REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST TO ADOPT
NATIONAL FIRE PROTECTION ASSOCIATION STANDARD 805
PERFORMANCE-BASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTOR GENERATING PLANTS
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
(TAC NOS. ME6629 AND ME6630)

Office of Nuclear Reactor Regulation
Division of Risk Assessment
Fire Protection Branch

RAI-01 Fire Modeling Verification and Validation

National Fire Protection Association (NFPA) 805, Section 2.7.3.2, "Verification and Validation," states: "Each calculational model or numerical method used shall be verified and validated through comparison to test results or comparison to other acceptable models."

Section 4.5.1.2 of the Donald C. Cook Nuclear Plant (CNP) NFPA 805 Transition Report states that a fire modeling study was performed as part of the fire Probabilistic Risk Assessment (PRA) development (NFPA 805, Section 4.2.4.2). During the Week of November 7, 2011, the NRC staff conducted an audit at the CNP site to review the NFPA 805 transition documents. During the audit, the NRC staff noted that fire modeling was done in support of the CNP NFPA 805 license amendment request in the form of a plant-specific Fire Modeling Database (FMDB), called, "Transient Analysis Worksheets". The FMDB 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" or EPRI Fire Induced Vulnerability Evaluation Methodology, Revision 1 (FIVE Rev. 1)?.

Regarding the verification and validation of the fire models:

a) Please describe how FMDB – Transient Analysis Worksheets were verified, i.e., how did the licensee ensure 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 (FDT^s) or FIVE-Rev.1?

b) The fire models that were used in support of the Fire PRA are listed in CNP NFPA 805 Transition Report Section 4.5.1.2. In this section, reference is made to Attachment J of the CNP NFPA 805 Transition Report for a discussion of the acceptability of the listed fire models. For the following models, Attachment J states that: "V&V was documented in NUREG-1824," and that "the correlation is used within the limits of its range of applicability:"

- Flame Height (Method of Heskestad)
- Plume Centerline Temperature (Method of Heskestad)
- Radiant Heat Flux (Point Source Method)
- Hot Gas Layer (Method of MQH)

- Hot Gas Layer (Method of Beyler)
- Hot Gas Layer (Method of Foote, Pagni, and Alvares [FPA])
- Hot Gas Layer (Method of Deal and Beyler)
- Ceiling Jet Temperature (Method of Alpert)
- Smoke Detection Actuation Correlation (Method of Heskestad and Delichatsios)

The fact that a correlation is used within its range of applicability does not guarantee that it is applied within the validated range reported in NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." Please provide technical details to demonstrate that the correlation has been applied within the validated range, or to justify the application of the correlation outside the validated range reported in NUREG-1824.

c) Attachment J of the CNP NFPA 805 Transition Report states that the following models are verified and validated on the basis that they are described in an authoritative publication in fire protection literature:

- Heat Detection Actuation Correlation
- Sprinkler Activation Correlation
- Corner and Wall Heat Release Rate
- Correlation for Heat Release Rates of Cables (Method of Lee)
- Correlation for Flame Spread over Horizontal Cable Trays (FLASH-CAT)

Furthermore, the CNP NFPA 805 Transition Report states that these models are used within their range of applicability, which does not guarantee that they are applied within the validated range. Please provide technical details to demonstrate that the model has been applied within its validated range, or to justify the application of the model outside its validated range.

d) Attachment J of the CNP NFPA 805 Transition Report states that the following model is verified and validated on the basis that it is described in an authoritative publication in the fire protection literature:

- Plume Radius (Method of Heskestad)

Please explain where Plume Radius correlation was used and provide technical details to demonstrate that the model has been applied within its validated range, or to justify the application of the model outside its validated range.

e) Attachment J of the CNP NFPA 805 Transition Report states that the verification and validation of the following applications of Fire Dynamics Simulator (FDS) are documented in NUREG-1824.

- Hot Gas Layer (HGL) Calculations using FDS
- Temperature Sensitive Equipment Zone of Influence Study using FDS
- Plume/Hot Gas Layer Interaction Study using FDS
- Fire Door Closure Calculation using FDS

Please provide technical documentation that demonstrates that FDS was either used within the range of its validity as described in NUREG-1824, or that the use of FDS outside the verification and validation range in NUREG-1824 is justified.

f) Attachment J of the CNP NFPA 805 Transition Report states that the verification and validation of the following applications of Consolidated Model of Fire and Smoke Transport (CFAST) are documented in NUREG-1824:

- HGL Calculations using CFAST (Version 6)
- Temperature Sensitive Equipment Hot Gas Layer Study using CFAST
- Control Room Abandonment Calculation using CFAST

Please provide technical documentation that demonstrates that CFAST was either used within the range of its validity as described in NUREG-1824, or that the use of CFAST outside the verification and validation range in NUREG-1824 is justified.

g) During the site audit the NRC staff observed that part of the fire modeling performed in support of transition to NFPA 805 is described in Engineering, Planning, and Management, Inc. (EPM) Report No. R1900-0411-001, "Verification and Validation of Fire Modeling Tools and Approaches for Use in NFPA 805 and Fire PRA Applications." Appendices B, C and D of this report describe FDS and CFAST fire modeling studies of plume/HGL interaction, temperature sensitive equipment zone of influence (ZOI), and HGL effects.

Please provide the basis of assurance that the use of the conclusions from these studies in subsequent fire modeling analysis was within the limits of applicability.

h) Attachment J of the CNP NFPA 805 Transition Report references verification and validation basis for fire door closure calculations using the FDS code. This was part of the fire modeling analysis for Fire Area [ ]. However, the fire modeling report for this fire area does not mention this.

Please provide a summary of the assumptions and results of the fire door closure FDS computer modeling for Fire Area [ ].

RAI-02 Fire Modeling Uncertainty Analysis

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

Section 4.5.1.2 of the CNP NFPA 805 Transition Report states that uncertainty analyses were performed as required by Section 2.7.3.5 of NFPA 805 and the results were considered in the context of the application.

a) Please explain in detail the uncertainty analyses for fire modeling that was performed, and describe how the uncertainties of the input parameters (geometry, heat release rate, radiative fraction, etc.) were determined and accounted for and substantiate the statement in Appendix J of the CNP NFPA 805 Transition Report which states that, "…the predictions …..are deemed to be within the bounds of experimental uncertainty…"

b) During the audit, the NRC staff reviewed EPM Report No. R1900-0411-[ ], "Detailed Fire Modeling of Compartment [ ] (Unit-2, ESS and MCC Room)". The NRC staff noted that cable tray obstructions were omitted in the FDS fire modeling analysis for Fire Area [ ] In a typical fire risk assessment, there are completeness uncertainties in the risk contribution due to scenarios not explicitly modeled (e.g., smoke damage), model uncertainties in the assessment of those scenarios that are explicitly modeled (e.g., uncertainties in the effect of obstructions in a plume), and parameter uncertainties regarding the true values of the model parameters (e.g., the mass burning rate of the source fuel). Please justify why cable tray obstructions could be omitted in the FDS fire modeling analysis for Fire Area [ ].

RAI-03 (Deleted)

RAI-04 Fire Modeling Quality

NFPA 805, Section 2.7.3, "Quality", describes requirements for calculational models, such as limitations of use, verification and validation of models, etc. This description includes justification of model input parameters, as it is related to limitations of use and validation.

a) The NRC staff noted that no specific discussion was found in the CNP NFPA 805 Transition Report with respect to how the input for the algebraic models was established for fires that involved multiple combustibles. Please explain how the input for the algebraic models was established for fires that involved multiple combustibles and justify the approach that was used.

b) Further, the staff noted that no specific discussion was found in the CNP NFPA 805 Transition Report with respect to how the input for the CFAST models was established for the Main Control Room (MCR) Evacuation study. Please describe the specific CFAST input parameters and provide the CFAST input files for the MCR Evacuation study.

b-1) The staff request the licensee describe the specific FDS input parameters and provide the FDS input files for the detailed fire modeling conducted as described in document R1900-0411-[ ]: Detailed Fire Modeling of Compartment [ ](Unit-2, ESS and MCC Room).

c) During the audit, the NRC staff noted that fire modeling reports, R1900-0411-[ ]: "Detailed Fire Modeling of Compartment [ ]" (Unit-2, ESS and MCC Room), and R1900-0411-[ ]: "Detailed Fire Modeling of Compartment [ ]" (Unit-1 Control Room HVAC and Computer Areas) state that, "The mesh size reflects the finest mesh feasibly allowable with the given computer resources." Please explain why the mesh size used is within the validated range and confirm whether a grid sensitivity study was performed or justify why such a study was not performed.

d) The NRC staff noted that several fire areas have dampers with fusible links, some of which are considered in the fire modeling. No specific discussion was found in the CNP NFPA 805 Transition Report with respect to how the input for the damper fusible link response characteristics was established for fire modeling analyses that included fire dampers. In addition, no specific discussion of how the uncertainty in these input parameters may or may not affect the outcome of the fire modeling calculations. Please explain how the input parameters for the fire damper fusible links were determined and how the uncertainty associated with the response characteristics (Response Time Index) and activation temperature) was accounted for in the fire modeling analysis.

e) During the audit, the NRC staff observed that in the detailed fire modeling report for Fire Area [ ], R1900-0411-[ ]: “Detailed Fire Modeling of Compartment [ ]” (Unit-1 Control Room HVAC and Computer Areas) (supporting documentation for the CNP transition to NFPA 805), a damage threshold temperature of 330 ºC (625 °F) was used for thermoset cable insulation. Please explain why 330°C (625 °F) was used as the damage threshold in Fire Area [ ], in spite of the fact that thermoplastic cables are present in this area, which would indicate that a damage threshold temperature of 205 ºC (400 °F) is appropriate. Further, please explain the consequences of using a lower damage threshold of 205 ºC (400 °F), in terms of the determination of the Limiting Fire Scenario and how that may or may not affect the safety margin analysis.

RAI-05 Fire Protection, Inspection, Testing, and Maintenance

The compliance basis for Element 3.2.3(1) of Attachment A to the CNP NFPA 805 Transition Report (page A-4) states that: “Where practical, performance-based surveillance frequencies may be established as described in Electric Power Research Institute (EPRI) Technical Report (TR) 1006756, “Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features.”"

The use of performance-based methods in lieu of the deterministic requirements of NFPA 805 Chapter 3 must first be approved by the Nuclear Regulatory Commission (NRC). However, it is not clear from the information provided in the CNP NFPA 805 Transition Report how the licensee intends to establish and monitor performance-based surveillance frequencies in the future. In this regard, please provide the following:

a) Describe how EPRI TR 1006756 will be utilized to establish performance-based surveillance frequencies. Specifically discuss how the initial surveillance frequencies will be established, the plant program for obtaining/using performance monitoring information to revise surveillance frequencies post-transition, and the fire protection systems/features to be included in this program.

b) Describe the changes to be made to inspection, testing, and maintenance procedures during the transition period to implement performance-based surveillance frequencies, including identifying the procedures to be changed or developed. Specifically discuss the elements of EPRI TR 1006756 that will be implemented, including identifying the applicable fire protection systems. Clarify which implementation item in Table S-2 of the CNP NFPA 805 Transition Report applies to these procedure changes/developments.

RAI-06 Fire Protection, Suppression Effects

In Table B-3 of the CNP NFPA 805 Transition Report, the description of fire suppression effects for a number of fire areas includes the following phrase (e.g., Fire Area [ ]): “Possible impacts to surrounding equipment due to manual fire fighting activities are bounded by the analysis approach of postulating whole room damage.”

The phrase is included in both deterministic and performance-based fire areas. Clarify the meaning of this phrase and how it applies to both deterministic and performance-based areas. Specifically, discuss what safe shutdown equipment is assumed to be failed in “postulating whole room damage,” and how equipment in deterministically-compliant areas in which both shutdown paths are present is addressed in the analysis.

Also include in the discussion why the statement is limited to manual fire fighting activities rather than all fire fighting activities, including fixed suppression systems.

RAI-07 Fire Protection, Hose Nozzle Clarification

NFPA 805, Section 3.6.3, specifies the requirements for hose nozzles supplied to each power block area. The compliance statement in CNP NFPA 805 Transition Report, Table B-1, Section 3.6.3, states “complies with clarification” however, the compliance basis states that: "The appropriate type of nozzle is provided to each power block area. All hose nozzles have shutoff capability and are able to control water flow from full open to full closed," but the statement is silent regarding the NFPA 805, Section 3.6.3, prohibition of use of straight stream nozzle capability in high voltage settings.

It is unclear what is being clarified. Please describe the clarification. Include a discussion of whether or not nozzles capable of straight stream settings are used in areas with high voltage equipment.

RAI-08 Fire Protection, Water Supply - Standpipes and Hose Stations

NFPA 805, Section 3.6.4, states: “Provisions shall be made to supply water at least to standpipes and hose stations for manual fire suppression in all areas containing systems and components needed to perform the nuclear safety functions in the event of a safe shutdown earthquake (SSE).”

This section of NFPA 805 also contains the following exception: “For existing plants that are not capable of meeting this requirement, provisions to restore a water supply and distribution system for manual fire-fighting purposes shall be made. This provisional manual fire-fighting standpipe/hose station system shall be capable of providing manual fire-fighting protection to the various plant locations important to supporting and maintaining the nuclear safety function. The provisions for establishing this provisional system shall be preplanned and be capable of being implemented in a timely manner following an SSE.”

The above exception is not endorsed by 10 CFR 50.48(c)(2)(vi) but may be used in accordance with 10 CFR 50.48(c)(2)(vii). The CNP NFPA 805 Transition Report, Table B-1, Section 3.6.4, compliance statement, states “compliance by previous NRC approval.” The basis for the previous approval as described in Table B-1 does not address the capability to provide manual suppression following an SSE. Please describe how CNP meets the requirement to provide manual suppression to areas containing systems and components needed for nuclear safety functions following an SSE.

RAI-09 Fire Protection, Fire Watch Duties

NFPA 805, Section 3.3.1.3.1, specifies the requirements for a hot work safety procedure. The CNP NFPA 805 Transition Report, Table B-1, states “Complies with clarification", and the compliance basis states: “A hot work safety procedure and a fire watch procedure have been developed, implemented, and are periodically updated as necessary. Fire watch personnel may have multiple duties.”

Describe what additional duties the fire watch may perform and what assures these duties will not impede the ability to perform the required fire watch duties.

RAI-10 Fire Protection, Clarifications of Fundamental Requirements in Chapter 3

CNP NFPA 805 Transition Report, Table B-1, compliance statements for Sections 3.3.1.2(1), 3.3.1.2(3), and 3.4.1 state: "Complies with clarification." The clarifications in the compliance basis statements include the following:

- Section 3.3.1.2 (1), "For short durations, covering the material with a fire retardant cloth." NFPA 805, Section 3.3.1.2 (1), does not include an exception for short durations.
- Section 3.3.1.2 (3), "and stored in proper containers." NFPA 805, Section 3.3.1.2 (3), does not include a provision for in-situ storage of "waste, debris, scrap, packing materials, or other combustibles."
- Section 3.4.1, "The composition of the fire brigade may be less than the minimum requirements for a period of time not to exceed 2 hours." NFPA 805, Section 3.4.1, does not include this provision.

These elements appear to deviate from the NFPA 805 requirements as opposed to being a clarification of how the requirement is met. Please provide further justification for these deviations from NFPA 805 and why they are not considered performance-based methods under 10 CFR 50.48(c)(2)(vii).

RAI-11 Fire Protection, Dedicated Fire Water Supply

NFPA 805, Section 3.5.16, states that: "The fire protection water supply system shall be dedicated for fire protection use only."

NFPA 805 indicates that there are two exceptions including:

- Fire protection water supply systems shall be permitted to be used to provide backup to nuclear safety systems, provided the fire protection water supply systems are designed and maintained to deliver the combined fire and nuclear safety flow demands for the duration specified by the applicable analysis.

- Fire protection water storage can be provided by plant systems serving other functions, provided the storage has a dedicated capacity capable of providing the maximum fire protection demand for the specified duration….

CNP NFPA 805 Transition Report, Table B-1, Section 3.5.16 and Attachment L requests NRC approval for use of the fire water supply system for non-fire protection related purposes. Please provide the following additional information or justification in support of this approval request:

a) Describe how fire water is used for non-fire protection purposes and any controls on maximum use or duration.

b) Provide more detail regarding the controls such as work control procedure(s), notifications, procedural limitations, temporary modification procedural controls, system restrictions or impairments, communications, and alarm responses required when non-firewater use is underway.

c) Describe in more detail the capacity, flow rates, and bounding values for the uses of non-firewater fire protection system.

d) Identify and describe the fire protection features, including procedures and work practices, if applicable, being credited for meeting the defense-in-depth (DID) criteria listed in paragraph (C) of Approval Request 1.

RAI-12 Fire Protection Program, Monitoring Program

NFPA 805, Section 2.6, “Monitoring,” states that: "A monitoring program shall be established to ensure that the availability and reliability of the fire protection systems and features are maintained and to assess the performance of the fire protection program in meeting the performance criteria." and that "Monitoring shall ensure that the assumptions in the engineering analysis remain valid."

Specifically, NFPA 805, Section 2.6, states that: (2.6.1) "Acceptable levels of availability, reliability, and performance shall be established." (2.6.2) "Methods to monitor availability, reliability, and performance shall be established. The methods shall consider the plant operating experience and industry operating experience." (2.6.3) "If the established levels of availability, reliability, or performance are not met, appropriate corrective actions to return to the established levels shall be implemented. Monitoring shall be continued to ensure that the corrective actions are effective."

Section 4.6 of the CNP NFPA 805 Transition Report states that the CNP NFPA 805 monitoring program will be implemented as part of the fire program transition to NFPA 805 (Attachment S, Table S-3, Implementation Items, Item S-3.2 of the CNP NFPA 805 Transition Report) after the safety evaluation is issued. Furthermore, the licensee has indicated that the monitoring program will be developed in accordance with, Frequently Asked Question (FAQ) 10-0059. The NRC staff noted that the information provided in Section 4.6, “Monitoring,” of the CNP NFPA 805 Transition Report, is insufficient for the NRC staff to complete its review of the monitoring program and as such is requesting that the following additional information be provided:

a) A description of the process by which systems, structures, and components (SSCs) will be identified for inclusion in the NFPA 805 monitoring program including an explanation of how SSCs that are already included within the scope of the CNP Maintenance Rule program will be addressed with respect to the NFPA 805 monitoring program.

b) A description of the process that will be used to assign availability, reliability, and performance goals to SSCs within the scope of the CNP NFPA 805 monitoring program including the approach to be applied to any SSCs for which availability, reliability, and performance goals are not readily quantified.

c) A description of the procedures that will be employed to address SSCs that fail to meet assigned availability, reliability, or performance goals.

d) A description of how the CNP NFPA 805 monitoring program will address response to programmatic or training elements that fail to meet performance goals (examples include fire brigade response or performance standards and discrepancies in programmatic areas such as combustible control programs).

e) A description of how the CNP NFPA 805 monitoring program will address fundamental fire protection program elements.

f) A description of how the guidance in EPRI Technical Report 1006756 will be integrated into the CNP NFPA 805 monitoring program.

g) A description of how periodic assessments of the monitoring program will be performed taking into account, where practical, industry wide operating experience including whether this process will include both internal and external assessments and the frequency at which these assessments will be performed.

RAI-13 Radioactive Release, Contaminated Equipment Storage Building

During the audit, the NRC staff observed that the Contaminated Equipment Storage Building identified in Table 3-2 of Technical Evaluation R1900-0041-0001, Rev. 0, “Plant Boundary Partitioning and Definition” does not appear to be included in the radioactive release transition review (CNP NFPA 805 Transition Report, Attachment E). It is unclear how the radioactive release performance criteria in NFPA 805, Section 1.5.2, can be shown as met given the exclusion of this facility from the review. Please clarify whether this facility contains radiological contamination and, if so, provide justification for not including it in the radioactive release transition review. Also, identify any other facility or area that is contaminated or contains radioactive material that has been excluded from the radioactive release performance review and provide justification for not including it.

RAI-14 Safe Shutdown Analysis, Non-Power Operations

The staff requests that the licensee provide the following pertaining to non-power operations (NPO) discussions provided in Section 4.3 and Attachment D of the CNP NFPA 805 Transition Report:

a) Identify and describe modifications to outage management procedures, and risk management tools (e.g., procedure PMP-4100-SDR-001, “Plant Shutdown and Risk Management”) resulting from incorporation of Key Safety Functions (KSF) identified as part of NFPA 805 transition. Include identification of administrative procedures such as “Control of Combustibles”.

b) Provide a list of pinch points by fire area that were identified in the non-power operation (NPO) fire area reviews using FAQ 07-0040 guidance. Describe how these locations will be identified to the plant for implementation.

c) Numerous resolutions for pinch points listed in Attachment B of Technical Evaluation 1900-005-001, Rev 0, call for “protection strategies to be used.” Describe what kinds of protective strategies will be used to prevent occurrences of fire related events.

d) During NPO modes, spurious actuation of valves can have a significant impact on the ability to maintain decay heat removal and inventory control. Provide a description of any actions being credited to minimize the impact of fire-induced spurious actuations on power operated valves (e.g., air-operated valves and motor-operated-valves) during NPO (e.g., pre-fire rack-out, actuation of pinning valves, and isolation of air supplies).

e) During normal outage evolutions, certain NPO-credited equipment will have to be removed from service. Describe the types of compensatory actions that will be used during such equipment down-time.

f) The description of the NPO review for the CNP NFPA 805 Transition Report does not identify locations where key safety functions (KSFs) are achieved solely via recovery actions or for which instrumentation is needed to support recovery actions required to maintain a safe and stable state. Describe consideration of recovery actions for NPO and how recovery action feasibility is evaluated. Include in the description whether there were any KSFs that are achieved solely by crediting recovery actions, whether different instrumentation is needed to support recovery actions, and whether these variables have been factored into operator procedures supporting these actions.

RAI-15 Safe Shutdown Analysis, Nuclear Safety Capability Assessment (NSCA) Capability Methods - Common Power Supply and Enclosure

CNP NFPA 805 Transition Report, Table B-2, states that CNP complies with the intent of NEI 00-01, Revision 1, Sections 3.3.1.7, 3.3.3.3, 3.5.2.4, and 3.5.2.5, with regard to common power supply and common enclosure criteria and the methodologies referenced in D.C. Cook Technical Evaluation 12.5, "Associated Circuits by Common Power Supply and Common Enclosure", Revision 0, Technical Evaluation 12.5.1, "Secondary Fire Evaluation," Revision 0, and Engineering Evaluation, Enercon Report AEP-DCC-11-001, "Associated Circuits by Common power Supply and By Common Enclosure", Revision 0. Please provide the following additional information regarding secondary fires:

a) With regard to Technical Evaluation 12.5.1, the only location of concern becomes the 4kV Switchgear Rooms. A fire in that switchgear room causes loss of 125 VDC control power and, subsequently, the capability to trip the power load breaker when it is faulted in the same fire. The secondary fire could occur along the upstream incoming power line from the startup transformer or DG to the switchgear room. This secondary fire may jeopardize the ability to achieve and maintain the nuclear safety performance criteria. Discuss how this issue has been resolved.

b) With regard to Technical Evaluation 12.5, deficiencies are identified in protection of 250 VDC cables associated with 1-MCAB, 1-HIV, 1-MCCD, 1-CRCD, 1-MDCD, 1-11PHC, 1-11PHA, 1-TBV, Bus1A, Bus1B, Bus1C, Bus1D, 2-MCCD, 2-HIV and 2-MCAB. Attachments 4 and 5 provide impact statements for these deficiencies for Unit 1 and Unit 2, respectively, that state, "Unless a justification can be provided for accepting the lack of cable protection secondary fires must be assumed." Have the deficiencies identified for the 250 VDC components been justified or have secondary fires been evaluated for these cables? Explain what impact secondary fires will have on the ability to achieve and maintain the nuclear safety performance criteria and provide supporting documentation.

RAI-16 Safe Shutdown Analysis, Current Transformers Outstanding Work

CNP NFPA 805 Transition Report, Table B-2, Section 3.5.2.1, provides guidance for evaluation of open circuit failures including those associated with current transformers (CTs). Technical Evaluation 12.6, "Open Circuiting Concern of Current Transformers," Revision 0, states that the issue of secondary ignition of CTs caused by open circuiting of the secondary side was sufficiently addressed for Appendix R in the Fire Protection Program Manual (FPPM), Section 12.6, but that a review was performed again in support of NFPA 805 transition. This evaluation also states that the CTs reviewed were located on both trains in both Unit 1 and Unit 2 480V, 600V, 4kV non-safety and safety buses, and main generator related circuits. However, the evaluation is described as unfinished because necessary manufacturer data was not available at the time the evaluation was performed. Three different possible actions are identified for resolving this incompleteness, including plant modifications and further circuit analysis. CNP NFPA 805 Transition Report, Attachment S, identifies this as an implementation item. Please describe the action chosen, the status of implementation, and how the resolution will affect the analysis, including delta core damage frequency/large early release frequency (CDF/LERF), as presented in the CNP NFPA 805 Transition Report.

<u>RAI-17 Safe Shutdown Analysis, Cross-connected U1 and U2 Systems</u>

The CNP NFPA 805 Transition Report identifies recovery actions requiring cross-connecting of Unit 1 and Unit 2 systems (i.e., chemical and volume control system, auxiliary feedwater system, emergency service water system) to achieve the nuclear safety performance criteria. Please provide additional information or discussion for the following:

a) Do these cross-connecting actions require staff from both units? If so, describe how the feasibility analysis reflects this Unit 1 – Unit 2 staffing, communication, and operational interface.

b) What are the operational impacts on the unaffected (by fire) unit created by the cross-connecting of these systems?

c) Provide the contribution to Unit 1 risk (CDF and LERF) due to a fire in Unit 2 and vice-versa.

d) During the site audit, the NRC staff observed that Procedure 1/2 – OHP-4025-LS-6 RCS, “Makeup with CVCS Cross Tie” indicated that once the cross tie with the opposite unit had been completed for fire safe shutdown, operators would enter Technical Specification (TS) 3.0.3 for the plant operating status. However, amendments Nos. 131 and 116 to Facility Operating Licenses DPR-74 for D.C. Cook Units 1 and 2, dated February 9, 1990, was approved by the NRC allowing this cross-connect be made. The TS change identified that the provisions of TS 3.0.3 are not applicable. This appears to be contradictory. Please provide clarification.

<u>RAI-18 Safe Shutdown Analysis, Gas Powered Blowers</u>

Provide additional information on the configuration and use of gasoline-engine-driven blowers for temporary control room ventilation. Please describe the timing required to perform these identified recovery actions (Table G-1: Variance from Deterministic Requirements (VFDRs) [ ] and [ ]). Also, discuss the means to exhaust the engine combustion gases and any special controls needed to safely handle and/or address fuel handling inside the Power Block.

<u>RAI-19 Safe Shutdown Analysis, NEI 00-01 Rev. 1 versus Rev. 2 and Rising Stem Valves</u>

Table B-2 of the CNP NFPA 805 Transition Report indicates that the Nuclear Safety Capability Assessment (NSCA) was evaluated against guidance in NEI 00-01, “Guidance for

Post-Fire Safe Shutdown Circuit Analysis," Rev. 1. During the site audit, the NRC staff observed that the CNP NSCA report, Technical Evaluation R1900-0024-001, Rev. 0, refers to guidance taken from NEI 00-01 and identifies NEI 00-01, Rev. 2, as the citation in the Section 13 reference list. During the site audit, the NRC staff also observed that Technical Evaluation 12.29, "D.C. Cook Nuclear Safety Capability Analysis Methodology Review," Section 1.0, states that changes between Revision 1 and 2 of NEI 00-01 represent "general clarifications" rather than changes in intent, criteria, or assumptions. In light of these statements and references it is not clear which version of the NEI 00-01 guidance was used to support the NSCA.

NEI 00-01, Rev. 2, is the version currently endorsed in RG 1.205, "Risk-Informed, Performance-Based Fire Protection For Existing Light-Water Nuclear Power Plants" Rev. 1. Clarifications provided in NEI 00-01, Rev. 2, in many cases provide focus or information on an issue not previously provided. For example, Section 3.2.1.2 of NEI 00-01 Rev. 2, provides clarification about possible fire damage to rising stem valves and the expected justification needed to support post-fire operation of these valves. There is evidence in the supporting documentation that this particular guidance was not followed. During the site audit, the staff observed that Technical Report R1900-0026-001, "Recovery Action Transition Report in Support of NFPA 805," Section 3, states, with regard to recovery actions, that: "Valve hand wheels are available for the manual operation of the valve after exposure to a fire."

Please clarify which version of the NEI 00-01 guidance was used to perform the NSCA, and if the Rev. 1 version was used, provide a gap analysis between Rev. 1 and Rev. 2 to demonstrate meeting the guidelines of Rev. 2. Include in this discussion explanation of whether credit was taken for manual operation of rising stem valves exposed to fire, and if so, provide justification of this credit.

RAI-20 Probabilistic Risk Assessment, Internal Events PRA

The gap analysis performed in 2004 identified gaps between the RA-Sa-2003 and the original Peer Review in 2001. The Focused Scope Peer Review conducted in 2009 was tightly focused on the new system models. Therefore, the internal events PRA was never reviewed against all the supporting requirements endorsed by Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Rev. 2. There were some changes between the internal events SRs between the RA-Sa-2003 and RA-Sa-2009 version of the standard (as endorsed by RG 1.200). The quality of the internal events PRA against the requirements of RG 1.200, Rev. 2 needs to be established by identifying differences between the SRs of the 2003 and 2009 Standards and the impact of these differences on the results of the previous reviews (e.g., a gap analysis). Please summarize this evaluation and the results and the impacts of these gaps on the application.

RAI-21 Probabilistic Risk Assessment, F & O Resolution

CNP NFPA 805 Transition Report, Attachment U, Table U-1 identifies previous Findings and Observations (F&Os) and provides a proposed resolution for each F&O. Many of the proposed resolutions do not provide sufficient information for the NRC staff to conclude that the internal events PRA, used as a basis for the fire PRA, is of sufficient quality to support the estimated change in risk analyses.

a) All proposed responses that “[t]he disposition for the Internal Events PRA model brought the affected Supporting Requirement up to at least Capability Category (CC) II” should be replaced with a description of how the F&O was resolved.

b) A number of F&Os address the use of an old version of Modular Accident Analysis Program (MAAP) on success criteria and time available for operator actions. Describe how the MAAP analyses required to support the change in risk for transition have been updated.

c) In some cases, the statement is made that the issue only impacts the internal events PRA not the fire PRA. Clarify how this conclusion was reached.

RAI-22 Probabilistic Risk Assessment, Safe and Stable

In many cases, e.g., FZ-144 in [ ], extensive use of Unit 2 equipment is credited to bring Unit 1 into a safe and stable state and vice versa. Please confirm that CNP has a Unit 1 PRA that appropriately includes Unit 2 equipment, and a separate Unit 2 PRA that appropriately includes Unit 1 equipment.

RAI-23 Probabilistic Risk Assessment, Defense-In-Depth Methodology

Describe the methodology that was used to evaluate defense-in-depth. The description should include what was evaluated, how the evaluation was performed, and what, if any, actions or changes to the plant or procedures were taken to maintain the philosophy of defense-in-depth.

RAI-24 Probabilistic Risk Assessment, Safety Margins

Describe the methodology used to evaluate safety margins.

RAI-25 Probabilistic Risk Assessment, Safe and Stable Actions and Risk

NFPA 805, Section 1.3.1, states: "The nuclear safety goal is to provide reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition."

Section 4.2.1.2 of the CNP NFPA 805 Transition Report describes an initial coping time of 24 hours, after which, actions are necessary to sustain Mode 3 beyond 24 hours. Provide additional discussion of the actions necessary beyond 24 hours to meet the specific nuclear safety performance criteria and maintain safe and stable conditions. Evaluate quantitatively or qualitatively the risk associated with the failure of actions and equipment necessary to extend safe and stable beyond 24 hours given the post-fire scenarios during which they may be required.

RAI-26 Probabilistic Risk Assessment, VFDR [ ]-003

CNP NFPA 805 Transition Report, Attachment C, “NEI 04-02, Table B-3 – Fire Area Transition,” states that no further action is required for VFDR [ ]-003. Describe the expected operator response to loss of refueling water storage tank (RWST) level. Clarify whether the loss of RWST level indication could impact procedures such as 1-OHP-4024-105 and 1-OHP-4023-ES-103. If so, describe how this will be reflected in the CNP NFPA 805 Transition Report. Clarify if this concern applies to Unit 2 (Area [ ]) and, if so, address accordingly.

RAI-27 Probabilistic Risk Assessment, VFDR [ ]-001

VFDR [ ]-001 credits operator actions to close steam generator (SG) 1 and 4 pilot-operated relief valves (PORVs) to support main steam isolation. Is there an estimated ΔCDF and LERF for this VFDR or is it assumed to be some low value such as epsilon? Clarify if this low value is based on the assumption that there are alternative water sources for the CVCS injection or that there is not a demand for CVCS injection from the RWST. If this operator action fails (0.1 assumed in calculation) then an overcooling event begins and a demand for CVCS is initiated. Provide the fire area and total delta CDF and LERF if this operator action failure is propagated through the fire PRA (i.e., wherever this action is credited not just [ ]-001). Explain how an operator knows to switch over to recirculation without RWST level and how this is reflected in the PRA.

RAI-28 Probabilistic Risk Assessment, Fire Risk Evaluation

In CNP NFPA 805 Transition Report, Attachment W, Tables W-3 and W-4, some entries under the last column (Fire Risk Eval delta CDF/LERF), are *"N/A"* and some are *"e /e "* (EPSILON). Describe what *"N/A"* and *"e /e "* (EPSILON) mean, and what the difference is.

RAI-29 Probabilistic Risk Assessment, Findings & Observation (F&O) FSS-A6-2

The probability of the operators failing to successfully shut down the plant after having to abandon the control room has been assigned a value of 0.1 as indicated in F&O FSS-A6-2. Please explain why this is an appropriate value.

RAI-30 Probabilistic Risk Assessment, Uncertainty

Uncertainty: PRA-FIRE-17633-015-LAR, Table 4-1, provides a sensitivity study for initiating event frequencies in accordance with NUREG/CR-6850 "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," instead of FAQ 48. Using the NUREG/CR initiating event values reportedly increases the (6850-based) change-in-risk estimate above the acceptance guidelines. Please provide the results of the sensitivity study and provide a description of any considerations (e.g., human error probability sensitivities) that, if credited, could reduce the 6850-based change-in-risk to below the acceptance guidelines. Clarify if the licensee will propose any changes, to the methods or facility that would reduce the total 6850-based change-in-risk below the CDF acceptance guideline of 1.0E-05 per year. Provide justification if no changes are being considered to reduce the 6850-based estimates to below the acceptance guidelines.

RAI-31 Probabilistic Risk Assessment, F&O UNC-A1-1

Although the response to F&O UNC-A1-1 indicates that parameter uncertainty analysis for risk-significant zones has been performed, inspection of the referenced document (FIRE-PRA-17633-015-LAR) indicates that a rigorous statistical propagation of parametric uncertainty has not been completed. Please provide the calculated mean value for fire CDF and LERF of the post-transition plant and its relation with the calculated point estimate used in the evaluations for the post-transition plant. Propagating parametric uncertainty may increase the change in risk estimate above the acceptance guidelines. Describe how you evaluated the effect of propagating parametric uncertainty on the change in risk estimate. Clarify if propagating the uncertainty would cause the change-in-risk estimate to increase beyond the acceptance guidelines. If not, provide your explanation.

RAI-32 Probabilistic Risk Assessment, Calculations

While Tables W-1 and W-2 of the CNP NFPA 805 Transition Report provide the Δ CDF and Δ LERF of the VFDRs for each of the fire areas, the CNP NFPA 805 Transition Report does not describe either generically or specifically how Δ CDF and Δ LERF were calculated. Describe the method(s) used to determine the changes in risk reported in the Tables in Appendix W. Include in the description discussion of specific model additions or modifications needed to determine the changes.

RAI-33 Probabilistic Risk Assessment, Plant/Procedure Changes

Provide a description of any changes to the plant or procedures that are necessary to meet the change in risk acceptance guidelines but that are not included in CNP NFPA 805 Transition Report, Attachment S.

RAI-34 Probabilistic Risk Assessment, Peer Review F&Os

Clarify the following dispositions to the fire PRA peer review F&Os (CNP NFPA 805 Transition Report, Attachment V, Table V-1) that appear to have the potential to noticeably impact the fire PRA results but which do not appear to be fully resolved:

a) Describe how the resolutions of open F&Os from the 2001 internal events peer review were incorporated into the fire PRA. Include in the response specifically how the three cited open F&Os in PRM-B2-1, judged to have significant potential impact to the fire probability assessment (FPRA) risk profile, were addressed.

b) Describe how the treatment of fire-induced spurious equipment operations was upgraded in response to FSS-A2-1 to address potential multiple concurrent cable failures and cause undesired equipment operations. Specifically address in the response how two cables that if affected and could cause spurious operations were considered.

c) Describe specifically how Technical Evaluation 11.57 has been updated in response to FSS-C8-1 to provide objective evidence that credited wraps are not subject to mechanical or heat related damage from high hazard sources. Include in the response how High Energy Arcing Faults (HEAFs) were considered.

d) Explain why Capability Category I using generic unavailability values for detection and suppression systems is acceptable for the application. Include in this explanation why an evaluation for plant specific outlier behavior cited in FSS-D7-1 as being conducted at the time of the peer review was not used.

e) IGN-A7-1 through IGN-A7-3. Describe the fire frequency apportioning methodology used for transient fires. Include in the response justification for: 1) the occupancy weighting factor of 0.2, storage weighting factor of 0.1 and maintenance weighting factor of 0.05 cited in IGN-A7-1, 2) application of the weighting factor to Bins 7, 25, and 37 cited in IGN-A7-2, and 3) use of a weighting factor of "0" for outside areas cited in IGN-A7-3. Provide a sensitivity study that shows the impact on both the total and change in fire risk of using the weighting factor for Low (or 1) suggested by Section 6.5.7.2 of NUREG/CR-6850, "PRA Procedures and Uncertainty for PTS Analysis," Vol. 2, for transients, rather than the "special weighting factors" presented in

Table 3-6 of PRA-FIRE-17663-006-LAR, Rev. 0. Provide the results of the sensitivity study and if the result of the sensitivity study is a Δ risk that is above the acceptance guidelines, describe additional means being used to reduce the total VFDR Δ risk below the CDF acceptance criteria of 1.0E-05 per year. Provide justification for use of this alternative method if no additional means are being considered to further reduce risk.

f) Explain how conditional failure probabilities for fire-induced circuit failures were determined Include in the response how each of the 11 suggestions provided in CF-A1-1 were addressed.

g) Explain how each of the unvalidated assumptions cited in SF-B1-1 about seismic fire interactions were updated.

h) (Deleted)

i) Describe how the multi-compartment analysis has been revised in response to FSS-G5 to take credit for active fire barriers, including how active barrier performance was considered for the high energy events (e.g., transformer explosions, HEAFs). Include in this description how the assessment performed in response to PP-B5 to evaluate high energy hazards on active fire barriers was considered

RAI-35 Probabilistic Risk Assessment, Modifications

Describe any methods used for the CNP NFPA 805 Transition Report, Attachment S, or other modifications that were not included at the time of the FPRA peer review. Include a discussion of any basic events or models added to the PRA to address these modifications.

RAI-36 Probabilistic Risk Assessment, PRA Changes

Implementation Item S-3.9 (CNP NFPA 805 Transition Report, Attachment S, Table S-3) indicates that fire PRA model changes may be made during the transition period based on the results of field walk downs of the recovery actions. PRA methods or approaches different than those used in the peer reviewed model and previously determined to be acceptable by the NRC must be reviewed and approved by the NRC. Clarify the types of PRA changes anticipated and consider whether an additional implementation item is needed to submit a request for NRC review and approval of identified PRA/plant changes resulting from completion of existing implementation items (e.g., S-3.9). Provide justification if an additional implementation item is determined to not be necessary.

RAI-37 Probabilistic Risk Assessment, Recovery Actions

CNP NFPA 805 Transition Report, Attachment G, states that the identified recovery actions were determined to be feasible, per the criteria in FAQ 07-0030, and reliable, based on the fire PRA Human Reliability Analysis. Describe how these evaluations considered potential staffing and priority conflicts with other operator manual actions included in plant procedures. In the response specifically address how the impact of these potential conflicts was considered for recovery actions and other operator manual actions needed in the same scenario.

RAI-38 Probabilistic Risk Assessment, Sensitivity Analysis

It was recently stated at the industry fire forum that the Phenomena Identification and Ranking Table Panel being conducted for the circuit failure tests from the DESIREE-FIRE tests may be eliminating the credit for Control Power Transformer (CPT) (about a factor 2 reduction) currently allowed by Tables 10-1 and 10-3 of NUREG/CR-6850, Vol. 2, as being invalid when estimating alternating current (AC) and DC circuit failure probabilities. Provide a sensitivity analysis that removes this CPT credit from your PRA and provide new results that show the impact of this potential change for CDF, LERF, Δ CDF, and Δ LERF.

RAI-39 Probabilistic Risk Assessment, Alternative Methods

In addition to the alternative approach used to modify the transient fire frequency apportionment method (see F&O IGN-A7-1) presented in NUREG/CR-6850, identify any other deviations from the methods described in NUREG/CR-6850 or in an NRC endorsed FAQ that were not previously identified as deviations in your license amendment. For each deviating alternative method used, provide a sensitivity study that shows the impact on the fire risk, and the change in risk, of using the method versus using a previous accepted method.