have not been Peer Reviewed for use in the Hatch Fire PRA model, a focused scope Peer Review will be held.

The updated text of IMP-19 is as follows:

> "Following installation of modifications, implementation items, and the as-built installation details, the Fire PRA will be updated to reflect the as-built, as-operated plant. If the updated Fire PRA does not meet the risk acceptance guidelines found in RG 1.174 Revision 2, measures will be taken to reduce the Fire PRA risk. Measures may include approaches such as plant modifications or analytical updates to the PRA model. If any of the measures taken include the use of an upgrade, that have not been Peer Reviewed for use in the Hatch Fire PRA model, a focused scope Peer Review will be conducted."

NFPA 805 PRA RAI 06 – Reduced Transient Heat Release Rates (HRRs)

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA. The key factors used to justify using transient fire reduced heat release rates (HRRs) below those prescribed in NUREG/CR-6850 are discussed in the June 21, 2012, letter from Joseph Giitter, U.S. Nuclear Regulatory Commission, to Biff Bradley, NEI, "Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires," (ADAMS Package Accession No. ML12172A406).

The LAR and detailed FM analysis indicate that although a bounding 98% HRR of 317 kW from NUREG/CR-6850 was typically used, a reduced transient fire HRR seems to have been applied as part of detailed FM for certain fire areas (e.g., the CSR, Intake Structure, and East Cableway).

Discuss the key factors used to justify the reduced rate below 317 kW. Include in this discussion:

(a) Identification of the fire areas where a reduced transient fire HRR is credited and what reduced HRR value was applied.

(b) A description for each location where a reduced HRR is credited, and a description of the administrative controls that justify the reduced HRR including how location-specific attributes and considerations are addressed. Include a discussion of the required controls for ignition sources in these locations along and the types and quantities of combustible materials needed to perform maintenance. Also, include discussion of the personnel traffic that would be expected through each location.

(c) The results of a review of records related to compliance with the transient combustible and hot work controls.

SNC Response

(a) Table 4-4 of the Hatch Fire Modeling Calculation identifies the fire zones where a reduced transient fire HRR is credited. A value of 65 kW was used with the fire zones identified in Table 4-4.

(b) Each of the four fire zones that credit a reduced transient fire HRR are classified as level A transient combustible-controlled areas. Per NMP-ES-035-014, SNC Fleet Transient Combustible Controls, these are areas where unattended combustible material are not allowed except for short breaks up to one hour. Three of the fire zones, 0501: Intake Structure, 1104: Unit 1 East Cableway, and 2104: Unit 2 East Cableway require a continuous fire watch when combustibles are left in these areas unattended. Two of the spaces, 0024A: Cable Spreading Room and 0501 have posted signage at the entrance, near the card access control box required to enter the zone, identifying that storage of combustible materials in the area are prohibited. Fire zones 1104 and 2104 do not have card access controls, but have floors painted with large red arrows, clearly indicating that no storage is allowed within the fire zone.

Since transient combustibles are strictly controlled, any temporary storage of transient materials for maintenance requires special permitting from the work control center.

Permitting requires controls on type, use and removal of transient materials as well as any compensatory measures that need to be enacted while the maintenance occurs. This enables proper notification and alert to the work control center of additional hazards and the duration for which this hazard exists to enable a prompt response in the event of an incident during maintenance.

With exception of the Cable Spread Room, each area has been classified with an occupancy transient influencing factor rating of medium – a compartment that is not continuously occupied but has regular foot traffic per FAQ 12-0064. The Cable Spreading Room has a low rating – a compartment with low foot traffic or out of the general traffic path.

(c) Procedure NMP-ES-035-014, SNC Fleet Transient Combustible Controls, was implemented 1/24/2017. A CR search was performed for the date range 1/24/17 through 3/18/19 and the following CR's were found for the areas of consideration related to transient combustible compliance and hotwork.

CR	Fire Zone	Date	Description and Location	Discussion
10591906	0024A	3/15/2019	Transient combustible 2x4 identified during walkdown in CSR during walkdown	A photograph of the wood material suggests it is a 2"x4", placed between two trays. This would limit its length to that of the separation between trays – approximately 0.5 m (1.5 ft). With an average heat release rate per unit area (HRRPUA) for wood of approximately 154 kW/m^2 (Ref. 1), the total HRR for the observed transient would be 23.1 kW – well below the reduced transient value of 65 kW. Assuming the wood was fire retardant treated plywood, this value would be reduced to approximately 11 kW.
10518914	0501	7/25/2018	Yellow scaffold tags tie wrapped to scaffolds in Intake Structure	These scaffolding tags and tie wraps represent a minor violation as they were likely not located close to an ignition source being attached to scaffolding. Additionally, two tags and any associated tie wraps would not likely exceed an HRR of 65 kW. Reviewing transient ignition sources in Table G-7 of NUREG/CR-6850, a number involve small amounts of paper and some plastic materials each of which involved quantities likely in excess of the items associated with this CR. Each of these experiments results in HRRs of less than the 65-kW value used for this fire zone.

CR	Fire Zone	Date	Description and Location	Discussion
10541361	0501	10/01/2018	10 Nylon wire ties, PVC nipple, and small plastic bag with wire seals and crimp found in hose cabinet in the Intake Structure	These items represent a minor violation as they were located within the hose cabinet, away from any potential ignition source.
10520707	0501	07/31/2018	Combustible material found stored in hose cabinet and removed from the Intake Structure.	These items represent a minor violation as they were located within the hose cabinet, away from any potential ignition source.

References

1. Brenden, John J. Measurements of Heat Release Rates on Wood Products and an Assembly. No. FSRP-FPL-281. FOREST PRODUCTS LAB MADISON WIS, 1977.

NFPA 805 PRA RAI 07 – Sensitive Electronics

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," Revision 2, (ADAMS Accession No. ML081130188), as providing methods acceptable to the NRC staff for adopting a FPP consistent with NFPA 805. In a letter dated July 12, 2006, to NEI (ADAMS Accession No. ML061660105), the NRC established the FAQ process where official agency positions regarding acceptable methods can be documented until they can be included in revisions to RG 1.205 or NEI 04-02. Methods that have not been determined to be acceptable by the NRC staff or acceptable methods that appear to have been applied differently than described require additional justification to allow the NRC staff to complete its review of the proposed method.

Though LAR Attachment H refers to FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics" (ADAMS Accession No. ML13322A085), the fire scenario development and detailed FM indicates that guidance from FAQ 13-0004 was not used. For example, it appears that for sensitive electronics enclosed in electrical cabinets that inspection and walkdowns of cabinet configurations as recommended by the guidance in FAQ 13-0004 were not performed. However, it appears that a sensitivity study on sensitive electronics may have been performed.
Still, the LAR does not describe a sensitivity study performed for sensitive electronics or present the quantitative results of such a study, and the study does not appear to be included as part of the FPRA uncertainty analysis. In light of these observations:

(a) Describe the treatment of sensitive electronics for the FPRA and explain whether it is consistent with the guidance in FAQ 13-0004, including the caveats about configurations that can invalidate the approach (i.e., sensitive electronics mounted on the surface of cabinets and the presence of louver or vents).

(b) If the approach cannot be justified to be consistent with FAQ 13-0004, then justify that the treatment of sensitive electronics has no impact on the NFPA 805 application. Alternatively, replace the current approach with an acceptable approach in the integrated analysis performed in response to PRA RAI 03.

SNC Response

(a) Consistent with the guidance in FAQ 13-0004, externally mounted sensitive electronics were treated with the lower damage threshold in the Fire PRA using a heat flux of 3kW/m2. Externally mounted sensitive electronics were identified during Fire PRA walkdowns.

However, the guidance in FAQ 13-0004 is not used explicitly to credit and model internally mounted sensitive electronics. Electrical cabinets are assumed damaged based on the applicable cable damage heat flux (11 kW/m2 for Thermoset cables and 6kW/m2 for Thermoplastic cables). The temperature criterion of 65°C for internally mounted sensitive electronics was accounted for using a screening approach. A screening approach was used to preclude internally inspecting each electrical cabinet. The plant equipment, raceway, cable database was used to identify the electrical cabinets with Fire PRA cables. Then, electrical cabinets were excluded based on the function of the electrical cabinet. For example, Switchgear, MCCs, and distribution cabinets were excluded. The remaining electrical cabinets were reviewed for risk and dispositioned based on the plant location and the fire modeling included in the Fire PRA. These included locations in which:

- A bounding room scenario is included with no development of non-suppression probabilities,
- The location is in a large open area where temperature increase is not a concern (e.g., general areas of the Reactor Building and Turbine Building),
- Fire risk is already bounded by the fire initiating event treatment (e.g., loss of the control panel is bounded by assumed plant trip).

This screening process left a small number of locations to evaluate for potential fire risk changes. The remaining areas included the Control Building working floor (0001) and the essential switchgear rooms (1404, 1408, 1412, 2404, 2408, 2409).

A sensitivity analysis was performed in these areas. For area 0001 the sensitivity analysis resulted in no noticeable change in CDF. The potential electrical cabinets identified are related to the air compressors and balance of plant annunciators; given a plant trip is assumed the results of the sensitivity analysis are expected.

For the essential switchgear rooms (areas 1404, 1408, 1412, 2404, 2408, and 2409) most propagating fires result in the loss of the applicable switchgear. Therefore, only postulated fires with a CCDP less than that equivalent to the loss of the switchgear were evaluated in the sensitivity analysis. Including the failure of the potential electrical cabinets in these scenarios resulted in an increase in CDF of less than 1E-8/yr.

Given the screening approach using the 65°C temperature criterion resulted in a small increase in CDF, the explicit treatment of enclosed sensitive electronics is either bounded by the existing fire modeling assumptions, the assumed turbine trip initiator, or has been evaluated in a sensitivity study to be negligible increase in risk if enclosed sensitive electronics were in the electrical cabinet.

(b) N/A

NFPA 805 PRA RAI 08 – Consideration of Violations in Determining Hot Work/Transient Fire Frequency Influence Factors

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," Revision 2, (ADAMS Accession No. ML081130188), as providing methods acceptable to the NRC staff for adopting a FPP consistent with NFPA 805. In a letter dated July 12, 2006, to NEI (ADAMS Accession No. ML061660105), the NRC established the FAQ process where official agency positions regarding acceptable methods can be documented until they can be included in revisions to RG 1.205 or NEI 04-02. FAQ 12-0064 provides guidance on determining hot work/transient fire frequency influence factors (ADAMS Accession No. ML12346A488). Methods that have not been determined to be acceptable by the NRC staff or acceptable methods that appear to have been applied differently than described require additional justification to allow the NRC staff to complete its review of the proposed method.

During the audit, the licensee presented a Condition Report (CR) where a violation was self-identified in the CSR that could affect the influence factors assigned to the CSR. Violations of transient combustible controls play a role in the assignment of influence factors. As a result, the staff is requesting that the licensee perform a review of CRs for all violations in Units 1 and 2, and evaluate the impact of these violations on their assignment of influence factors for Units 1 and 2. Should changes be made in the influence factors, update the PRA as needed, and incorporate that update into the response for PRA RAI 3 if needed.

SNC Response

A CR search was performed using different key words: transient, combustible, housekeeping, hot work, and improper for the date range 1/24/17 through 3/18/19. As shown in the table provided within Attachment 1, many of the condition reports (CRs) have occurred in areas with Medium or High ratings assigned to the maintenance and storage influencing factors, which indicates that some storage of combustibles is allowed.

The following fire zones, with rankings of Very Low, or No, have CRs that occurred during a refueling outage:

- 1201: U1 Drywell and Torus
- 2101A: Under Main Condenser
- 2101K: Main Condenser Area
- 2201: U2 Drywell and Torus.

The transient weighting factors described in NUREG/CR-6850 were developed for at power operations and should not influence the weightings used in the Hatch Fire PRA. The reports suggest the CR process is working as items found during refueling are being found and removed.

CRs from other fire zones, with Storage and Maintenance rankings of Very Low and Low, are dispositioned as follows:

- 0024A: Cable Spreading Room – At least one event has occurred in the past five years. A photograph of the wood material identified in the CR suggests is it was 2"x4", placed between two trays. This would limit its length to that of the separation between trays – approximately 0.5 m (1.5 ft). With an average heat release rate per unit area (HRRPUA) for wood of approximately 154 kW/m^2 (Ref. 1), the total HRR for the observed transient would be 23.1 kW – well below the reduced transient value of 65 kW. Assuming the wood was fire retardant treated plywood, this value would be reduced to approximately 11 kW. Additionally, the CR notes that the wood was found at an elevation of approximately 10-12 ft above the floor. The fire modeling in the CSR assumes transient fires would occur near the floor. This assumption results in the ignition and subsequent propagation of trays above the fire. Wood is not an ignition source, so, the only impact would be the additional HRR (at most 23 kW) added to a fire that would ignite the nearby cable trays. The quantity of wood is insigificant compared to the existing combustible loading of the cable trays. Finally, in 2017 the combustible controls associated with the CSR were increased and it was designated a Transient Combustible Control Level A Area (Ref. 2). This designation requires that combustible material not be left unattended, except for short breaks for up to an hour. In the HNP Fire PRA, the CSR is rated with a Very Low storage rating. Per, FAQ 12-0064, this rating is to be applied to areas where long term and temporary storage of combustible materials are prohibited by administrative controls. This rating also requires no violations of these administrative controls be observed for a reasonable period (suggested 5 years). This observed 2"x4" does not represent a violation of long term or temporary storage as it was not an intentional storage of material. Instead the wood, was likely inadvertently left behind after the completion of maintenance work. The Very Low maintenance rating, specifically called out in FAQ 12-0064 as appropriate for the CSR, requires 1) strict access controls and 2) no plant equipment or components other than cables, fire detectors, and junction boxes. The HNP CSR meets these requirements. Therefore, the Very Low ratings in storage and maintenance as applied to fire zone 0024A: Cable Spreading Room is appropriate.

- 2205B: SE Corner Pump Room – The CR identified a full trash that was corrected by Radiation Protection. This fire zone has a low rating assigned for the Storage influencing factor. Per FAQ 12-0064, a Low Storage rating is to be used for an area where no combustible/flammable materials are stored by practice but where combustibles may be introduced subject to a permitting process.

- 1105: East Cableway Foyer – A single event was found to be associated with this area. This fire zone has a low rating assigned for the Maintenance, Storage, and Hotwork influencing factors. Per FAQ 12-0064, a Low Storage rating is to be used for an area where no combustible/flammable materials are stored by practice but where combustibles may be introduced subject to a permitting process. The CR identifies that items found, gloves and small pieces of trash, were removed because by practice they should not have been stored in the fire zone.

- 2103: SE Stairwell – On one occasion some combustible material was found within the stairwell. This fire zone has a low rating assigned for the Maintenance, Storage, and Hotwork influencing factors. Per FAQ 12-0064, a Low Storage rating is to be used for an area where no combustible/flammable materials are stored by practice but where combustibles may be introduced subject to a permitting process. The CR indicates that material was removed, and the workers were coached on the fleet transient combustible controls.

NFPA 805 PRA RAI 09 – Minimum Joint Human Error Probability

NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report," (ADAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple human failure events (HFEs) in human reliability analyses (HRAs). NUREG-1921 refers to Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," (ADAMS Accession No. ML051160213), which recommends that joint human error probability (HEP) values should not be below 1E-5. Table 4-4 of Electrical Power Research Institute (EPRI) 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Therefore, the guidance in NUREG-1921 allows for assigning joint HEPs that are less than 1E-5, but only through assigning proper levels of dependency.

The FPRA uncertainty analysis appears to include a sensitivity study to evaluate the impact of the minimum joint HEP on the fire risk estimates. The study concludes that the FPRA CDF and LERF are not sensitive to assumptions made about the joint HEP value. However, the results of this sensitivity study and description about how the study was conducted is not on the docket. The LAR does not provide this information and does not explain what minimum joint HEP value is currently assumed in the FPRA. Also, even if the assumed minimum joint HEP values are shown to have no impact on the current FPRA risk estimates, it is not clear to the NRC staff how it will be ensured that the impact remains minimal for future PRA model revisions supporting post-transition changes. In light of these observations:

(a) Explain what minimum joint HEP value was assumed in the FPRA.

(b) If a minimum joint HEP value less than 1E-05 was used in the FPRA, then provide a description of the sensitivity study that was performed and the quantitative results (i.e., CDF, LERF, ΔCDF, and ΔLERF) that justify that the minimum joint HEP value has no impact on the application.

(c) If, in response part (b), if it cannot be justified that the minimum joint HEP value has no impact on the application, then provide the following:

   i. Confirm that each joint HEP value used in the FPRA below 1E-5 includes its own justification that demonstrates the inapplicability of the NUREG-1792 lower value guideline (i.e., using such criteria as the dependency factors identified in NUREG-1921 to assess level of dependence). Provide an estimate of the number of these joint HEP values below 1.0E-5, discuss the range of values, and provide at least two different examples where this justification is applied.

   ii. If joint HEP values used in the FPRA below 1E-5 cannot be justified, set these joint

HEPs to 1E-5 in the integrated analysis provided in response to PRA RAI 03.

(d) If a minimum joint HEP value of less than 1E-05 was used but justified because it has no impact on the FPRA results, then add an implementation item that provides an action to confirm that the impact of joint HEP value continues to have a minimal impact on the FPRA estimates in future FPRA models used for post-transition changes.

SNC Response

As recommended within NUREG-1792, the minimum joint human error probability (JHEP) for the FPRA model will be modified to use a floor value of 1.0E-05. Updated analysis results will be provided within the response to PRA RAI 03. Therefore, responses to (a) – (d) are not applicable.

NFPA 805 PRA RAI 10 – Obstructed Plume Model

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the staff for adopting a FPP consistent with NFPA-805. NUREG-2178, Volume 1 "Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE -FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume," (ADAMS Accession No. ML16110A140)16) contains refined peak HRRs, compared to those presented in NUREG/CR-6850, and guidance on modeling the effect of plume obstruction.

The FM performed in support of the FPRA appears to use the guidance from NUREG-2178, Volume 1, though it is not clear whether guidance on modelling the effect of an obstructed plume was used. NUREG-2178 provides guidance that indicates that the obstructed plume model is not applicable to cabinets in which the fire is assumed to be located at elevations of less than one-half of the cabinet.

If obstructed plume modeling was used, then indicate whether the base of the fire was assumed to be located at an elevation of less than one-half of the cabinet.

Justify any modelling in which the base of an obstructed plume is located at less than one half of the cabinet's height, or remove credit for the obstructed plume model in the integrated analysis provided in response to PRA RAI 03.

SNC Response

The Hatch Fire PRA single compartment detailed fire modeling calculations assumed that the base of the fires in fixed ignition sources occurred at the top of the electrical cabinet. Therefore, there were no instances where credit was taken for obstructed plume modeling where the base of the fire was assumed to be located at an elevation of less than one-half the cabinet height.

NFPA 805 PRA RAI 11 – Treatment of Main Control Room Fires

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the staff for adopting a FPP consistent with NFPA-805. In a letter dated July 12, 2006, to NEI (ADAMS Accession No. ML061660105), the NRC established the FAQ process where official agency positions regarding acceptable methods can be documented until they can be included in revisions to RG 1.205 or NEI 04-02. Methods that have not been determined to be acceptable by the NRC staff or acceptable methods that appear to have been applied differently than described require additional justification to allow the NRC staff to complete its review of the proposed method. FPRA FAQ 14-0008, "Main Control Board Treatment" (ADAMS Accession No. ML14190B307) provides guidance on modeling fires in the MCR.

The MCBs in the MCR appear to consist of a front and rear side connected by a single enclosure with a continuous ceiling. However, the FM performed for the MCBs appears to treat the cabinets behind the MCB 'horseshoe" as separate electrical cabinets instead of treating them as the rear side of the MCB. The guidance in FAQ 14-0008 indicates that if the front and rear side of such a configuration are connected together in an enclosure where "the presence of a MCB cabinet ceiling would connote a single cabinet," then the rear cabinets should "classified as an integral part of the MCB." For this MCB configuration, the guidance in FAQ 14-0008 provides three options for applying Appendix L of NUREG/CR-6850 to address fire progression associated with the MCB. HNP's treatment of the MCB appears to deviate from NRC accepted guidance.

In a separate MCB modelling concern, the NRC staff notes that a damage delay of 15 minutes was credited due to the presence of solid barriers between MCB cabinets. However, it seems that a number of, or all of, the MCR MCBs have open backs (or backs that are open within the large MCB enclosure). NUREG/CR-6850 Section 11.5.2.8 indicates that the approach described in NUREG/CR-6850 Appendix L may be used for "cabinets separated by a single wall with back covers." It is not clear to NRC staff how the presence of solid barriers between MCB cabinet segments can be credited for a 15 minute damage delay for MCB cabinets with open backs.

In light of the observations above, address the following:

(a) Describe the MCB configuration for the MCR and compare its configuration with those elements of FAQ 14-0008. Include discussion of the area between the cabinets that comprise the "MCB horseshoe" and the cabinets on the backside of the "MCB horseshoe" that appear to NRC staff to be part of single MCB enclosure.

(b) Justify that the cabinets behind the MCB horseshoe are not part of single integral MCB enclosure using the definition in FAQ 14-0008.

(c) Describe the mechanisms that were considered in the fire PRA which produced fire damage of targets across the walkway between the MCB and the cabinets just behind them. Include summary of the relevant fire modeling.

(d) If it cannot be shown in response to part (b) above that the cabinets behind the MCB horseshoe are not part of single integral MCB enclosure using the definition in FAQ 14- 0008, then justify treatment of the cabinets on the rear side of the MCB as separate electrical cabinets. Include clarification of how the back side cabinets are modelled and an explanation of how the treatment aligns with NRC accepted guidance.

(e) If in response to parts (c & d) above, the current treatment of the MCB horseshoe and the cabinets behind the MCB horseshoe cannot be justified using NRC accepted guidance, then update the treatment of the MCB enclosure to be consistent with the guidance in FAQ 14-0008 in the integrated analysis provided in response to PRA RAI 03.

(f) Clarify whether the MCB, or whether certain individual cabinets of the MCB, have an open back (or backs that are open within the large MCB enclosure).

(g) If the MCB, or individual cabinets of the MCB, have an open back, then justify the credit taken in the FPRA for a damage delay of 15 minutes due to the presence of solid barriers between MCB cabinets. Include a description of the FM that supports the damage delay assumption of 15 minutes.

(h) If in the response to part (g) above, the credit for a 15 minute delay in damage cannot be justified then update the fire propagation assumptions for MCB cabinets to be consistent with NRC guidance concerning cabinets with open backs in the integrated analysis provided in response to PRA RAI 03.

SNC Response

(a) The MCB configuration for each unit in the MCR consists of bench boards arranged in a horseshoe shape. The Unit 1 MCB consist of nine panels and the Unit 2 MCB consists of seven panels. Each panel is separated from adjacent panels by metal partitions. Each panel is also subdivided into bays which in some, but not all cases are also separated by metal barriers. These panels are considered the front panels. The back side of the front panels are open to a narrow walkway behind the panels. On the far side of the walkway there are additional back panels, also in a horseshoe shape. The majority of these back panels are also open to the walkway. However, the front panels and back panels are not connected to form a single enclosure, there is open space between the top of the front and back panels to the ceiling of the MCR. In addition, the entrances to the walk way do not include solid doors, wire mesh gates are installed to access the walkway.

(b) The FAQ states, "For the rear side of the MCB to be treated as part of the MCB, both the rear and front sides should be connected together as a single enclosure." Given the front and back panels are not connected and do not form a single enclosure, it was determined that the back panels did not satisfy the definition in the FAQ to be classified as MCB.

(c) A fire originating in the back panels is postulated to damage cables in the front panels (MCB) when the heat flux criterion is exceeded using the same fire modeling methods performed in the fire PRA for other electrical cabinets. In addition, transient fires in the walkway are included and are postulated to damage cables in the front (MCB) and back panels. Fires originating in the front panels (MCB) are treated using the guidance of NUREG/CR-6850 Appendix L.

(d) See response to (b). The front (MCB) and back panels are not connected; therefore, the definition in FAQ 14-0008 is not satisfied and the guidance in NUREG/CR-6850 Appendix L is not applied to the back panels. The back-panel fire modeling is consistent with other electrical cabinets in the fire PRA which use the guidance of NUREG/CR-6850 as supplemented with other accepted NRC guidance (i.e., FAQs and subsequent NUREGs).

(e) See response to (d). The guidance in FAQ 14-0008 is used to define the MCB.

(f) See response to parts (a) and (c). The MCB and several of the back panels across the walkway have an open back. The open back configuration is considered in the fire modeling.

(g) When applying the guidance in NUREG/CR-6850 Section 11.5.2.8 and Appendix L to the MCB, a 15-minute delay was not credited given the panels have open backs. However, a 10-minute delay was used to credit the steel walls separating certain bays and panels. The 10-minute delay is consistent with NUREG/CR-6850 Appendix S information for fire spread given a single steel wall barrier.

(h) See response to (f). A 15-minute delay is not used given the open back configuration of the MCBs.

NFPA 805 PRA RAI 12 – MCR Abandonment on Loss of Habitability

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the staff for adopting a FPP consistent with NFPA-805. Methods that have not been determined to be acceptable by the NRC staff or acceptable methods that appear to have been applied differently than described require additional justification to allow the NRC staff to complete its review of the proposed method.

The LAR does not describe how MCR abandonment scenarios due to loss of habitability (LOH) were modelled. NCR staff notes that LAR Attachment W, Table W-2 includes among the top fire CDF contributors three MCR abandonment scenarios due to LOH with Conditional Core Damage Probabilities (CCDPs) ranging as large as 2.5E-01 to 8.18E-01. Nonetheless, it is still not completely clear to the NRC staff how the treatment of MCR abandonment due to LOH addresses the complexity associated with the full range of fire impacts that can occur from fires in the MCR. NRC staff notes that this complexity can present a significant modelling challenge.

In light of the observations above, address the following:

(a) Explain how the CCDPs and conditional large early release probabilities (CLERPs) were estimated for MCR abandonment scenarios due to LOH. Include:

i. Identification of the actions required to execute successful alternate shutdown and how they are modeled in the FPRA, including actions that must be performed before leaving the MCR.

ii. Explanation of how command and control is performed given that Unit 1's Remote Shutdown Panel is divided between four panel locations in four separate fire zones.

iii. Explanation of how the complexity associated with actions performed from multiple panel locations is considered in the HEPs that are used to estimate the CCDP and CLERP.

iv. Discussion of the challenge of maintaining communication between operators at different panels who are coordinating plant control and how this is factored into development of the MRC abandonment HEPs.

(b) Explain how various possible fire-induced failures are addressed in the CCDP and CLERP estimates for fires that lead to abandonment due to loss of habitability. Specifically include in this explanation, a discussion of how the following scenarios are addressed:

i. Scenarios where fire fails only a few functions aside from forcing MCR abandonment and successful alternate shutdown is straightforward;

ii. Scenarios where fire could cause some recoverable functional failures or spurious operations that complicate the shutdown, but successful alternate shutdown is likely; and,

iii. Scenarios where the fire-induced failures cause great difficulty for shutdown by failing multiple functions and/or complex spurious operations that make successful shutdown unlikely.

(c) Provide the range of CCDP and CLERP values for MCR abandonment scenarios due to loss of habitability for the appropriate fire areas for the post-transition plant model. Include explanation for why the range of CCDPs and CLERPS for MCR abandonment scenarios were similar values for both units even though the complexity of the alternate shutdown actions is much greater for Unit1 than for Unit 2.

(d) Provide the range of frequency of MCR abandonment scenarios due to loss of habitability for the post-transition plant cases.

(e) If in the response to part (b) and (c) above, it cannot be justified that the current modelling of MCR abandonment due to LOH addresses the complexity associated with the full range of fire impacts that can occur from fires in the MCR, then replace the current approach with an approach that does address the full range of fire impacts that can occur from fires in the MCR due to LOH in the integrated analysis provided in response to PRA RAI 03.

SNC Response

(a) CCDPs and CLERPs were estimated for MCR abandonment scenarios as described below.

i. Table 5-11 in the Hatch Fire HRA Notebook, partially reproduced below, lists the specific actions modeled as human failure events (HFEs) in the Fire PRA for the MCRAB scenario. For loss of control scenarios, a separate HFE (OPHE-LCC-RSP-F) was defined and quantified for the decision to abandon and establish control at the RSP, which is performed while the operators are working in the EOPs and procedures 34AB-X43-001-1/2, depending on the impacted unit. Note that actions directed to be taken prior to abandoning the MCR are not critical to the modeled scenarios (e.g., manual scram is not required because a boundary condition for the MCRAB scenarios is that successful scram occurs and placing FW in single element control is not required because FW is not credited in the Fire model).

The major actions are to start and control RCIC, depressurize, start and control LPCI, manually start a diesel when RCIC is not running, start shutdown cooling, and start torus cooling. Separate actions were modeled and quantified for Unit 1 vs. Unit 2 actions. The actions are fully integrated into the plant response model rather than being considered as a lumped failure (i.e., the action to start LPCI is only required/used if RCIC has failed and the RPV depressurization function was successful).

Basic Event	**Description**	**HEP**	**Error Factor**
OPHE-LCC-RSP-F	FAILURE TO TRANSFER COMMAND AND CONTROL TO RSP	1.58E-01	5
OPHEDEPRESREMOTE-F1 OPHEDEPRESREMOTE-F1_U2	OPERATORS FAIL TO DEPRESSURIZE FROM RSP (TRANSIENT)	5.26E-02	5
OPHEEDGREMOTE-F OPHEEDGREMOTE-F_U2	OPERATORS FAIL TO MANUALLY START EDG for MCRAB WITHOUT RCIC RUNNING	7.78E-02	5
OPHELPCIREMOTE-F	OPERATORS FAIL TO START AND CONTROL LPCI AT RSP (Unit 1)	1.12E-01	5
OPHELPCIREMOTE-F_U2	OPERATORS FAIL TO START AND CONTROL LPCI AT RSP (Unit 2)	8.79E-02	5
OPHERCICREMOTE-F	OPERATORS FAIL TO START AND CONTROL RCIC AT RSP (UNIT 1)	6.10E-02	5
OPHERCICREMOTE-F_U2	OPERATORS FAIL TO START AND CONTROL RCIC AT RSP (UNIT 2)	5.70E-02	5
OPHESDCREMOTE-F OPHESDCREMOTE-F_U2	OPERATORS FAIL TO START SHUTDOWN COOLING FROM REMOTE SHUTDOWN PANEL	2.92E-02	5
OPHETORUSREMOTE-F OPHETORUSREMOTE-F_U2	OPERATORS FAIL TO START TORUS COOLING FROM REMOTE SHUTDOWN PANEL	3.89E-03	5

ii. The operator actions are developed with consideration of unit specific complexities, which include the fact that Unit 1 Remote Shutdown Panel includes four individual panels in different locations. Each unit has its own Emergency Remote Shutdown procedure, 31RS-OPS-001-1 for Unit 1 and 31RS-OPS-001-2 for Unit 2, which specifies the responsibilities of the operators during the MCRAB scenario and explicitly identifies the tools, equipment and keys needed to complete each specified task. Following MCR abandonment, the shift supervisor for each unit will move to their designated Remote Shutdown Panel (RSP). For Unit 1 this will be 1H21-P173 and for Unit 2 this will be 2C82-P001; these serve as the locus for command and control for the MCRAB scenario. From these locations, the shift supervisors maintain control over progress through the procedures for their respective unit and serve as the coordinator of tasks that are performed at the main RSP as well as those that must occur at the dispersed panels or at specific equipment locations. Once command and control has been shifted to the RSPs, operators can communicate with each other via radios, sound powered phones or the plant paging system.

iii. The HRA conducted a specific evaluation of the Unit 1 and Unit 2 operators involved in the MCRAB scenario, their locations, their tasks, and required communications. A task breakdown was conducted to identify the key actions required, the operators involved in implementing that action, and the location where each operator's task was performed. This task breakdown was developed separately for Unit 1 and Unit 2. These tables provided the qualitative basis for understanding the operator interactions and communications by specifically noting where communication is required between operators at different panels. For example, for Unit 2 Torus Cooling, 3 way communication is needed between operators at panels 106RJR24 and 2H21-1P173 as well as between operators at 087RLR21 and 2C82-P001. In addition, a timeline of operator actions for MCRAB was developed to allow the HRA analyst to visualize the required action timing and the coordination between them.

In order to account for command and control related coordination between the RSP and local actions in the MCRAB HFE quantification, additional timing was added to the cognition time (Tcog) for the EDG start, Shutdown Cooling, and Torus cooling actions. The execution time (Texe) of the LPCI actions is based on Job Performance Measure (JPM) timing from the plant and includes time for coordination with other operators. As described in the response to RAI 12.a.iv, there were some subtasks that were considered to warrant explicit modeling of errors related to performing and coordinating tasks at different panels. Beyond that, the THERP execution analysis inherently reflects some degree of an action's complexity through the process of modeling errors for each critical task (i.e., the HEP for an action with many steps is generally larger than simple actions because there are more steps contributing to the HEP).

iv. Inter-operator communications are required for many actions, whether they are performed in the MCR or between operators at different remote shutdown panels (RSP). The Cause Based Decision Tree method, which is used in the MCRAB HRA, accounts for the potential to incorrectly communicate information between operators in the PcC decision tree ("Misread/Miscommunicate Data") as part of the cognitive error quantification process. Some MCRAB tasks introduce the potential for significant communication errors that are separate from those that are addressed by the Cause Based Decision Tree method because the tasks are performed in the execution phase of the action rather than in the diagnosis/situation assessment phase of the action. Communications between operators that are used to keep track of operator progress in the procedures when the order of completion of the tasks is not critical to the success of the plant control strategy are not considered to be significant sources of failure and are not explicitly modeled. Communications errors may be important in complex coordination activities when there is a significant consequence of a miscommunication, such as when a series of steps must be performed in a proper order and equipment damage/strategy failure will occur if they are not, or potentially for cases in which multiple operators are in communication to perform a dynamic system control task. The following characterize examples of how communications tasks were treated:

- For the case in which the operators must communicate that power alignment to a 4KV bus is complete before command and control relays that information to the LPCI pump operator, the communication task was not modeled because attempting to start the pump before the bus is powered has no significant impact on the equipment.

- For the case in which one operator is using a local indicator to direct another remote operator in a different location to make adjustments to RHRSW flow, it is possible for a communication error to lead to an error in the flow setting that would not be immediately noticed given that the operators may move on to other tasks in the procedure. A communications error was explicitly modeled for this step even though there would likely be enough time to recover from an RHRSW flow setting error.

This is consistent with the draft version of NUREG-1921 Supplement 2, which indicates in Table 4-1 that command and control errors are not required for tasks that do not need to be coordinated, or for which coordination failures would not lead to irreversible consequences. Command and control errors are broader in definition than communication errors, but communications are part of the command and control tasks and similar treatment is considered to be appropriate. As noted in the response to RAI 12.a.iii, however, the time required for communications activities was accounted for in the action timelines even if explicit communications errors were not. While the qualitative issues related to communications tasks were examined in the Hatch analysis, the HRA methodologies commonly used in U.S nuclear industry, including those that are available in the EPRI HRA Calculator, are not well suited to quantifying HEPs for the coordination and communication tasks. THERP is used to evaluate the execution portion of the MCRAB tasks, but THERP does not include failure data for these types of tasks. In order to represent the contributions of the communication and coordination failures in the execution analysis, an error of omission was assigned (a standard step omission from THERP table 20-7b) along with the Basic Error of Commission (from the ASEP methodology) to represent a failure in a non-specific task. The draft revision of NUREG-1921 Supplement 2 provides a quantitative means of accounting for potential command and control errors, which may support future MCRAB evaluations, but at this time, there is not a consensus approach for modeling these types of errors.

(b) The plant response using alternate shutdown due to loss of habitability is fully integrated into the logic model. The model includes the available functions for alternate shutdown and operator actions to establish control. For each scenario for which loss of habitability is postulated the PRA logic transfer the scenario to the LOH model logic. Given the plant response using alternate shutdown is fully integrated into the logic model, each of the scenarios described above are included.

i. Postulated fire scenarios where minimum fire damage aside from requiring MCR abandonment because of loss of habitability result in lower CCDPs. These scenarios would include those that typically only require successful operator actions to establish alternate shutdown. That is, there is no fire damage or spurious operations that challenge success of a function. This fire scenarios are the lower bound of the calculated CCDPs.

ii. Postulated fire scenarios where fire damage may challenge the success of a function because of fire damage results in mid-range CCDPs. These scenarios may result in loss of a single function that can be mitigated once operators successfully take control using alternate shutdown. For example, failure of high-pressure injection can be mitigated by emergency depressurization and use of low pressure injection.

iii. Postulated fire scenarios with damage that challenge alternate shutdown are also included. These postulated scenarios include spurious operation events that may challenge the time available to establish alternate shutdown. For example, the PRA does not model the success of alternate shutdown when a fire may result in the spurious opening of multiple SRVs. These fire scenarios are the upper bound of the calculated CCDPs.

(c) The loss of habitability CCDP values range approximately from 7.0E-2 to 1.0E+0. The CLERP values range approximately from 1.0E-4 to 1.0E-1. The scenarios with a CCDP value of 1.0E+0 include postulated transient fire scenarios in the MCR, as well as postulated instrument panel fires in the MCR. These fires may result in the spurious opening of multiple SRVs. The lower bound ranges of CCDP and CLERP values for Unit 1 are greater than those for Unit 2 which reflect the difference in complexity of performing alternate shutdown between Unit 1 and Unit 2. It is noted that the HRA for the operator actions already account for the complex nature of alternate shutdown in general (e.g., high stress situation). Therefore, different procedural steps does not result in significant differences in the HRA.

LOH	Location	Unit 1	Unit 2
CCDP	MCR	8.0E-2 to 1.0E+0	7.0E-2 to 1.0E+0
CLERP	MCR	1.0E-3 to 1.0E-1	1.0E-4 to 4.0E-2

(d) The plant location loss of habitability CDF values range approximately from 6.7E-6/yr. to 6.9E-6/yr. The LERF values range approximately from 1.8E-7/yr. to 1.9E-7/yr. Similar to the above discussion, the ranges of CDF and LERF values may not reflect the difference in complexity of performing alternate shutdown between Unit 1 and Unit 2. These are for the same reasons discussed above.

LOH Frequency	Location	Unit 1 and Unit 2 LOH Fire Frequency	Unit 1 Fire Risk CDF/LERF (Fire Frequency and CCDP/CLERP)	Unit 2 Fire Risk CDF/LERF (Fire Frequency and CCDP/CLERP)
CDF	MCR	4.2E-5/yr.	6.7E-6/yr.	6.9E-6/yr.
LERF	MCR	4.2E-5/yr.	1.9E-7/yr.	1.8E-7/yr.

(e) See response to parts (a) and (b). The current modeling of MCR abandonment due to loss of habitability and the complexities associated with establishing loss of habitability use the current industry guidance and address the state-of-knowledge of the full range of fire impacts.

NFPA 805 PRA RAI 13 – MCR Abandonment on Loss of Control

The LAR does not describe how MCR abandonment scenarios due to loss of control (LOC) were modelled in the FPRA. LAR Attachment W, Tables W-2 and W-3 do not include among the top contributors to fire CDF MCR abandonment scenarios due to LOC. Based on the information provided, it is not clear to the NRC staff whether the treatment of MCR abandonment due to LOC addresses the complexity associated with the full range of fire impacts that can occur from fires in MCR abandonment areas (which appear to be the MCR, Cable Spreading Room (CSR), and Computer Room). The NRC staff notes that this complexity can present a significant modelling challenge. The LAR does not describe what cues and procedures would be used by operators in an actual fire scenario to trigger the decision to abandon the MCR due to LOC. Accordingly, it is not clear to the NRC staff that the failure of operators to make the decision to abandon the MCR and perform alternate shutdown is modeled in the FPRA.

In light of the observations above, address the following:

(a) Identify those locations in the plant in which fire could lead to LOC for which MCR abandonment and alternate shutdown actions are credited in the FPRA.

(b) Explain how various possible fire-induced failures are addressed in the CCDP and CLERP estimates for fires that lead to MCR abandonment due to LOC. Specifically include in this explanation, a discussion of how the following scenarios are addressed. As a part of this response, indicate if the plant response is fully integrated into the PRA.

i. Scenarios where fire fails only a few functions aside from forcing MCR abandonment and successful alternate shutdown is straightforward;

ii. Scenarios where fire could cause some recoverable functional failures or spurious operations that complicate the shutdown, but successful alternate shutdown is likely; and,

iii. Scenarios where the fire-induced failures cause great difficulty for shutdown by failing multiple functions and/or complex spurious operations that make successful shutdown unlikely.

(c) Identify the range of CCDP and CLERP values for MCR abandonment scenarios for the appropriate fire areas due to LOC for the post-transition models. Identify those scenarios which have a CCDP of 1, or explain why there are no such scenarios.

(d) Provide the range of frequency of MCR abandonment scenarios due to LOC for the appropriate fire areas for the post-transition plant case.

(e) Explain how command and control is performed given that Unit 1's Remote Shutdown Panel is divided between four panels in four separate fire zone locations. Include discussion of the challenges of maintaining communication between operators who must perform actions at the four different panels and how this is factored into development of the HEPs that are used to estimate the CCDP and CLERP.

(f) If in the response to part (b) and (c) above, if it cannot be justified that the current modelling of MCR abandonment due to LOC addresses the complexity associated with the full range of fire impacts that can occur from fires in MCR abandonment areas, then replace the current approach with an approach that does address the full range of fire impacts that can occur from fires in MCR abandonment areas in the integrated analysis provided in response to PRA RAI 03.

(g) Indicate how the decision to abandon the MCR due to LOC is made procedurally by operators. Include discussion of the cues that would trigger the decision to abandon the MCR due to LOC.

(h) Explain how the failure of operators to make the decision to abandon the MCR and perform alternate shutdown actions is modeled in the FPRA. Include in the explanation justification that the modeling is consistent with the guidance in NUREG-1921.

(i) If failure of operators to make the decision to abandon the MCR and perform alternate shutdown is not modeled in the FPRA, then justify that this exclusion does not impact the application. Alternatively, incorporate failure of operators to make the decision to abandon the MCR and perform alternate shutdown in the integrated analysis provided in response to PRA RAI 03 consistent with the guidance in NUREG-1921.

SNC Response

(a) The MCR, computer room, and CSR are the plant locations in which alternate shutdown is used. The PRA credits alternate shutdown because of loss of control for postulated fires in the MCR and CSR.

(b) The plant response using alternate shutdown due to loss of control is fully integrated into the logic model. The model includes the available functions for alternate shutdown and operator actions to establish control, as well as the decision to use alternate shutdown. Each scenario fire risk is calculated and based on the results the decision is made in the PRA to transfer the scenario to the LOC model logic. An approximate CCDP of 0.1 was used to identify those scenarios to transfer. This CCDP was used as it is representative of a situation in which enough functions (e.g., loss of high-pressure injection and emergency depressurization) would be lost by the fire and operators would be challenged in the MCR. Given the plant response using alternate shutdown is fully integrated into the logic model, each of the scenarios described above are included.

    i. Postulated fire scenarios where minimum fire damage aside from requiring MCR abandonment because of loss of control result in lower CCDPs. These scenarios would include those that typically only require successful operator actions to establish alternate shutdown. That is, there is no fire damage or spurious operations that challenge success of a function. This fire scenarios are the lower bound of the calculated CCDPs.

    ii. Postulated fire scenarios where fire damage may challenge the success of a function because of fire damage results in mid-range CCDPs. These scenarios may result in loss of a single function that can be mitigated once operators successfully take control using alternate shutdown. For example, failure of high-pressure injection can be mitigated by emergency depressurization and use of low-pressure injection.

iii. Postulated fire scenarios with damage that challenge alternate shutdown are also included. These postulated scenarios include spurious operation events that may challenge the time available to establish alternate shutdown. For example, the PRA does not model the success of alternate shutdown when a fire may result in the spurious opening of multiple SRVs. This fire scenarios are the upper bound of the calculated CCDPs.

(c) The loss of control CCDP values range approximately from 1.0E-2 to 1.0E+0. The CLERP values range approximately from 1.0E-5 to 1.0E-1. The scenarios with a CCDP value of 1.0E+0 include postulated transient fire scenarios in the MCR and CSR, as well as postulated instrument panel fires in the MCR. These fires may result in the spurious opening of multiple SRVs. The ranges of CCDP and CLERP values may not reflect the difference in complexity of performing alternate shutdown between Unit 1 and Unit 2 because of the HRA for the operator actions already account for the complex nature of alternate shutdown in general (e.g., high stress situation). Therefore, different procedural steps do not result in significant differences in the HRA. Also, spatial difference between the two units offset some of the alternate shutdown differences in complexity. For example, cable routing for Unit 1 and Unit 2 in the plant results in differences in LOC consequences.

LOC	Location	Unit 1	Unit 2
CCDP	CSR	1.0E-2 to 1.0E+0	1.0E-2 to 1.0E+0
	MCR	1.0E-2 to 1.0E+0	1.0E-2 to 1.0E+0
CLERP	CSR	1.0E-5 to 1.0E-1	5.0E-5 to 3.0E-1
	MCR	1.0E-5 to 1.0E-1	5.0E-6 to 2.0E-2

(d) The plant location loss of control CDF values range approximately from 1.0E-6/yr. to 7.0E-6/yr. The LERF values range approximately from 1.0E-7/yr. to 5.0E-7/yr. Similar to the above discussion, the ranges of CDF and LERF values may not reflect the difference in complexity of performing alternate shutdown between Unit 1 and Unit 2. These are for the same reasons discussed above.

LOC Location	Unit 1 LOC Fire Frequency	Unit 1 Fire Risk CDF/LERF (Fire Frequency and CCDP/CLERP)	Unit 2 LOC Fire Frequency	Unit 2 Fire Risk CDF/LERF (Fire Frequency and CCDP/CLERP)
CSR CDF	5.1E-6/yr.	1.5E-6/yr.	1.2E-5/yr.	3.9E-6/yr.
MCR CDF	9.4E-5/yr.	5.6E-6/yr.	2.4E-5/yr.	6.7E-6/yr.
Total CDF	**9.9E-5/yr.**	**7.1E-6/yr.**	**3.6E-5/yr.**	**1.1E-5/yr.**
CSR LERF	5.1E-6/yr.	1.0E-7/yr.	1.2E-5/yr.	4.7E-7/yr.
MCR LERF	9.4E-5/yr.	1.1E-7/yr.	2.4E-5/yr.	7.7E-8/yr.
Total LERF	**9.9E-5/yr.**	**2.1E-7/yr.**	**3.6E-5/yr.**	**5.5E-7/yr.**

(e) Each unit has its own Emergency Remote Shutdown procedure, 31RS-OPS-001-1 for Unit 1 and 31RS-OPS-001-2 for Unit 2, which specifies the responsibilities of the operators during the MCRAB scenario and explicitly identifies the tools, equipment and keys needed to complete each specified task. For loss of control scenarios, a separate human failure event

(HFE) OPHE-LCC-RSP-F was defined and quantified for the decision to abandon and actions taken prior to abandoning the MCR, when the operators will be working in the EOPs and procedures 34AB-X43-001-1/2, depending on the impacted unit.

Following MCR abandonment, the shift supervisor for each unit will move to their designated Remote Shutdown Panel (RSP). For Unit 1 this will be 1H21-P173 and for Unit 2 this will be 2C82-P001; these serve as the locus for command and control for the MCRAB scenario. From these locations, the shift supervisors maintain control over progress through the procedures for their respective unit and serve as the coordinator of tasks that are performed at the main RSP as well as those that must occur at the dispersed panels or at specific equipment locations. Once command and control has been shifted to the RSPs, operators can communicate with each other via radios, sound powered phones or the plant paging system.

The HRA conducted a specific evaluation of the Unit 1 and Unit 2 operators involved in the MCRAB scenario, their locations, their tasks, and required communications. For the worst fire scenario, one in which EDGs are required to be started and all high-pressure injection systems are failed by the fire, a task breakdown was conducted to identify the key actions required, the operators involved in implementing that action, and the location where each operator's task was performed. This task breakdown was developed separately for Unit 1 and Unit 2. These tables provided the qualitative basis for understanding the operator interactions and communications by specifically noting where communication is required between operators at different panels. For example, Note 3 says that for Unit 2 Torus Cooling, 3 way communication is needed between operators at panels 106RJR24 and 2H21-1P173, as well as between operators at 087RLR21 and 2C82-P001. In addition, a timeline of operator actions for MCRAB was developed to allow the HRA to visualize the required action timing and the coordination between them.

In order to account for command and control related coordination between the RSP and local actions in the MCRAB HFE quantification, additional timing was added to the cognition time (Tcog) for the EDG start, Shutdown Cooling, and Torus cooling actions. The execution time (Texe) of the LPCI actions is based on Job Performance Measure (JPM) timing from the plant and includes time for coordination with other operators.

(f) See response to parts (b) and (c). The current modeling of MCR abandonment due to loss of control and the complexities associated with establishing loss of control use the current industry guidance and address the state-of-knowledge of the full range of fire impacts.

(g) HNP fire response procedures identify the specific fire areas that may require control room evacuation or operation of equipment from outside the Control Room due to a major fire. However, MCR evacuation due to LOC is at the discretion of the Shift Supervisor.

The following conditions are evaluated for cues of LOC:

- LOC of containment heat removal capability
- LOC of high-pressure injection and emergency depressurization systems
- LOC of low-pressure injection systems

(h) Detailed guidance for addressing MCR abandonment scenarios is included in NUREG-1921 for Quantification and its Supplement 1 for Qualitative Analysis. Per the guidance in Supplement 1, a separate human failure event (HFE), OPHE-LCC-RSP-F, was defined and quantified for the decision to abandon for loss of control scenarios, and actions taken prior to abandoning the MCR, when the operators will be working in the EOPs and procedures 34AB-X43-001-1/2, depending on the impacted unit. Following MCR abandonment, the shift supervisor for each unit will move to their designated Remote Shutdown Panel (RSP). For Unit 1 this will be 1H21-P173 and for Unit 2 this will be 2C82-P001; these serve as the locus for command and control for the MCRAB scenario.

By reviewing the alternate shutdown procedures and interviewing operators, the Hatch HRA conducted a specific evaluation of the Unit 1 and Unit 2 operators involved in the MCRAB scenario, their locations, their tasks, and required communications. A task breakdown was conducted to identify the key actions required, the operators involved in implementing that action, and the location where each operator's task was performed. This task breakdown was developed separately for Unit 1 and Unit 2 and provided the qualitative basis for understanding the operator interactions and communications by specifically noting where communication is required between operators at different panels.

Table 5-1 in the Hatch Fire HRA Notebook lists the specific actions modeled as human failure events (HFEs) in the FPRA for the MCRAB scenario. NUREG-1921 Supplement 1 discusses the need to review the procedures to identify the major actions and to coordinate with the FPRA to ensure these actions are represented properly in the model. The actions identified for the Hatch MCRAB scenario are to start and control RCIC, depressurize, start and control LPCI, manually start a diesel when RCIC is not running, start shutdown cooling, and start torus cooling. Separate actions were modeled and quantified for Unit 1 vs. Unit 2 actions to reflect any unit-specific differences in procedures, timing and task steps.

Once specified, the HFEs were quantified using the EPRI methods contained in the EPRI HRA Calculator and described in NUREG-1921, Appendix B, namely the Human Cognitive Reliability/Operator Reliability Experiment (HCR/ORE) method for the time-dependent portion of the cognitive actions, the Cause Based Decision Tree Method (CBDTM) for cognitive procedure following and information processing actions, and the Technique for Human Error Reliability Prediction (THERP) for the physical manipulation (execution) actions. The HRA Calculator allows the user to input specific information for each HFE regarding cues, procedure steps, timing (including breakdowns into time available vs. time required), and Performance Shaping Factors including workload and stress.

In order to account for command and control related coordination between the RSP and local actions in the MCRAB HFE quantification, additional timing was added to the cognition time (Tcog) for the EDG start, Shutdown Cooling, and Torus cooling actions. The execution time (Texe) of the LPCI actions is based on Job Performance Measure (JPM) timing from the plant and includes time for coordination with other operators. This is consistent with guidance in the NUREG-1921 Supplement 1 for considering command and control aspects in MCRAB HFEs and draft guidance in the forthcoming Supplement 2 for quantifying these aspects.

(i) See above responses. The current modeling for the operator action to make the decision for MCR abandonment due to loss of control and the complexities associated with establishing loss of control use the current industry guidance and address the state-of-knowledge using NUREG-1921.

NFPA 805 PRA RAI 14 – PRA Treatment of Dependencies between Units 1 and 2

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current FPP to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. Section C of RG 1.174 states that PRAs supporting risk-informed applications should be "based on the as-built, as-operated and maintained plant."

The LAR indicates that Unit 1 and 2 are adjoined and it makes reference to common areas, a cross-tie (e.g., RHRSW to RHR cross-tie) and systems shared between units (e.g., the Diesel Generator 1B). LAR Attachment W shows contribution by fire area for CDF, LERF, Δ CDF, and Δ LERF, but does not explain how the risk contribution from fires originating in one unit is addressed for impacts to the other unit given the physical proximity of the other unit, common areas, and the existence of shared systems. Therefore, address the following:

(a) Explain how the risk contribution of fires originating in one unit is addressed for the other unit given impacts due to the physical proximity of equipment and cables in one unit to equipment and cables in the other unit. Include identification of locations where fire in one unit can affect components in the other unit and explain how the risk contributions of such scenarios are allocated in LAR Attachment W, Tables W-4 and W-5.

(b) Explain how the contributions of fires in common areas are addressed, including the risk contribution of fires that can impact components in both units.

(c) Explain the extent to which systems are shared by both units and whether shared systems are credited in the PRA models for both units. If shared systems are credited in the PRA models for each unit, then explain how the PRAs address the possibility that a shared system is demanded in both units in response to a single IE or fire initiator.

SNC Response

(a) Fire scenarios were developed for all unscreened PUAs for the Hatch Fire PRA. Each scenario is quantified for each Unit, regardless of where the ignition source is physically located. For example, a fire in the Unit 1 cable spreading room is still quantified for Unit 2. Therefore, the risk of a fire in the Unit 1 cable spreading room is still included in the Unit 2 results. This is true for common areas as well, or those areas where both Unit 1 and Unit 2 cables or equipment exist. For example, the main control room is considered to be a shared PAU, and each fire scenario (regardless of the unit it is associated with) is quantified for both Units. The risk results presented in LAR Attachment W include all scenarios postulated for the given PAU.

(b) See response in part (a)

(c) There are common systems (main control room air conditioning, intake structure ventilation) and shared (1B diesel generator).

Common systems are in both unit PRA models. That is, the Unit 2 PRA models has Unit 1 power supplies and dependencies in the model and vice versa. The common systems are not mitigating equipment, but they cause initiating events if failed in the PRA. Opposite unit fire impacts are included given the PRA has both Unit 1 and Unit 2 fire scenarios in the models.

The 1B diesel generator is the only shared system between units. A selector switch controls which unit the 1B diesel would go to automatically if an initiator happens on both units at the same time. The PRA model also includes an operator action to manually re-align the 1B diesel generator to the unit that needs it most. Thus, shared systems are only partially credited in each hazard model (this applies to all hazards). In the internal events and internal flooding PRA's, a loss of offsite power is assumed to happen on both units at once. The 1B diesel generator is normally aligned to Unit 1; therefore, the Unit 2 Fire PRA has a lot of operator actions to swap the diesel to Unit 2 that the Unit 1 Fire PRA does not. This results in the 1B diesel generator not being as significant in the Unit 2 models.

NFPA 805 PRA RAI 15 - Calculation of the Change in Risk

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current FPP to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of RI changes. LAR Attachment W, Section W.2.1 provides a general description of how the change-in-risk associated with variances from deterministic requirements (VFDRs) is determined, including discussion about setting fire-induced failures events to "false" in the FPRA model as a way to "mimic" the compliant plant condition. Based on the information provided in the LAR, the NRC staff was unable to fully understand how the change in risk is calculated, therefore, the NRC staff requests that the licensee:

(a) Describe the kinds of model adjustments (if there is more than one type) made to remove different types of VFDRs from the compliant plant model, such as adding events or logic, or the use of surrogate events. Clarify whether the approach used is consistent with guidance in FAQ 08-0054, "Demonstrating Compliance with Chapter 4 of NFPA 805" (ADAMS Accession No. ML15016A280 and associated references therein). In addition, identify any major changes made to the FPRA models or data for the purpose of evaluating VFDRs.

(b) Because the determination of the change-in-risk for MCR abandonment scenarios can be more complex than for other scenarios in the FPRA:

   i. Describe the model adjustments that were made to remove the VFDRs to create the compliant plant model for MCR abandonment scenarios due to both LOH and LOC.

   ii. Describe the criteria used to identify Primary Control Station (PCS) locations.

iii. Explain whether VFDRs were identified differently for fire areas in which MCR abandonment (alternate shutdown) may be required compared to fire areas where MCR abandonment would not occur. If VFDRs were identified differently for MCR abandonment scenarios compared to other areas of the plant, then describe that difference.

iv. If assumptions were made, specific to MCR abandonment scenarios, about modeling the compliant plant (e.g., assumptions about how the CCDP values were determined), then describe and justify those assumptions. As part of the justification, provide an indication of the impact that those assumptions make on the NFPA 805 transition change-in-risk.

(c) Describe the types of VFDRs identified, and discuss whether and how the VFDRs identified but not modeled in the FPRA impact the risk estimates. Describe the qualitative rational for excluding VFDRs from the change-in-risk calculations.

(d) Explain, for both the compliant and transition plant PRA models, whether plant modifications are credited in the model. Clarify whether plant modifications that do not resolve VFDRs are credited in the transition (variant) plant model, but not in the compliant plant model, as a way to reduce risk (i.e., indicative of a "combined change" as discussed in Section 1.1 of RG 1.174). If modifications are credited in the transition plant model to reduce risk but do not resolve a VFDR, then provide the total risk increase associated with unresolved VFDRs and the total risk decrease associated with non-VFDR modifications.

SNC Response

(a) The types of model adjustments made to generate the compliant case by 'removing' different types of VFDRs from the analysis were done using a bounding and a detailed method. The two approaches are described:

Bounding Evaluations

A Bounding Risk Evaluation is used when the initial evaluation achieves acceptable results which are low enough to not warrant detailed evaluation of the fire area delta risk. The threshold is a 'guideline' used to simplify the analysis and the screening process may be expanded or decreased to meet the overall acceptance criteria of RG 1.174. For the Bounding Risk Evaluation, the total area CDF and LERF from the variant model is assumed to be the calculated delta risk for the fire area. This is a conservative estimate as the total area risk bounds the delta risk resulting from the VFDRs. For the Bounding Risk Evaluation, the additional risk of recovery actions is conservatively assumed to be equal to the total delta risk. Risk insights, recovery actions, and plant modifications are summarized and the delta CDF and delta LERF for each fire area are included. No changes are made to the variant model. The compliant cases will assume to have a CDF/LERF equal to zero to model the 'removal' of the VFDRs and all delta risk calculations will be compared to the results of the variant model.

Detailed Fire Risk Evaluations

A Detailed Risk Evaluation is used when the Bounding Risk Evaluation screening criteria above is not met. The Detailed Risk Evaluation requires risk analysis for two plant configurations: the variant model and the deterministically compliant model. The variant condition to be evaluated for risk is defined by the VFDRs from the deterministic analysis. The deterministically compliant condition is a plant configuration that meets section 4.2.3 of NFPA 805. The risk of the compliant case is determined by removing the VFDR failure modes in the Fire PRA. Basic events associated with the VFDRs function (e.g. decay heat removal, or injection) are set to nominal value (via the Fire Impacts Table in the FRANX model) to preclude the fire induced failures for the compliant case calculation. Setting the fire induced failure to nominal will mimic the compliant plant condition modeling the function of the VFDR to be free from fire damage calculating only the random, non-fire impacted failure rate for the credited safe shutdown train for every scenario in the fire area evaluation. Attachment D of SMNH-16-093 identifies those basic events, and the associated VFDR ID, which are added to the Fire Impact Table of the Hatch U1 and U2 baseline FRANX models.

The difference in risk between the variant/baseline risk and calculated compliant cases is the delta risk that is evaluated against the acceptance criteria of RG 1.174.

The methodology used to define the compliant case and calculate the delta risk is consistent with the guidance provided in FAQ 08-0054.

The methodology used to define the compliant case and calculate the delta risk is consistent with the guidance provided in FAQ 08-0054.

(b)

i. For alternate shutdown, fire area 0024 (Main Control Room, Cable Spreading Room, and Computer Room), the delta risk calculations are performed in the same manner as every other fire area using the detailed evaluation approach. However, the compliant case assumes shutdown is being performed at the remote shutdown panels. Therefore, only abandonment scenarios are considered in the delta risk calculation. The base model of record includes remote shutdown logic for LOH and LOC, therefore, the VFDRs are evaluated by setting to their nominal, non-fire probability, at the RSP. The compliant case assumes for every fire that shutdown is performed using the remote shutdown panels.

ii. The criteria used at HNP to identify PCS locations are defined within RG 1.205 and FAQ 07-0030. Specifically, Primary control station(s) is defined in RG 1.205 Section C.2.4 as follows:

> *There are two cases where operator actions taken outside the main control room may be considered as taking place at a primary control station. These two cases involve dedicated shutdown or alternative shutdown controls, which have been reviewed and approved by the NRC. In either case, the location or locations become primary when command and control is shifted from the main control room to these other locations. For these two cases, the operator actions are not considered recovery actions, even if they are necessary to achieve the nuclear safety performance criteria.*
>
> a. *The first case involves the controls for a system or component specifically installed to meet the "dedicated shutdown" option in Section III.G.3 of Appendix R. Operation of this equipment is*

considered as taking place at a primary control station. A system or component that has been specifically installed under the dedicated shutdown concept is a system or component that is operated from a location outside the control room and is fully separated from the fire area where its use is credited. These systems or components cannot be operated from the control room. Operation of dedicated shutdown equipment would not be considered a recovery action, since this would be the primary control station.

b. The second case involves controls for systems and components that have been modified to meet the "alternative shutdown" option in Section III.G.3 of Appendix R, to provide independence and electrical separation from the control room to address a fire-induced control room evacuation. These alternative shutdown controls may be considered the primary control station, provided that, once enabled, the systems and equipment controlled from the panel are independent and electrically separated from the fire area, and the additional criteria below are met.

(1) The location should be considered the primary command and control center when the main control room can no longer be used. The control room team will evacuate to this location and use its alternative shutdown controls to safely shut down the plant.

(2) The location should have the requisite system and component controls, plant parameter indications, and communications so that the operator can adequately and safely monitor and control the plant using the alternative shutdown equipment.

(3) More than one component should be controlled from this location (a local control station provided to allow an individual component to be locally controlled, as in the local handwheel on a motor-operated valve, does not meet this definition).

These criteria are reiterated within FAQ 07-0030. The Remote Shutdown Panels (RSPs) for HNP Units 1 and 2, which meet the criteria defined in the second case, were accepted as alternate shutdown panels by the NRC in SER dated February 11, 1983.

Upon evacuating the control room, operators transfer control to the RSPs and establish control for the pertinent systems (identified below). The systems and equipment controlled from the panel are independent and electrically separated from the control room and cable spreading room. As such, the RSPs are considered the primary command and control centers when the control room can no longer be used. The RSPs are also capable of displaying plant parameters required by operators to ensure that the plant is maintained in a safe and stable condition. Each of the RSPs credited as part of each unit's PCS provide capability for operators to control more than a single component.

For Unit 1, the SER dated February 11, 1983 identifies 6 panels that are part of the RSP system: 1C82-P001, 1C82-P002, 1H21-P173, 1H21-P175, 1H21-P176, and 1H21-P177. The combination of panels 1C82-P001, 1C82-P002, 1H21-

P173, and 1H21-P175 allow for control of two safety relief valves (S/RVs), RCIC, RHR, and Process Monitoring required to monitor the plant. This combination of panels is credited as the Unit 1 PCS. Panels 1H21-P176 and 1H21-P177 provide redundant injection systems via the Control Rod Drive System. Because these systems are not credited in the NSCA, panels 1H21-P176 and 1H21-P177 are not considered part of the PCS for Unit 1.

For Unit 2, SER dated February 11, 1983 identifies 2 panels that are part of the RSP system: 2C82-P001 and 2H21-P173. The combination of these panels allows for control of two S/RVs, RCIC, RHR, PSW, and Process Monitoring required to monitor the plant. This combination of panels is credited as the Unit 2 PCS.

iii. MCR abandonment is only postulated for a fire in Fire Area (FA) 0024, the Control Complex which includes the Main Control Room, Cable Spreading Room and Computer Room. VFDRs are identified differently for FA 0024 in that the compliant case includes the transfer of control for Unit 1 and 2 to the respective Remote Shutdown Panels (RSPs). The RSPs are considered Primary Control Stations as described in HNP LAR Attachment G. Activities that take place at, or those necessary to activate, turn on, power up, transfer control or indication, or otherwise enable the RSPs are not recovery actions and therefore no VFDRs are associated with these actions.

The one exception to this is for motor operated valves (MOVs) subject to the failure modes identified in IN 92-18, prior to transfer of control to the RSP. VFDRs are identified for MOVs subject to the IN 92-18 failure modes based on potential fire induced cable damage that could affect operation of the valve at the RSP.

Any potential failure that challenges the credited train for safe shutdown or requires a recovery action to mitigate the failure that does not occur in the MCR or RSP, is identified as a VFDR regardless of whether that fire area requires MCR abandonment.

iv. The delta risk calculations for MCR abandonment are performed in the same manner as every other fire area with one exception. The compliant case modeling sets a lower bound limit on the CCDP to a minimum of 7E-02. This assumed value was justified by using the CCDP of an abandonment scenario due to loss of habitability with no PRA equipment failures. In some instances, this modeling assumption was implemented due to conservatism in the modeling logic for LOC and transferring to the RSP for compliant model scenarios only. In doing so, this assumption has established a quantified 'floor value' for a more accurate change in risk between the compliant case and the variant case. This assumption is considered conservative given the HEP for transferring control to the RSP is approximately 7E-02. No lower bound limits were used for CLERP in the abandonment compliant cases.

(c) The only type of VFDR identified are considered separation type VFDRs.

In some cases, there are modeling differences between the Fire PRA and the NSCA models. Although VFDRs that are not modeled in the Fire PRA remain in the NSCA, their associated delta risk in the Fire PRA is zero. The VFDRs that are not modeled in the Fire PRA include:

- Differences between the system window requirements between the Fire PRA and the NSCA
- Differences in credited indications between the NSCA and the Fire PRA for required cues to support Fire PRA credited actions
- Conservatism in the NSCA that resulted in a non-credible failure in the Fire PRA that have been refined for realism in the Fire PRA model.

(d) Credited modifications are listed for each fire area. If the modification is associated with a VFDR, the delta risk calculation eliminates the variance via modification. If the modification does not mitigate a specific VFDR the modification is credited in both the compliant and variant models to estimate the delta risk between the post transition plant and the compliant model.

NFPA 805 PRA RAI 16 – Assumed Cable Routing and Other Conservative Modeling

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current FPP to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of RI changes.

Table 6-3, "Fire PRA Sources of Uncertainty," of your application to adopt 10 CFR 50.69, risk-informed (RI) categorization and treatment of structures, systems, and components for nuclear power reactors (ADAMS Accession No. ML18158A583), states that that a sensitivity study was performed to address the uncertainty associated with un-located/untraced secondary-side cables given the conservative assumption made in the FPRA that secondary-side systems are failed in all fires. The LAR does not discuss this sensitivity study nor does it provide the quantitative results of the sensitivity study. The assumption that untraced cables are failed in all fire sequences is a conservative approach for modeling untraced cables in the post-transition plant model, but can lead to underestimation of the change-in-risk when used in the compliant plant model

(a) Describe the extent of untraced FPRA cables and how they were treated in the FPRA. Include an explanation of how they were modelled in both the compliant and post-transition plant FPRA models.

(b) Justify that assumptions made about untraced cables do not contribute to underestimation of the transition change-in-risk. Include a description of the sensitivity study that was performed to address un-located/untraced cables as well as the quantitative results of that sensitivity study.

(c) If failing all untraced cables in the FPRA leads to underestimation of the transition change-in-risk, then demonstrate that the application is not impacted by the underestimation of the transition change-in-risk. Alternatively, replace this conservative approach with an acceptable approach that does not underestimate the change-in-risk in the integrated analysis requested in PRA RAI 03.

(d) If other conservative treatments used in the compliant plant model can be identified as contributing to the underestimation of the total change-in-risk, then identify those conservatisms and demonstrate that the application is not impacted by the corresponding underestimation of the transition change-in-risk. Alternatively, replace such approaches with more realistic approaches in the integrated analysis requested in PRA RAI 03 that do not underestimate the change-in-risk.

SNC Response

(a) The extent of untraced Fire PRA cables include approximately 15% of the Fire PRA components with cables. These were treated in the Fire PRA by globally failing these components in the compliant and post transition plant Fire PRA models.

(b) The Fire PRA risk associated with each group of components with untraced cables was assessed based on the functions failed. The sensitivity performed globally credited these components. The sensitivity resulted in approximately 25% reduction in Fire PRA risk. A review of the results indicated that the 25% reduction was largely because of the global credit for the feedwater system. For many locations in the plant there is a high likelihood that feedwater cables are located (e.g., Turbine Building, Reactor Building, and Control Building) and/or the main steam lines would receive an isolation signal (e.g., Reactor Building and Control Building) precluding use of feedwater. Also, the significant contribution of loss of offsite power accident sequences preclude the use of the feedwater. Additionally, there are no VFDRs associated with the feedwater system. Therefore, the reduction in fire risk in the sensitivity analysis globally crediting untraced Fire PRA cables is exaggerated and untraced Fire PRA cables do not contribute to the underestimation of the transition change in risk.

(c) See response to (b)

(d) Other uncertainties in the PRA identified that may be conservative is based on the current industry guidance. The generic conservatisms include fire ignition frequencies, circuit failure likelihoods, and HRA. Therefore, for NFPA 805, no other treatments are currently considered conservative resulting in the underestimation in the total change in risk.

NFPA 805 PRA RAI 17 – Defense-in-Depth (DID) and Safety Margin

NFPA 805, Section 1.2 indicates that defense-in-depth (DID) shall be achieved when an adequate balance of each of the following DID elements is provided: (1) Preventing fires from starting, (2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage, and (3) Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed. LAR Section 4.5.2.2 provides a high-level description of how impacts on DID and safety margin were reviewed for the transition to NFPA 805, but did not provide sufficient information for the NRC staff to determine whether each DID element was properly addressed. Also, LAR Section states that "[f]ire protection features and systems relied upon to ensure DID were identified as a result of the assessment of DID," but LAR Attachment C, Table C-2 does not identify any fire protection systems or features to be credited for DID. Based on the above identified issues, the NRC staff requests that the licensee:

(a) Explain the criteria used to determine when a substantial imbalance between DID echelons exist in the fire risk evaluations (FREs), and identify the types of plant features and administrative controls credited for providing DID for each of the three DID echelons.

(b) Clarify what fire protection features and systems were relied upon to ensure DID and explain why none are identified in LAR Attachment C, Table C-2.

(c) Discuss the approach for reviewing safety margins using the NEI 04-02, Revision 2, criteria for assessing safety margin in the FREs.

SNC Response

(a) The criteria used to determine whether a substantial imbalance between DID echelons exists in the HNP Fire Risk Evaluations (FREs) were based on the guidance in Nuclear Energy Institute (NEI) 04-02, Revision 2. Specifically, a review was performed of each DID echelon on a fire area basis based on the following considerations:

Echelon 1 (Prevent fires from starting):

Combustible and hot work controls are fundamental elements of defense-in-depth and, as such, are always in place. The issue considered during the FRE was whether this element needed to be strengthened to offset a weakness in another echelon, thereby providing a reasonable balance.

Considerations included:

- Creating a new transient combustible-free area
- Creating a new hot work restriction area
- Modifying an existing transient combustible-free area or hot work restriction area

The fire scenarios involved in the fire risk evaluation quantitative calculation were reviewed to determine if additional controls should be added.

The remaining elements of DID were reviewed to ensure an over-reliance was not placed on programmatic activities for weaknesses in plant design.

Echelon 2 (Rapidly detect, control and extinguish promptly those fires that do occur, thereby limiting fire damage):

Automatic suppression and detection may or may not exist in the fire area in question. The issue considered during the FRE was whether installed suppression and/or detection was required for DID or whether suppression and/or detection needed to be strengthened to offset a weakness in another echelon, thereby providing a reasonable balance.

Considerations included:

1) Risk Insights:

If existing variances form deterministic requirements (VFDRs) were never affected in a "potentially risk significant" fire scenario, manual suppression capability may be adequate with no additional systems required.

For purposes of defense-in-depth, "potentially risk significant" fire scenarios were characterized as follows:

- A scenario in which the calculated risk is equal to or greater than 1.0E-6/year for CDF and/or 1.0E-7/year for LERF.
- A scenario in which the calculated risk falls between 1.0E-6/year and 1.0E-8/year for CDF, or between 1.0E-7/year and 1.0E-9/year for LERF, and where DID echelon 1 and 2 attributes are causing a significant reduction in risk.
- A scenario with a high consequence (i.e., CCDP>1.0E-1) could be considered "potentially risk significant".

2) Recovery Actions:

If the fire area required recovery actions, typically detection and manual suppression capability were considered required. Additionally, requiring automatic suppression was considered.

3) Firefighting Activities:

If firefighting activities in the fire area were expected to be challenging (either due to the nature of the fire scenario or accessibility to the fire location), then the addition of both suppression and detection were considered, if absent.

4) Fire Scenarios:

If fire scenarios credited fire detection and/or fire suppression systems, then these were already considered to form an integral part of DID.

<u>Echelon 3 (Provide adequate level of fire protection for systems and structures so that a fire will prevent essential safety functions from being performed):</u>

If fires occur and they are neither rapidly detected nor promptly extinguished, then the third echelon of DID would be invoked. The issue considered during the FRE was whether existing separation was adequate (or whether an over-reliance was placed on existing separation) and whether additional measures (e.g., supplemental barriers, fire rated cable, or recovery actions) were required to offset a weakness in another echelon, thereby providing a reasonable balance.

Considerations included:

1) Risk Insights:

If existing VFDRs were not affected in a "potentially risk-significant" fire scenario, internal fire area separation was considered adequate and no additional reliance on recovery actions was considered necessary.

If existing VFDRs were affected in a "potentially risk-significant" fire scenario, internal fire area separation may not be adequate and reliance on a recovery action, supplemental barrier, or other modification was considered.

If the consequence associated with existing VFDRs were considered high (e.g., CCDP>1E-01 or by qualitative safe shutdown (SSD) assessment), regardless of whether it is in a risk-significant fire scenario, a recovery action, supplemental barriers, or other modification was considered.

There are known modeling differences between a Fire PRA and NSCA due to different success criteria, end states, etc. Although a VFDR may be associated with a function that is not considered a significant contribution to core damage frequency, the VFDR may have been considered important enough to the NSCA to retain a recovery action as credited for DID, but not required for risk.

For purposes of defense-in-depth, "potentially not risk-significant" fire scenarios were characterized as follows:

- A scenario in which the calculated risk is less than 1.0E-8/year for CDF and/or 1.0E-09 for LERF, regardless of reliance on DID echelon 1 and 2 attributes, may be characterized as "potentially not risk significant". These values are considered "potentially not risk significant" based on being two orders of magnitude below the acceptance criteria of RG 1.174 as referenced by RG 1.205, Rev. 1.

2) Operations Insights:

If the sequence to perform a recovery action was particularly challenging, then including the action for DID was considered.

The fire scenarios involved in the FRE quantitative calculation were reviewed to determine the fires evaluated and the consequence in the area to best determine options for this element of DID.

(b) If a defense-in-depth attribute was credited for NSCA deterministic criteria, licensing action or engineering evaluation, then the system/feature was already considered an integral part of the defense-in-depth determination. The parent echelon of the system/feature was then evaluated to determine if any improvements or changes were necessary, such as to offset a weakness in another echelon.

If the Fire PRA credited any of the fire protection features or a recovery action to improve a risk profile, then these attributes or features were already considered integral parts of the defense-in-depth determination. The parent echelon of the system/feature was then evaluated to determine if any improvements or changes were necessary, such as to offset a weakness in another echelon.

Attributes credited above and beyond the existing requirements with the purpose of bolstering derived weaknesses within the defense-in-depth elements to maintain an overall balance would have been designated as a change or improvement necessary for defense-in-depth. At the conclusion of the review, it was determined that no additional attributes/features were required to be designated as credited for DID in LAR Attachment C, Table C-2.

(c) In accordance with NEI 04-02, Revision 2, the maintenance of adequate safety margin was assessed by the consideration of categories of analyses used by the fire risk evaluations (FREs). Safety margins were considered maintained if:

- Codes and standards or their alternatives accepted for use by the Nuclear Regulatory Commission (NRC) are met; and
- Safety analysis acceptance criteria in the licensing bases (UFSAR, supporting analyses, etc.) are met, or provide sufficient margin to account for analysis and data uncertainty.

The requirements related to safety margins for the FREs were evaluated for each specific analysis type. These analyses were grouped into the following categories:

1) Fire Modeling:

   The methods and input parameters for the specific plant configurations were reviewed. Consideration was given to the verification and validation, state of knowledge, and uncertainties related to the application of the fire modeling tools.

2) Plant System Performance:

   The methods, input parameters, and acceptance criteria used in these analyses were reviewed against those used for the plant design basis events. This review verified that the safety margin inherent in the analyses for the plant design basis events has been preserved in the analysis for the fire event.

   The evaluation of the plant system performance addressed the following:

   - Were input parameters for plant performance analyses (for example, heat transfer coefficients, pump performance curves) altered from those used for plant design basis events such that margin was lessened?
   - Were codes and standards used to determine plant system performance acceptable to the NRC?

3) PRA Logic Model:

   The evaluation of the PRA logic model addressed the following:

   - Was the quantification for fire related CDF / LERF based on the plant PRA model and were plant PRA model modifications required for the Fire PRA, including altered basic event failure probabilities, added basic events, and changes to the logic structure?
   - Changes against the methods and criteria for the overall internal events PRA model development to verify consistency, or confirmation of bounding treatment, to confirm that the Safety Margin is preserved.
   - Was the quantified model sufficient to treat the fire induced core damage sequences? If the analysis was performed using the plant PRA model with no modifications other than normalizing the initiating event frequency to 1.0 and setting other non-credited events to 'TRUE' or 1.0, then the Safety Margin is

preserved and no further assessment for Safety Margin was necessary for this category.

- Was the Fire PRA developed in accordance with NUREG/CR-6850, which was developed jointly between the NRC and EPRI?

NFPA 805 PRA RAI 18 – Impact of a Key Source of Uncertainty on Application

Note: This RAI response will be provided with supplemental correspondence.

Edwin I. Hatch Nuclear Plant Units – 1 and 2
<u>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</u>

Enclosure 5
Response to Fire Protection Engineering (FPE) RAIs

NFPA 805 Fire Protection Engineering (FPE) RAI 01

The compliance strategy for NFPA 805 Section 3.3.5.1, in LAR Attachment A, Table B-1, is identified as "Complies with Required Action," with the actions being revising plant documentation and submitting for NRC approval. In LAR Attachment L Approval Request 3, the licensee indicated that fire zones contain wiring above suspended ceilings that is not in compliance with NFPA 805, Section 3.3.5.1.

The licensee indicated that one of the basis for their approval request is that, "...there are small quantities of low voltage video, communication, and data cables, which are not susceptible to self-ignition." However, the licensee provided no justification for its statements that these types of cables are not susceptible to self-ignition.

Provide the technical basis for this statement and include whether the quantity and material properties of the cables impact the basis for the request.

SNC Response

Technical basis is provided in NUREG-1805, *Fire Dynamics Tools (FDT^s): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program, as follows:*

> "It is common practice to consider only self-ignited cable fires to occur in power cable trays since they carry enough electrical energy for ignition. Control and instrumentation cables typically do not carry enough electrical energy for self ignition."
>
> NUREG-1805 further clarifies, "Instrument circuits generally use low voltages (50 volts or less). Control circuits are commonly in the 120–250-volt range. Power circuits encountered within an NPP generally range from 120 to 4,160 volts, with offsite power circuits ranging to 15 kV or higher."

Therefore, self-ignition of low-voltage cables can be excluded, regardless of quantity and material properties of the cables, on the basis that such cables do not carry enough electrical energy for self-ignition.

NFPA 805 FPE RAI 02

The compliance strategy for NFPA 805, Section 3.3.5.2, "Electrical Raceway Construction Limits" in LAR Attachment A, Table B-1, is identified as "Complies." However, in LAR Attachment L, Approval Request 4, the licensee is requesting approval for the use of polyvinylchloride (PVC) coated flexible conduit in lengths up to 6 feet and embedded non-metallic conduit.

It is not clear whether there are any non-embedded, non-metallic conduits installed at the plant.

Describe whether there are any non-embedded, non-metallic conduits installed at the plant. In addition, discuss whether approval is being requested for future installations or if future installations will be installed in accordance with the requirements of NFPA 805.

SNC Response

HNP Electrical Design Criteria do not allow for non-embedded, non-metallic conduits to be installed within the power block areas of the plant.

Clarifications are also made to Approval Request 4. The first clarification is to request approval for existing installations of flexible metallic and PVC coated flexible metallic conduits in lengths greater than 3 feet (current cable/raceway installation guidance limits flexible metallic conduit installations to 6-feet, unless approved by the Architect/Engineer). The second clarification is to request the future use of flexible metallic and PVC coated flexible metallic conduits in lengths up to 6-feet.

Table S-3, Implementation Item IMP-20 has been created to ensure revision to cable/raceway installation guidance to limit future flexible metallic conduit installations to a maximum of 6 feet. Reference to IMP-20 is included in Attachment A.

Based on the above, Attachment L, Approval Request 4 is revised as follows (new text is italicized and underlined, deleted text is lined through):

"**Approval Request 4**

NFPA 805 Section 3.3.5.2 states:

"*Only metal tray and metal conduits shall be used for electrical raceways. Thin wall metallic tubing shall not be used for power, instrumentation, or control cables. Flexible metallic conduits shall only be used in short lengths to connect components.*"

Plant Hatch uses embedded PVC conduits. In addition, *flexible metallic and* PVC coated flexible metallic conduits in ~~lengths of up to 6 feet~~ *lengths greater than 3-feet* are used to route cables between equipment and rigid conduits. This exceeds the ~~3-foot~~ maximum allowable "short length" as clarified in FAQ 06-0021.

SNC requests NRC approval for*: 1)* the use of nonmetallic conduit in embedded applications*, 2) existing installations* ~~and for the use~~ of flexible *metallic and* PVC coated *flexible* metallic conduits in lengths *greater than 3-feet, and 3) the future use of flexible metallic and PVC coated flexible metallic conduits in lengths* up to 6-feet*,* as acceptable variances from the requirements of NFPA 805, Chapter 3.

Basis for Request:

The basis for the approval request of the deviation for the use of *flexible metallic and* PVC coated flexible metallic conduits in lengths ~~up to 6~~ *greater than 3*-feet *in existing installations* is:

- *Current cable/raceway installation procedures allow for flexible metallic conduit installations up to 6-feet. Flexible metallic conduit installations that exceed 6-feet require Architect/Engineer (A/E) approval. An implementation item will ensure revision to cable/raceway installation guidance that limits future flexible metallic conduit installations to a maximum of 6 feet (See Attachment S, Table S-3, Implementation Item IMP-20).*

- PVC coated flexible metallic conduit provides equivalent physical and electrical protection to uncoated flexible metallic conduit, because the characteristics of the metallic body of the conduit are not affected by the coating.

- According to vendor specifications, the PVC coating on the metallic conduit is very thin and is not expected to provide any credible influence on fire propagation behavior and the amount of PVC introduced to a given fire area is considered negligible.

- If a fire were to occur in a fire area containing these conduits, existing controls such as fire-rated barriers, electrical raceway fire barrier systems, spatial separation, etc. would ensure redundant cabling and circuitry would not be affected by the fire.

- *Flexible metallic and* PVC coated flexible metallic conduits exceeding the 3-foot length clarified in FAQ 06-0021 are installed such that the conduits are not in danger of being damaged by equipment or personnel.

The basis for the approval request of the deviation for the future use of flexible metallic and PVC coated flexible metallic conduits in lengths up to 6-feet is similar to that for flexible metallic and PVC coated flexible metallic conduits in lengths greater than 3-feet above, with the exception of the assurance that future installation of flexible conduit will be no greater than 6 feet in length. As committed above, an implementation item will ensure revision to cable/raceway installation guidance that limits future flexible metallic conduit installations (including PVC coated flexible metallic conduit) to a maximum of 6 feet (See Attachment S, Table S-3, Implementation Item IMP-20).

The basis for the approval request of the deviation for the use of nonmetallic conduit in embedded applications is:

- For instances where nonmetallic conduit is used in concrete embedded applications, the concrete provides physical protection and separation for the conduit.

- The embedded PVC conduits, while combustible material, are not subject to flame or heat impingement from an external source which would result in structural failure, contribution to the fire load, and/or damage to circuits contained within where the conduit is embedded in concrete and exposure is minimal.

- NFPA 70 (National Electric Code (NEC)), Article 352, allows the use of rigid nonmetallic conduit for underground and embedded applications.

- Failure of circuits within embedded *non-metallic* conduits resulting in a fire would not result in damage to external targets (i.e., other circuits would not be exposed to the effects of a circuit failure in the embedded conduit).

- The non-metallic conduits are installed such that the conduits are not in danger of being damaged by equipment or personnel.

- ~~Failure of circuits with non-metallic conduits resulting in a fire would not result in damage to external targets (i.e., other circuits would not be exposed to the effects of a circuit failure in the conduit).~~

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The use of nonmetallic conduit in embedded applications ~~and the use of flexible PVC coated metallic conduits in lengths up to 6-feet~~ does not affect nuclear safety as the material in which conduits are run are located such that they are not subject to failure mechanisms that potentially result in circuit damage or damage to external targets. Additionally, NFPA 70 allows for the use of rigid nonmetallic conduit for underground and embedded applications.

Also, the use of flexible conduit (both metallic and PVC coated metallic) in lengths greater than 3-feet does not affect nuclear safety. If a fire were to occur in a fire area containing these conduits, existing controls such as fire-rated barriers, electrical raceway fire barrier systems, spatial separation, etc. would ensure redundant cabling and circuitry would not be affected by the fire. Therefore, there is no impact on the nuclear safety performance criteria.

The use of nonmetallic conduit in embedded applications and the use of flexible *conduit (both metallic and* PVC coated metallic) ~~conduits~~ in lengths *greater than 3-feet* ~~up to 6-feet~~ have no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the type of conduit material. The conduit material or length of conduit does not change the radiological release evaluation, which concluded that potentially contaminated water is contained and smoke is monitored. The conduits for which NRC approval is requested do not add additional radiological materials to the area or challenge system boundaries.

Safety Margin and Defense-in-Depth:

The areas with nonmetallic conduit in embedded applications and flexible ~~PVC coated metallic~~ conduits *(both metallic and PVC coated metallic)* in lengths ~~up to 6~~ *greater than 3-*feet have been analyzed in their current configuration. The precautions and limitations of the use of these materials do not impact the analysis of the fire event. PVC coated flexible metallic conduit introduces a negligible amount of combustibles to a fire area due to the thickness of the PVC coating. Although, the PVC coating introduces a potential smoke toxicity issue due to its corrosive nature to electrical circuits and sensitive electronics in the event of a fire, the PVC coating is of minimal thickness and would not result in smoke production that would impact electrical circuits or sensitive electronics. *This conclusion also applies to any future installations of PVC coated flexible metallic conduit in lengths not to exceed 6-feet.* Embedded nonmetallic conduit is protected from an exposure fire and possible mechanical damage. ~~PVC conduit that is not embedded~~ *The PVC coating on flexible metallic conduit* introduces a negligible amount of combustibles to an area. Therefore, the inherent safety margin and conservatisms in these methods 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)
(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)

Per NFPA 805 Section 1.2, defense-in-depth is achieved when an adequate balance of each of these elements is provided.

The current configuration of the conduit at Plant Hatch, *and the future use of flexible conduit (metallic and PVC coated metallic) in lengths up to 6-feet,* does not impact fire protection defense-in-depth.

PVC coated flexible metallic conduit used within the plant is constructed of a metallic core coated with a thin layer of PVC. The metal core is expected to withstand any potential exposure fire or flame impingement and the PVC coating is thin enough so that it is not expected to provide any credible influence on fire propagation behavior, therefore not affecting the three echelons of defense-in-depth. When installed in configurations exceeding 3 feet in length, the conduit is not expected to negatively affect the three echelons of defense-in-depth as the additional combustibles added by exceeding 3 feet in length is negligible.

Nonmetallic conduit in embedded applications does not affect the three echelons of defense-in-depth. The use of nonmetallic conduits *in embedded applications* has no effect on the ability for the plant to rapidly detect, control and extinguish any fires that may occur. Additionally, embedded conduit will be shielded from an exposure fire. *Lastly, failure of circuits within embedded non-metallic conduits resulting in a fire would not result in damage to redundant circuits and would not prevent essential safety functions from being performed.* ~~in every area of the plant where redundant pathways or required safe shutdown-related cables area located, one pathway is protected with a fire protection barrier allowing for essential safety functions to be completed.~~

The use of these conduits does not directly result in compromising automatic fire suppression functions, manual fire suppression functions, or post-fire safe shutdown capability, and will not prevent essential functions from being performed.

Conclusion:

NRC approval is requested for*: 1)* the use of nonmetallic conduit in embedded applications*, 2) existing installations* ~~and for the use~~ of flexible *metallic and* PVC coated *flexible* metallic conduits in lengths *greater than 3-feet, and 3) the future use of flexible metallic and PVC coated flexible metallic conduits in lengths* up to 6-feet. 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 shutdown capability)."

NFPA 805 FPE RAI 03

Fire protection systems and features that require NFPA code compliance are reflected in NFPA 805, Chapter 3, "Fundamental Fire Protection Program and Design Elements of NFPA 805." The LAR does not contain a list of codes of record that establishes whether or how the licensee meets Chapter 3 of NFPA 805, therefore, the NRC staff requests that the licensee provide a complete list of the applicable NFPA codes and standards designated as the code of record, including identification of the edition (years), that will be in place post transition. For codes and standards with numerous editions, identify which editions pertain to which particular plant areas and systems.

SNC Response

The table below provides a listing of the NFPA codes of record that will be in place post-transition. For codes and standards with multiple editions, the plant areas or systems to which that edition applies are identified.

NFPA	Title	Code of Record for NFPA 805 Transition	Notes
10	Standard for Portable Fire Extinguishers	1975	
12	Standard on Carbon Dioxide Systems	1973	
13	Standard for the Installation of Sprinkler Systems	1983	
14	Standard for the Installation of Standpipe and Hose Systems	1983	
15	Standard for Water Spray Fixed Systems for Fire Protection	1982	The following systems in Fire Areas/Zones 0101A, 0101B, 0101C, 0101D, 0101J, 0101K, 0101L, 1101J, 0101I, 1606, 0501, 2101J, 2104, 2606 and 2608 are reviewed per the 1982 Edition of NFPA 15: 1U43130W03, 1U43164W01, 1U43164W03, 1U43164W05, 1U43164W07, 1U43164W09, 1U43164W11, 1U43164W13, 1U43164W15, 1U43164W17, 1U43164W18, 1U43164W19, 1X43129W02, 1X43129W03, 1X43129W04, 1Y43111W01, 2U43130W03, 2U43130W06,

NFPA	Title	Code of Record for NFPA 805 Transition	Notes
			2U43164W01, 2U43164W03, 2U43164W05, 2U43164W07, 2U43164W09, 2U43164W11, 2U43164W13, 2U43164W15, 2U43164W17, 2U43164W18, 2X43129W02, 2X43129W05, 2X43129W06
15	Standard for Water Spray Fixed Systems for Fire Protection	2001	The following systems in Fire Areas/Zones 1606 and 2606 are reviewed per the 2001 Edition of NFPA 15: 1X43129W01, 1X43129W07, 1X43129W08
15	Standard for Water Spray Fixed Systems for Fire Protection	2007	System 2X43129W01 in Fire Area 2606 is reviewed per the 2007 Edition of NFPA 15.
15	Standard for Water Spray Fixed Systems for Fire Protection	2012	System 2X43129W02 in Fire Area 2606 is reviewed per the 2012 Edition of NFPA 15.
20	Standard for the Installation of Centrifugal Fire Pumps	1972	
22	Water Tanks for Private Fire Protection	1976	
24	Standard for Outside Protection	1973	
30	Flammable and Combustible Liquids Code	1973	The following liquid storage tanks in Fire Areas/Zones 1003, 2003, 1610, 1611, 1612, 2610, 2612, and Yard are reviewed per the 1973 Edition of NFPA 30: Y52-A001, X43-A002A, X43-A002B, 1R43-A002A, 1R43-A002B, 1R43-A002C, 2Y52-A001A, 2Y52-A001C, N34-A001A, N34-A001B, 2N34-A001A, 2N34-A001B
30	Flammable and Combustible Liquids Code	2008	Applies generically to programmatic storage, handling, and use within the NFPA 805 Power Block
50A	Standard for Gaseous Hydrogen System at Consumer Sites	1973	
51B	Standard for Fire Prevention During Welding, Cutting, and Other Hot Work	2003	
72D	Standard for the Installation, Maintenance and Use	1979	

NFPA	Title	Code of Record for NFPA 805 Transition	Notes
	of Proprietary Protective Signaling Systems		
72E	Standard on Automatic Fire Detectors	1982	
80	Standard for Fire Doors and Windows	1975	
80A	Recommended Practice for Protection of Buildings from Exterior Fire Exposure	1975	
90A	Installation of Air Conditioning and Ventilating Systems	1976	
241	Standard for Safeguarding Construction, Alteration, and Demolition Operations	2000	
600	Standard on Industrial Fire Brigades	2005	

NFPA 805 FPE RAI 04

In LAR Attachment A, Table B-1, the compliance strategies for NFPA 805 Sections 3.3.7.1, 3.3.8, and 3.4.1(a)(1) (for example), are identified as Complies with the use of existing engineering equivalency evaluation (EEEE) but does not describe whether or how non-compliances were resolved. Therefore, the NRC staff requests that the licensee describe how any non-compliances identified during these evaluations were addressed. If any non-compliances are still outstanding, describe how these will be addressed prior the completion of NFPA 805 implementation.

SNC Response

The code compliance reviews completed in support of the NFPA 805 transition at HNP are considered engineering evaluations and, in some instances, provide bases for functional equivalency. Therefore, the "Complies with Use of EEEE's" was used as the compliance statement.

Instances of identified non-compliances were addressed one of three ways: 1) maintenance type items were entered into the site's Corrective Action Program (CAP) and prioritized based on significance; 2) non-maintenance-type deviations were added to Tables S-2 or S-3 of the Transition Report; or 3) in some instances, an engineering evaluation was written justifying the non-compliance as acceptable.

Based on the above, the following implementation item has been added to Table S-3:

Table S-3 Implementation Item 21: Bollards or other appropriate protection will be provided for yard post indicator valves 1Y43-F308P and 1Y43-F316P in accordance with approved plant design. Reference to IMP-21 is included in Attachment A.

Additionally, disposition of and action on two additional items will be provided within a forthcoming LAR supplement. These items are as follows:

1) Heat collectors installed on multiple sprinklers in specific configurations.

2) Fire dampers bolted rigidly to walls and ceilings. These dampers are installed in accordance with approved plant design details; however, the installations are contrary to vendor requirements and Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) guidelines.

Edwin I. Hatch Nuclear Plant Units – 1 and 2

<u>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</u>

Attachment 1

Attachment to NFPA 805 PRA RAI 08

PRA RAI 08 – Transient Control/Hotwork Condition Reports

Fire Zone	Description	Location	Maintenance	Storage	Hotwork	Count of CRs	Discussion
0001	Working Floor & Corridor	Control Building	High	High	High	4	High Ratings allow for some storage of combustible materials.
0002B	Control Building Freight Elevator	Control Building	Medium	Medium	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
0024A	Cable Spreading Room	Control Building	Very Low	Very Low	Very Low	1	See discussion provided in response to PRA RAI 08.
0101A	RFP Oil Conditioner, U1 Turbine Building #8	Turbine Building	Medium	Medium	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
0101D	U1 Reactor Feed Pump B	Turbine Building	Medium	Medium	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
0101J	Main Turbine Deck Area, U2 Turbine Bearing #8	Turbine Building	Medium	Medium	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
0401	Diesel Building Hallway	Diesel Generator Building	Medium	High	Low	3	Medium and High Ratings allow for some storage of combustible materials.
0501	Intake Structure	Intake Structure	High	Medium	Medium	3	Medium and High Ratings allow for some storage of combustible materials.
0704	East Fire Pump Room	Fire Pump House	High	Medium	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
1003	Oil Storage Room	Control Building	Medium	High	Low	1	Medium and High Ratings allow for some storage of combustible materials.
1006	U1 Water Analysis Room	Control Building	Medium	High	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
1105	East Cableway Foyer	Turbine Building	Low	Low	Low	1	See discussion provided in response to PRA RAI 08.
1201	U1 Drywell and Torus	Reactor Building	No	No	No	2	Identified during Refuel Outage
1205F	Work Floor - North CRD, NE&NW Corner, East Water Curtain	Reactor Building	High	High	High	2	High Ratings allow for some storage of combustible materials.
1205S	Work Floor - SE Water Curtain	Reactor Building	Medium	Medium	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
1301I	Chemical Treatment Room	Radwaste Building	Medium	High	Low	2	Medium and High Ratings allow for some storage of combustible materials.
1301Q	Ventilation Room	Radwaste Building	Medium	High	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
2006	Water Analysis Room	Control Building	Medium	High	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
2101A	Under Main Condenser	Turbine Building	No	No	No	3	Identified during Refuel Outage
2101C	Condensate Pump Area	Turbine Building	Medium	High	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
2101G	Offgas Recombiner	Turbine Building	Medium	Medium	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
2101H	(U2) East Corridor	Turbine Building	Medium	Medium	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
2101J	Work Floor - H2 Seal oil Unit, NW Switchgear Area	Turbine Building	High	High	High	2	High Ratings allow for some storage of combustible materials.
2101K	Main Condenser Area	Turbine Building	Very Low	No	No	1	Identified during Refuel Outage

PRA RAI 08 – Transient Control/Hotwork Condition Reports

Fire Zone	Description	Location	Maintenance	Storage	Hotwork	Count of CRs	Discussion
2103	SE Stairwell	Turbine Building	Low	Low	Low	1	See discussion provided in response to PRA RAI 08.
2201	U2 Drywell and Torus	Reactor Building	No	No	No	1	Identified during Refuel Outage
2205B	SE Corner - Pump Room	Reactor Building Below	Medium	Low	Medium	1	See discussion provided in response to PRA RAI 08.
2205F	Work Floor - South CRD, East Water Curtain	Reactor Building	High	High	High	3	High Ratings allow for some storage of combustible materials.
2205N	Chiller Room	Reactor Building	High	Medium	Medium	3	Medium and High Ratings allow for some storage of combustible materials.
2205R	Work Floor - South	Reactor Building	Medium	High	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
2205V	TB Exhaust Filter D004 - Demin, D005, 7, 8 Room	Reactor Building	Medium	Medium	Medium	2	Medium and High Ratings allow for some storage of combustible materials.
2301J	Dry Waste Storage Area	Radwaste Building	Medium	High	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
2605	Circulating Water Pump Pit	Circulating Water Pump Pit	Medium	Medium	Medium	1	Medium and High Ratings allow for some storage of combustible materials.
Yard	Yard	Yard	Medium	Medium	Medium	1	Medium and High Ratings allow for some storage of combustible materials.

Edwin I. Hatch Nuclear Plant Units – 1 and 2
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

Attachment A
Revisions to Transition Report Attachment A
NEI 04-02 Table B-1 – Transition of Fundamental FP Program and Design Elements

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.5.2	Only metal tray and metal conduits shall be used for electrical raceways. Thin wall metallic tubing shall not be used for power, instrumentation, or control cables. Flexible metallic conduits shall only be used in short lengths to connect components.	Complies	Except as identified below, HNP complies with no additional clarification.	Drawing A29500, Conduit and Conduit Support, Ver. 3.0 / Section 3.1 Drawing A29501, General Design Document and Details for the Installation of Nonsafety-Related Electrical Work, Ver. 2.0 / Section 6.0 Drawing B13000, Conduit & Grounding Installation Notes, Ver. 5.0 E-1-03, SNC Raceway Design Standard, Rev. 6 Specification SS-2123-009, Technical Specification for Cable Trays and Cable Tray Accessories for the Edwin I. Hatch Nuclear Plant - Unit 2, Rev. A / All
		Submit for NRC Approval	FAQ 06-0021 defines "short lengths" as approximately three feet of flexible metallic conduit. NRC approval of the use of PVC coated flexible conduit in lengths up to 6 feet and embedded non-metallic conduit is being requested in Attachment L, Approval Request 4.	FAQ 06-0021, Cable Air Drops, Rev. 0 / All
		Complies, with Required Action	Implementation items are identified below.	None

IMPLEMENTATION ITEMS (See Attachment S, Table S-3):

IMP-20 Current cable/raceway installation procedures allow for flexible metallic conduit installations up to 6-feet, or greater if approved by the Architectural Engineer. Cable/raceway procedures will be revised to ensure that installation guidance is limited to a maximum of 6 feet for future flexible metallic conduit installations.

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.7	Individual fire pump connections to the yard fire main loop shall be provided and separated with sectionalizing valves between connections.	Complies	No Additional Clarification	Drawing H-11033 Sheet 1, Fire Protection- P&ID Pumphouse Layout, Ver. 51.0 / All
3.5.8	A method of automatic pressure maintenance of the fire protection water system shall be provided independent of the fire pumps.	Complies	No Additional Clarification	Drawing H-11033 Sheet 1, Fire Protection- P&ID Pumphouse Layout, Ver. 51.0 / All
3.5.9	Means shall be provided to immediately notify the control room, or other suitable constantly attended location, of operation of fire pumps.	Complies	No Additional Clarification	A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All Procedure 34SV-X43-001-1, Fire Pump Test, Ver. 3.5 / Section 7.0
3.5.10	An underground yard fire main loop, designed and installed in accordance with NFPA 24, Standard for the Installation of Private Fire Service Mains and Their Appurtenances, shall be installed to furnish anticipated water requirements.	Complies with Use of EEEE's	The underground yard fire main loop is designed in accordance with NFPA 24 as identified in Calculation SMNH-16-031, NFPA 24 Code Compliance Review.	Calculation SMNH-16-031, NFPA 24 Code Compliance Review, Ver. 1 / All Drawing H-11033 Sheet 1, Fire Protection- P&ID Pumphouse Layout, Ver. 51.0 / All NFPA 24, Standard for Outside Protection, 1973 Edition / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.10		Complies, with Required Action	Implementation items are identified below.	None

IMPLEMENTATION ITEMS (See Attachment S, Table S-3):

IMP-21 Bollards or other appropriate protection will be provided for yard post indicator valves 1Y43-F308P and 1Y43-F316P in accordance with approved plant design.