PORTIONS OF ENCLOSURE 2 TO THIS LETTER CONTAIN SENSITIVE INFORMATION SUBMITTED UNDER 10 CFR 2.390(d)(1).

Progress Energy

Christopher L. Burton
Vice President
Harris Nuclear Plant
Progress Energy Carolinas, Inc.

Serial: HNP-10-008
10 CFR 50.90

FEB 04 2010

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

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
RESPONSE TO SECOND REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO ADOPT NATIONAL FIRE PROTECTION ASSOCIATION STANDARD 805, "PERFORMANCE-BASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTOR GENERATING PLANTS" (TAC NO. MD8807)

References:

1. Letter from M. Vaaler, Nuclear Regulatory Commission, to C. L. Burton, "Shearon Harris Nuclear Power Plant, Unit 1 – Second Request for Additional Information Regarding the License Amendment Request to Adopt National Fire Protection Association Standard 805, 'Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants' (TAC NO. MD8807)," dated January 14, 2010

2. Letter from C. L. Burton to the Nuclear Regulatory Commission (Serial: HNP-09-094), "Third Response to Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Association Standard 805, 'Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants' (TAC NO. MD8807)", dated October 09, 2009

Ladies and Gentlemen:

On January 14, 2010, the Harris Nuclear Plant (HNP) received a second request from the NRC (Reference 1) for additional information needed by the NRC to facilitate review of HNP's License Amendment Request to Adopt National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants."

In accordance with the revised due date of February 05, 2010, Enclosure 1 provides HNP's responses to the NRC's requests for additional information.

As a result of these responses, certain parts of HNP's Transition Report Supplement 3, as submitted via Reference 2, also required revision. Enclosure 2 contains the updated portions as follows:

- Section 2.0, "Overview of Existing Fire Protection Program," new pages 9 and 10
- Section 4.1.1, "Overview of Evaluation Process," new pages 16 and 17

P.O. Box 165
New Hill, NC 27562

T> 919.362.2502
F> 919.362.2095

A006
NRR

- Section 4.6.2, "Overview of Post-Transition NFPA 805 Monitoring Program, new pages 44 and 45
- Table 4-5, "Fire Area Compliance Summary," new pages 52 and 53
- Table 4-8, "Required Suppression and Detection Systems," new page 67
- Table 4-8-1, "Required Automatic Suppression Systems," new pages 68 through 82
- Table 4-8-2, "Required Automatic Fire Detection Systems," new pages 83 through 98
- Section 5.0. "Regulatory Evaluation," new pages 99 through 105
- Section 6.0, "References," new pages 106 through 107
- Attachment A, "NEI 04-02 Table B-1 – Transition of Fundamental FP Program and Design Elements (NFPA 805 Chapter 3)," new pages A-1 through A-59 – **Security Related Information**
- Attachment E, "NEI 04-02 Table G-1 – Radioactive Release Transition," new pages E-1 through E-30
- Attachment H, "NEI 04-02 Frequently Asked Question – Summary Table," new pages H-1 through H-5
- Attachment L, "NFPA 805 Chapter 3 Requirements for Approval," new pages L-1 through L-2
- Attachment P, "Performance-Based Methods – NFPA 805 Chapter 3 – 10 CFR 50.48 (c)(2)(vii)," new page P-1
- Attachment R, "FSAR Changes," new pages R-1 through R-10

In accordance with 10 CFR 50.91(b), HNP is providing the state of North Carolina with a copy of this response.

This document contains no new or revised regulatory commitments.

Please refer any questions regarding this submittal to Mr. Dave Corlett, Supervisor – Licensing/Regulatory Programs, at (919) 362-3137.

I declare under penalty of perjury that the foregoing is true and correct. Executed on FEB 04 2010.

Sincerely,

Christopher L. Burton

Christopher L. Burton

CLB/kms

Enclosures:

1. Response to Second Request for Additional Information Regarding License Amendment Request to Adopt NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants"

2. Updated pages for HNP Transition Report – **Portions contain Security Related Information**

Cc:

Mr. J. D. Austin, NRC Sr. Resident Inspector, HNP
Mr. W. L. Cox, III, NC DENR
Mr. L. A. Reyes, NRC Regional Administrator, Region II
Ms. M. G. Vaaler, NRC Project Manager, HNP

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By letter dated May 29, 2008, as supplemented by letters dated November 14, 2008, December 11, 2008, August 13, 2009, August 28, 2009, and October 9, 2009, Carolina Power & Light Company (the licensee), now doing business as Progress Energy Carolinas, Inc., submitted a proposed amendment for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP or Harris).

The proposed amendment would transition the fire protection program to a performance-based, risk-informed program based on the National Fire Protection Association Standard 805 (NFPA 805), "Performance-Based Standard for Fire Protection For Light Water Reactor Generating Plants," 2001 Edition, in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.48(c) (10 CFR 50.48(c)). NFPA 805 allows the use of performance-based methods, such as fire modeling, and risk-informed methods, such as Fire Probabilistic Risk Assessment, to demonstrate compliance with the nuclear safety performance criteria.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's submittals and determined that it needs responses to the following requests for additional information (RAIs), whose numbering corresponds to that found in the August 6, 2009, RAI letter (ML092170715), in order to continue its review of the subject request.

HNP RAI 1-3.1

Implementation of Fire Protection Engineering Evaluations Post Transition

Attachment P, "Performance-Based Methods – NFPA 805 Chapter 3 – 10 CFR 50.48.(c)(2)(vii)," of the HNP NFPA 805 Transition Report (Harris Transition Report) does not fully reflect the structure and content of the approved version of Frequently Asked Question (FAQ) 06-0008, "Fire Protection Engineering Evaluations," Revision 9. Revision 9 addressed three major situations: conditions that are functionally equivalent to the NFPA 805 Chapter 3 requirement, conditions that can be shown to be adequate for the hazard for four specific sections of NFPA 805 Chapter 3, and conditions that can be shown to be acceptable using a bounding analysis approach through submittal to the NRC staff.

Attachment P appears to be a combination of the approved revision to FAQ 06-0008 (i.e., Revision 9) and the previous version (i.e., Revision 8). The attachment does address conditions that can be shown to be adequate for the hazard for the four specific sections of NFPA 805 Chapter 3 (i.e., Alarm and Detection Systems, Water-Based Suppression, Gaseous Suppression, and Passive Features). However, this attachment does not address functional equivalents or bounding evaluations for any of the conditions discussed.

The method discussion in Attachment P also refers to "referenced codes, standards, and listings," as well as "secondary features of the referenced codes, standards, and listings," with a *table that lists the NFPA 805 Chapter 3 sections that fit this description. However, the endorsed version of FAQ 06-0008 no longer includes this approach.*

As currently presented, Attachment P does not propose to allow functionally equivalent Fire Protection Engineering Evaluations to address NFPA 805 Chapter 3 issues. Without including this feature of FAQ 06-0008, future minor deviations that can be shown to be compliant would require a submittal to address. Please clarify whether or not this is the intent of Attachment P.

In the current form, Attachment P is not acceptable since a major portion of the approach refers to features of the previous revisions to FAQ 06-0008 that have not been endorsed. Absent revision of Attachment P to reflect other desired portions of Revision 9 to FAQ 06-0008, the NFPA 805 Safety Evaluation for HNP will only address conditions that can be shown to be adequate for the hazard for the four specific sections of NFPA 805 Chapter 3 (e.g., Alarm and Detection Systems, Water-Based Suppression, Gaseous Suppression, and Passive Features).

Response: Prior to the completion of transition, those licensees that have adopted the standard fire protection license condition are allowed to make certain types of changes as described without prior NRC approval as long as the changes do not adversely affect the plant's ability to safely shutdown in the event of a fire. The method used to perform these changes, as originally described in Generic Letter 86-10, has been referred to as Generic Letter 86-10 Evaluation, Fire Protection Engineering Evaluation, Fire Protection Engineering Equivalency Evaluation, etc. For the purposes of making minor changes to fire protection program attributes post-transition, these evaluations will be called Fire Protection Engineering Evaluations.

Fire Protection Engineering Evaluations (FPEEs) may be used to demonstrate compliance to NFPA 805 requirements using the three different types of FPEEs (functional equivalency, adequate for the hazard, and bounding approach) within the bounds defined in FAQ 06-0008 Revision 9 (Closure Memo dated 3-12-09, ML073380976). Two of these approaches, functional equivalency evaluations and adequate for the hazard, are allowable under the existing framework of NFPA 805 and do not require a submittal or prior NRC staff approval. The third approach, bounding analysis, does require prior NRC staff approval. The use of the bounding analysis approach requires the licensee to obtain prior NRC approval through the submittal of a license amendment request in accordance with 10 CFR 50.48(c)(2)(vii) and the addition of a section to the Fire Protection license condition addressing the change. Upon NRC approval, the licensee can make changes to the plant using FPEEs within the approved envelope for the bounding analysis performed to support the license amendment request.

HNP is not requesting the approval of a bounding analysis performance-based method.

A revised Attachment P of Transition Report is provided with this submittal.

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HNP RAI 2-16

Attachment H - Frequently Asked Question Summary Table

Attachment H to the Harris Transition Report (Nuclear Energy Institute (NEI) document NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," Frequently Asked Question – Summary Table) contains the FAQs that were used to clarify the guidance in Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," and NEI 04-02 during preparation of the HNP NFPA 805 License Amendment Request (LAR).

The NRC staff has identified a number of errors and omissions in the table included with Attachment H (e.g., referencing an incorrect Revision for FAQ 06-0008 on page H-2, out-of-date information for FAQ 07-0035 on page H-4, etc.). Accordingly, please review the FAQ Summary Table and ensure that the information presented is correct and current.

Response: An updated and validated version of Attachment H is provided with this submittal.

HNP RAI 2-17

Table B-1 – "Transition of Fundamental Fire Protection Program and Design Elements"

a) *Process*

1) *The process for documenting the "Complies via Previous NRC Approval" compliance statements described in the previous RAI responses (and as followed in the B-1 Table) does not match that described in the revised Harris Transition Report (as provided on October 9, 2009), Section 4.1, "Fundamental Fire Protection Program Elements and Minimum Design Requirements" (including Figure 4-1, "Fundamental Program and Design Elements Transition Process [based on NEI 04-02 Figure 4-2/FAQ 07-0036]"), which documents the original process. Additionally, Figure 4-1 documents the original process for handling compliance via Engineering Evaluations, and not the currently understood process. Accordingly, the licensee should ensure that the revised Harris Transition Report correctly documents the process used to populate the B-1 Table compliance statements.*

   In addition, please explain the use of the phrase "...or industry submittals..." in the description of excerpts provided to demonstrate previous NRC approval. Please clarify what documents are typically being referenced by this description, as well as where HNP uses these documents to support the NFPA 805 LAR.

Finally, the change to using excerpts from the HNP Final Safety Analysis Report (FSAR) instead of from the original submittal documents should be addressed, described, and justified. Please also address the potential issue of using FSAR excerpts in the B-1 Table in light of the "living" nature of the FSAR. Specifically, please provide revision information for the FSAR references used in the B-1 Table.

Response: The correct process for documenting "Complies via Previous NRC Approval" compliance statements is included in the revised LAR Section 4.1. This is the current process used in developing the B-1 Table and is consistent with the information presented in Figure 4-1, ensuring that the transition report correctly documents the process used to populate the B-1 Table.

Reference to "industry submittals" has been removed from the guidance in LAR section 4.1.1.

Attachment A (NEI 04-01 Table B-1) has been revised to eliminate the FSAR excerpts for those Chapter 3 elements where compliance is by previous regulatory approval.

The revised section 4.1 of LAR and Attachment A (NEI 04-01 Table B-1) are provided with this submittal.

2) *In revising the Harris Transition Report, Section 4.1, to correct the errors identified by the original RAI 2-12, the licensee has inadvertently created new errors in Section 4.1.1. Specifically, the licensee provides a bulleted list of five compliance strategies for B-1 Table entries, and then proceeds to detail the requirements for six strategies. Accordingly, please correct this discrepancy and perform a quality assurance check of Section 4.1 of the Harris Transition Report in its entirety.*

Response: As described in the response for RAI 2-17(a)(1) (above), the current process as applied in developing the B-1 Table has been included in the revised LAR Section 4.1. Specifically, the bulleted information provided in the later version of portions of Section 4.1.1 has been aligned to reflect the current process, ensuring that the transition report correctly documents the B-1 Table population method.

b) <u>*Follow-up to HNP RAI 2-15, Part 1*</u>

The current RAI response is insufficient. Firstly, the licensee proposed, during a September 4, 2009, RAI clarification call, to provide a revision to the original RAI response and to Attachment L, "NFPA 805 Chapter 3 Requirements for Approval," of the Harris Transition Report. However, neither was included in the October 9, 2009, submittal; instead, Attachment L was deleted. Please discuss the basis for this choice. In addition:

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1) *The B-1 Table entry for NFPA 805 Chapter 3 Element 3.5.16, regarding the dedication of the fire protection water supply system, currently describes a non-compliant condition as compliant via a "Complies with Clarification" compliance statement. However, the NRC staff cannot approve this element in its present state, despite deletion of Attachment L. If the licensee is seeking staff approval of the condition as currently described in the B-1 Table element, formal approval should be requested, and the licensee should:*

   - *Provide a regulatory basis (i.e., 10 CFR 50.48(c)(2)(vii) or 10 CFR 50.48(c)(4)) as well as an appropriate regulatory justification; and*

   - *Provide a level of technical detail and degree of technical justification equivalent to that which would be submitted for a stand-alone licensing action.*

   Response: A revised Attachment L with additional detail and a regulatory basis is provided with this submittal.

2) *The B-1 Table entry for NFPA 805 Chapter 3 Element 3.6.5, regarding the cross-connections of the seismic hose stations, currently depends upon a draft calculation to demonstrate compliance via a "Complies with Clarification" compliance statement. However, the NRC staff cannot approve this element based on a draft calculation. Accordingly, the licensee should justify compliance using a final calculation.*

   Response: A review of the seismic hose station/standpipes configuration shows that the hose station flow can be provided by either the A or B trains of Emergency Service Water (ESW). This is accomplished by manually aligning an isolation valve to provide water to the standpipe from the desired ESW train.

   Since single failure criteria do not apply for either fire or seismic scenarios, both trains can be assumed to be operable at the time of the event. In the event manual fire suppression is necessary, one train of ESW could be aligned to the fire protection system. The other train would be available and is capable of providing all necessary cooling functions for safe shutdown of the facility.

   Based on the above, it is concluded that alignment of the ESW system to provide water to the seismic fire hose stations does not degrade the capability of the ESW system to perform its intended function.

   Element 3.6.5 in Table B-1 has been revised to delete the reference to the draft calculation and to add the wording from the above RAI response.

c) *B-1 Table: Element 3.3.1.2 – Control of Combustible Materials*

Please provide a positive compliance strategy statement for the parent element in the B-1 Table that addresses compliance with the requirements to develop and implement procedures for the control of general housekeeping and transient combustibles.

Response: The parent statement ("Requirements/Guidance") for Element 3.3.1.2 states:

> "3.3.1.2* Control of Combustible Materials.
> Procedures for the control of general housekeeping practices and the control of transient combustibles shall be developed and implemented. These procedures shall include but not be limited to the following program elements:"

"Procedures for the control of general housekeeping practices and the control of transient combustibles have been developed and implemented as part of the HNP, FPP-001, Fire Protection Program Manual and administrative controls. FPP-001 includes reference to guidance regards, but is not limited to the following programmatic elements of NFPA 805, Section 3.3.1.2."

This statement has been added to Attachment A (NEI 04-02, Table B-1), "Compliance Basis".

d) *B-1 Table: Element 3.3.1.2.(6) – Regarding controls on the use and storage of flammable gases*

The reference to page 9-9 of NUREG-1038, "Safety Evaluation Report related to the operation of the Shearon Harris Nuclear Power Plant, Unit No. 1," appears to be incorrect. The information on this page of NUREG-1038 concerns the fuel-load handling system. If applicable, please correct this reference accordingly.

Response: The reference to page 9-9 of NUREG-1038 has been researched and found to be Supplement 2 (SSER 2), June 1985, Page 9-9. This has been included in the updated version of Attachment A (NEI 04-02, Table B-1).

e) *B-1 Table: Element 3.3.8 – Bulk Storage of Flammable and Combustible Liquids*

There is an apparent inconsistency in the code of record list located in Attachment R, "FSAR Changes," of the Harris Transition Report. NFPA 37, "Standard for the Installation and Use of Stationary Combustion Engines and Gas Turbines," which is identified in the B-1 Table, Element 3.3.8, as applying to the Diesel Oil Day Tanks, does not appear in the Attachment R list. Please rectify this apparent discrepancy.

Response: The reference to NFPA 37, "Standard for the Installation and Use of Stationary Combustion Engines and Gas Turbines," has been removed as part of the revision to Attachment A (NEI 04-01, Table B-1) to eliminate the FSAR excerpts for those Chapter 3 elements identified as "Complies by Previous NRC Approval" (Reference RAI 2-17(a)(1) response above). Since this was replaced with an excerpt from NLS-86-137 (C.5.d(4)), which also refers to NFPA 37, NFPA 37 has been added to Attachment R.

f) *<u>B-1 Table: Element 3.3.10 – Hot Pipes and Surfaces</u>*

The NFPA 805 Chapter 3 requirement for this B-1 Table element is as follows:

> *Combustible liquids, including high flashpoint lubricating oils, shall be kept from coming in contact with hot pipes and surfaces, including insulated pipes and surfaces. Administrative controls shall require the prompt cleanup of oil on insulation.*

HNP's B-1 Table entry for this element provides a compliance statement of "Complies" and gives HNP procedure FPP-004, "Transient Combustible Control," Revision 22, as the reference for where this compliance is documented. However, the NRC staff has examined the cited reference and does not agree that it fully documents compliance with this NFPA 805 Chapter 3 requirement. Accordingly, please rectify this inconsistency, paying particular attention to the final sentence of the associated requirement.

Response: HNP procedure FPP-004, "Transient Combustible Control," Revision 22, serves as the site administrative procedure for control of transient combustibles, such as combustible liquids and high flashpoint lubricating oils described in NFPA 805 Chapter 3, Element 3.3.10. Per internal B-1 Table Open Item 3.3.10-1, inclusion of the "Administrative controls shall require the prompt cleanup of oil on insulation" in plant or fleet level procedures during the implementation process, will be completed as described in LAR Section 5.4 within 180 days after NRC approval.

g) *<u>B-1 Table: Element 3.3.12 – Reactor Coolant Pumps</u>*

The NRC staff has determined that this parent element contains requirements related to seismic and other accident / off-normal conditions that are not addressed in the detailed sub-parts to this element. Accordingly, please provide a compliance statement and strategy that addresses the seismic (and other) requirements contained in the parent element. Previous NRC guidance on this issue can be found in Branch Technical Position (BTP) CMEB [Chemical Engineering Branch] 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," Position C.7.a(1)(e), regarding the fire protection requirements for the reactor coolant pump oil collection system in primary and secondary containment during normal operation.

Response: The B-1 table has been updated as follows:

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Compliance Statement
"Complies via Previous NRC Approval"

Compliance Basis
"The reactor coolant pumps will be equipped with an oil collection system. The oil collection system will be so designed, engineered, and installed that failure will not lead to fire during normal or design basis accident conditions and that there will be reasonable assurance that the system will withstand the safe shutdown earthquake."

"Based on its review, the staff concludes that fire protection inside containment meets Section C.7.a of BTP CMEB 9.5-l and is, therefore, acceptable."

Reference Document
"NUREG-1038, Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2 – Docket Nos. STN-50-400 and STN-50-401, Rev. Original, 11/1/1983, Page 9-52"

h) *B-1 Table: Element 3.4.2 – Pre-Fire Plans*

Please provide a reference to where Open Item No. 9, as identified in the August 28, 2009, HNP RAI response, is documented (as required by Figure 4-1), as well as a commitment regarding when it will be closed / completed.

Response: Internal ttool open item #9 for Table G-1 addresses the "Need to develop a Fire Pre-Plan for outside Yard areas to address Radioactive Materials Areas (RMAs) and Sea-Land type container storage." This open item will be completed during implementation of the new program. This includes peer reviews, procedure changes, process updates, and training to affected plant personnel to implement the NFPA 805 FP program. This will occur 180 days after NRC approval (reference LAR Section 5.4). HNP Fire Pre-Plan FPP-012-10-TRM, "Transient Radioactive Material Fire Pre-Plan," has been developed and was issued on November 10, 2009.

i) *B-1 Table: Element 3.6.4 - Standpipe and Hose Station Earthquake Provisions*

Please provide a description of the "alternative means of manual firefighting" described in the referenced compliance document as one of the bases for approval of the granted deviation. This detail should be provided for each fire zone where the deviation is credited.

Response: Although the use of a specifically defined "alternative means of manual firefighting" was not presented by HNP in the cited Compliance Basis Approval Document (NLS-86-315 Deviation Request), this term was included by the NRC in NUREG 1038 HNP SSER 4 Section 9.5.1 response.

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The three items originally presented as justification for the deviation as discussed in SERIAL: NLS-86-315, dated August 25, 1986, submittal were:

"The Company considers this deviation justified because:

- the above redundant safe shutdown equipment is separated from each other by three-hour rated barriers, which are Seismic Class I structures,
- these areas are provided with non-seismic fire protection systems, and
- the combustible loading in these areas is considered low, except in the case of the diesel day tank and storage tank area where the enclosures are Seismic Class I or ASME Section III."

While the term "alternative means of manual firefighting" is not specifically defined, the "alternate method of manual firefighting" that would support the NUREG 1038 HNP SSER 4 Section 9.5.1 response is the use of fire hoses from outside fire hydrants. This is discussed in the applicable Pre-Fire Plans for the affected HNP buildings.

Affected HNP Plant Fire Areas:
Plant Location:
a) Diesel Generator Building
b) Diesel Fuel Oil Storage Building
c) Emergency Service Water Intake Structure

Fire Area:
a) 1-D-DGA, 1-D-DGB, 1-D-DTA, 1-D-DTB
b) 12-O-TA, 12-O-TB, 1-O-PA, 1-O-PB, 5-O-BAL
c) I2-I-ESWPA, I2-I-ESWPB

In the case of the Diesel Generator Building, interior fire hose stations arranged off of the nonseismic standpipe system in that building would be used as well.

j) <u>*B-1 Table: Element 3.7 – Fire Extinguishers*</u>

The August 28, 2009, HNP RAI response letter identifies an Open Item that would extend a code compliance evaluation to additional plant buildings. However, this situation is incorrectly identified in the associated B-1 Table element as a compliant configuration via a "Complies with Clarification" compliance statement. Accordingly, the licensee should correct the associated B-1 Table entry, appropriately document the Open Item, and provide a commitment regarding when it will be closed / completed.

Response: Compliance statements provided in the Table B-1 represent the as-built conditions as operated conditions of the plant at the time of program implementation. As discussed in LAR Section 5.4, implementation of the new program will include peer reviews, procedure changes, process updates and training to affected plant personnel to implement the NFPA 805 FP program. This will occur 180 days after NRC approval (issuance of the SER).

k) *B-1 Table: Element 3.11.3 – Fire Barrier Penetrations*

1) *It appears that HNP has not completely documented all of the appropriate deviations for this element. One example is the fire barrier protective device deviation detailed in NLS-86-219, "Fire Protection – Deviations from BTP 9.5-1," which is not included in this element. Please ensure that all appropriate deviations are referenced in the B-1 Table.*

   Response: NUREG 1038, Supplement 4, Page 9-5 provides the previous NRC approval for the deviations described in NLS-86-219 and associated with NFPA 805 Chapter 3, Element 3.11.3. This compliance basis and reference has been incorporated into the LAR Attachment A (NEI 04-02, Table B-1).

2) *It appears to the NRC staff that the second referenced deviation does not apply to this element. Instead, it appears to be a NFPA 805 Chapter 4 separation issue (i.e., a missing barrier). If this is the case, the second deviation should be removed. If it is not the case, it appears that similar deviations should be documented here as well.*

   Response: The second referenced deviation and associated compliance basis and reference information have been removed and replaced with the corrected deviation information as described in RAI 2-17(k)(2), above.

HNP RAI 3-23.m.1

Information Notice (IN) 92-18 – "Potential for Loss of Remote Shutdown Capability During a Control Room Fire"

In the August 28, 2009, HNP RAI response letter, the licensee's responses to HNP RAI 3-23.m and HNP RAI 3-31 state that a modification will ensure that following control transfer from the control room to the auxiliary control panel (ACP) in the event of a fire requiring the use of the ACP, valves 1AF-137, 1AF-143, and 1AF-149 are isolated from the 305 foot elevation and a clean fuse is available if the primary fuse has already been blown.

However, the responses do not discuss any potential vulnerability with respect to fire-induced damage to the valves as a result of by-passing the torque and/or limit switches as described in NRC IN 92-18. Accordingly, please describe how these valves were evaluated against the *issues discussed in IN 92-18. If the valves are indeed susceptible to damage prior to control transfer, provide a performance-based analysis demonstrating the acceptability of the risk, defense-in-depth, and safety margins for the affected valves.*

Response: This referenced EC is not addressing the IN 92-18 concern directly. Defense-in-depth is being increased by doing the following:

- The addition of incipient detection per EC 69501 to reduce the probability of a fire that could result in evacuation of the MCR.
- Rather than only modifying the credited flow path by protecting 1AF-143, all three flow path isolation valves (1AF-137, 1AF-143 and 1AF-149) are being modified in EC 70895. With this EC, a fire in the MCR cannot affect the circuit and the control circuit from the ACP will function following transfer to the ACP.
- Although not credited or fully protected, additional defense in depth is provided by having the controls for both MD AFW Pumps and associated valves transferred to the ACP. AOP-004 directs stopping the MDAFW pumps only after the step to "**CHECK** TDAFW supplying SGs".

From a risk standpoint, no credit for Alternate Shutdown is given in the PSA except for the case where the MCR is abandoned due to environmental reasons.

HNP RAI 3-24.1

Limiting Conditions for Operation (LCOs) for New Equipment

HNP RAI 3-24 requested a list of structures, systems, and components (SSCs) not currently covered by existing Technical Specification (TS) LCOs, and requested that the licensee propose TS LCOs or describe the administrative controls that will assure availability of the new equipment, including required compensatory measures to be taken when the equipment is not available. The response provided by the licensee stated that FPP-013, "Fire Protection – Minimum Requirements, Mitigating Actions, and Surveillance Requirements," will continue to track safe shutdown equipment not covered by the HNP TS. However, the NRC staff has the following concerns regarding the response to HNP RAI 3-24:

a) The Alternate Seal Injection System (ASI) has not been characterized as "safe shutdown equipment" in the NFPA 805 LAR. It is not clear whether the ASI is being installed as part of the Fire Protection Program or as Balance of Plant equipment since it is being credited for risk reductions in both areas. Accordingly, for the new ASI equipment, please identify the administrative program that will address the system (Fire Protection Program, Technical Requirements Manual, Maintenance Rule, etc.) and provide a description of the *administrative controls that apply to it, including any LCO-type limitations, allowed out of service time (AOT), and required actions when the AOTs are exceeded.*

Response: The Alternate Seal Injection System (ASI), installed via EC 70350, will not be TS required equipment. This equipment will be controlled by FPP-013, "Fire Protection – Minimum Requirements, Mitigating Actions and Surveillance Requirements," as is other non-TS equipment, and will have allowed out of service time as necessary to support the PSA assumptions. The allowed out of service time and compensatory actions have not been developed at this time.

b) *For those SSCs that are covered by FPP-013, please provide a description of the administrative controls applied (i.e., exactly what does "track safe shutdown related equipment not covered by HNP TS" mean?).*

Response: The administrative controls applied to SSCs required for Safe Shutdown such as Sound Powered Phone System, SSD Emergency Lighting System and SSD Support Systems and Equipment are contained in FPP-013, "Fire Protection – Minimum Requirements, Mitigating Actions and Surveillance Requirements," Section 8.7. Key attributes include Operability, Mitigating Actions, and Mode Applicability.

SSD support systems and equipment that have been determined to be required, are non-safety related, have non-safety related controls or have specific SSD/Fire Protection requirements, are listed in Attachments 5 through 7 of FPP-013. Safe Shutdown Equipment not currently covered by Technical Specifications is provided in FPP-013, Attachment 7, "SSD Support Systems and Equipment," which will be revised as necessary to include the additional equipment added by the NFPA 805 transition.

c) *Please describe the relationship, if any, between FPP-013 and the NFPA 805 Monitoring Program. Is there any potential for a plan to adjust LCO and/or AOT time frames based on Fire Probabilistic Risk Assessment (PRA) results?*

Response: FPP-013, "Fire Protection – Minimum Requirements, Mitigating Actions and Surveillance Requirements," will be revised during the implementation process to capture risk-informed insights for those FP SSCs identified in the NFPA 805 Monitoring Program. Compensatory action will be defined based on risk significance, operational need and management discretion for those same SSCs. Since the NFPA 805 Monitoring Program does not impact other new or existing plant SSCs, it would not create a change in LCO and/or AOT times for those systems.

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HNP RAI 3-65.1

Electrical Coordination (Installed Fuses)

In the August 28, 2009, HNP RAI response letter, the licensee's response to HNP RAI 3-65 states that the plant was originally designed with general coordination to ensure that a fault on a cable would not damage the cable itself. The response goes on to state that a fuse control program is in place to administratively control fuse replacement such that the original fuse coordination is maintained. Please address the following concerning this response:

a) How long has the fuse control program been in place?

b) Has the fuse control program been in place since construction?

If the fuse control program has been in place since construction, please state that HNP is in compliance with NFPA 805 Section 2.4.2.2.2.

If the fuse control program has not been in place since construction, please describe how the licensee plans to provide reasonable assurance that the fuses currently installed will provide the required level of protection.

Responses:

a) As referenced in Licensee Event Report (LER) 2001-002-00, HNP instituted a fuse program in 1987 due to concerns with fuse coordination at different plants and other fuse concerns unrelated to the LER issue.

b) While the fuse program has not been in place since construction, it was implemented in 1987 (initial fuel load). Additionally, many of HNP's fuses have been field verified as discussed in non-cited licensee identified violation 400/93-25-01, "Failure to properly control fuses" (NRC Inspection Report 50-400/93-25).

   - In November 1990, field verifications of 740 fuses in the safety-related 125 VDC system were performed with 64 replaced (PCR-5549);
   - During 1992 and 1993, field walkdowns of 1260 fuses were performed to support the safety-related AC fuse schedule with replacement of 136 (PCR 6450)

   ESR 94-00072 converted the existing fuse schedule to EDBS (the predecessor to EDB, the Equipment Database). In this ESR, an additional fuse walkdown was performed that completed the review of all Safety Related fuses and approximately 1218 out of 1409 non-safety related fuses. The majority of fuses not checked were either not in associated circuits so did not impact coordination related to safe shutdown, were known to have been replaced later during the LK Breaker replacement project, or were the small glass type fuses that should not impact associated circuits. Other fuses located in Security, Warehouses, office area, etc., were also checked but not included in this count.

Currently, each time an equipment clearance removes a fuse, that fuse is verified against the approved fuse listed in the EDB. Any fuse found not as specified in EDB is replaced with the proper fuse and a notification sent to the Fuse Program manager per PLP-628, "Plant Fuse Control Program for 1E and non-1E Applications." This requirement has been in OMM-014, "Operation of the Work Coordination Center," since 1994 (OMM-014, Revision 2). This provides reasonable assurance that the fuses currently installed will provide the required level of protection.

A summary of fuse replacements performed in accordance with the Fuse Control Program procedure (PLP-628) and other HNP procedures found that notifications of approximately 321 fuses were received. Approximately 50% of these were for the replacement of Renewable Fuses (REN) with Non-Renewable Fuses (NON) for reliability purposes. Other identified replacement reasons included blown fuses, same fuse finally installed, installed fuse is no longer available and either a different type of fuse is specified or a better fuse has become available and is specified when the fuses are replaced. In four cases the installation of a smaller fuse would not adversely impact coordination with the upstream device. In addition, these were in locations that were not associated circuits.

Since 1994, the only fuse issue impacting safe shutdown was the improper design of fuses for the Pressurizer PORV block valves. This was detailed in LER 2001-002-00 and did not involve maintenance of the existing fuse program.

HNP RAI 3-69

Identified Deviation for Exterior Penetrations Lacking Fire Dampers

In Section 2.2, "NRC Acceptance of HNP Fire Protection Licensing Basis," of the Harris Transition Report, several deviations are noted as having been previously granted by the NRC. Deviation 8 is identified on Page 10 of the Harris Transition Report as "a deviation to the requirements of NFPA-90A, ["Standard for the Installation of Air-Conditioning and Ventilating Systems,"] for providing fire dampers at exhaust and intakes at external walls, stairs, and roofs." It is also noted as a licensing action to be carried forward on Page K-38 of Attachment K, "Existing Licensing Action Transition," of the Harris Transition Report. Attachment K identifies the affected fire areas as Switchgear Rooms A and B in the Reactor Auxiliary Building and Emergency Service Water Pump Rooms A and B.

In the entry on Page 10 of the Harris Transition Report, HNP letter NLS-86-219, "Fire Protection – Deviations from BTP 9.5-1," is identified as providing supplemental information *regarding the deviation. However, this letter does not appear to contain this information. Accordingly, please provide a corrected supplemental reference for this deviation.*

Additionally, please provide details about where (wall or roof, as well as other location details) the penetrations pierce the exterior membranes of the affected fire areas. Include the relative spacing between the penetrations in the related pairs of fire areas (i.e., what is the distance between the openings in the two switchgear rooms as well as between the openings in the two pump rooms). In the case of the switchgear rooms, please provide the distance between the exterior openings and the nearest fire area of concern from an exposure fire standpoint (i.e., Turbine Building). Based on the provided location details, also provide a technical justification for the acceptability of these openings not having appropriately rated fire dampers installed.

Response: NLS-84-090 is Fire Damper deviation request (prior to NLS-86-137). LAP-83-326, LAP-83-485, and NLS-86-188 have the same information as NLS-86-137.

NLS 86-219 is for Bus Duct deviation in 1-A-SWGRA and 1-A-SWGRB. This does not deal with opening to the exterior of the buildings and has been removed from LAR Section 2.2(8).

For 1-A-SWGRA and 1-A-SWGRB, the openings are for roof mounted intakes/exhaust for AH-12 and AH-13 and associated smoke purge. For 1-A-SWGRA, the normal intake has been abandoned and blanked off, and the opening used as the alternate access path for 1-A-BAL-J. For AH-13, the damper is locked closed due to an inability to maintain proper Control Room differential pressure. The remaining openings are for smoke purge inlet and exhaust and battery room exhaust. These are all in missile/tornado protected enclosures. The 42 inch exhaust air reliefs located on the roof of the SWGR rooms are away from the Turbine Building by greater than 55 feet. The closest interaction between fire areas is approximately 34 feet with a wall of the Main Steam Pipe Tunnel (1-A-BAL-K) in between. The two 42 inch valves for the SWGR Rooms are approximately 68 feet apart with 1-A-BAL-K in between. (Reference 8-G-0506 S01). The Turbine Building is free standing, with a gap between it and the RAB.

The RAB also has roof mounted or RAB wall mounted openings that are located above RAB 305' (lowest level of the RAB Roof), well away from other buildings. The opening which is closest to any building is the Main RAB intake. The Main RAB intake is located in a tornado/missile protected enclosure with the intake approximately 32 feet from the Turbine Building. (Reference 7-G-1310, 1311). The top elevation of the Turbine Building is the 314 foot elevation.

The ESW structures are located remote from other building. The buildings have two ventilation flow paths. The E-88s draw air from the pump room and exhaust to a missile protected/tornado protected enclosure in the center of the building. These intakes are approximately 40 feet apart. AH-86s draw air from the North of the building through a missile protected/tornado protected enclosure to supply the MCC area. These intakes are approximately 29 feet apart. (Reference 8-G-0568 and 6-SK-E-0542 S27).

Each of these intakes opens into an area with low combustible loading.

Based on the above, we believe that there is no need for fire dampers at the exterior of the buildings at HNP.

HNP RAI 3-70

Table 4-8 – "Required Suppression and Detection Systems"

The NRC staff has identified a number of apparent deficiencies in Table 4-8 of the Harris Transition Report. Accordingly, please address the following:

a) *It does not appear that HNP considers the thermal detection systems installed in conjunction with pre-action and multi-cycle suppression systems to be installed detection systems. Some example fire zones where this is the case are: 1-A-1-PA (Reactor Auxiliary Building (RAB) Elevation 190'), 1-A-3-COR (RAB Elevation 236'), 1-A-1-PB (RAB Elevation 190'), 1-A-3-COMC (RAB Unit 1 – Analysis Area D), 1-D-DTA (Diesel Generator Fuel Oil Day Tank A Enclosure), 1-D-DTB (Diesel Generator Fuel Oil Day Tank B Enclosure), 1-G-314 (Turbine Generator Building), and 5-F-2-FPC (Fuel Handling Building Fuel Pool Heat Exchangers).*

   It is the NRC staff's position that these thermal detectors constitute detection systems. Additionally, these thermal detectors are called out by HNP in deviation requests (e.g., the penetration protection deviation request identified in HNP letter NLS-86-219) as installed detection systems. Given this apparent inconsistency, HNP should perform an extent of condition review to ensure that all installed fire protection systems are correctly identified.

   Response: The "Required Suppression and Detection Table" (LAR Table 4-8) has been revised to create two individual tables, one each for automatic suppression and detection systems. The detection systems that are installed as actuation systems for suppression systems have been included in the "Required Fire Detection Systems" table as requested. These tables will be finalized as part of the HNP NFPA 805 program implementation process as described in LAR Section 5.4. Currently these thermal detection systems are included in the HNP Plant Procedure FPP-013, "Fire Protection – Minimum Requirements, Mitigating Actions and Surveillance Requirements," Attachment 4, "Fire Detection Instrumentation."

b) *A number of fire protection systems identified in HNP-retained licensing actions are not denoted in Table 4-8 as required systems, or do not appear in the table at all.*

   Examples from the deviation identified in HNP letter NLS-86-219:

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- *Table 2 of NLS-86-219 identifies an ionization detection system installed in fire zone 12-A-7-HV (Heating, Ventilating Room, elevation 324 feet), which is missing dampers FD-W-10, -11, -12, and -13. However, Table 4-8 of the Harris Transition Report states that the detection system installed in the fire zone is not required.*

- *Table 2 of NLS-86-219 identifies an ionization detection system installed in fire zone 5-F-2-DEC (Decontamination Area and Transfer Tank, elevation 236 feet), which is missing damper FD-73. However, Table 4-8 of the Harris Transition Report states that there are no detection systems installed in the fire zone.*

- *Table 3 of NLS-86-219 identifies a multi-cycle suppression system and a thermal detection system installed in fire zone 1-G-261 (Turbine Generator Building, Ground Floor, elevation 261 feet), which is missing bus duct penetration seals BD-5, -6, -9, -10, -17, and -18. However, Table 4-8 of the Harris Transition Report states that the suppression system is not required.*

Given these apparent inconsistencies, HNP should perform an extent of condition review to ensure that all of the fire protection systems identified in the licensing actions being carried forward are correctly identified and dispositioned.

Response: The fire suppression and detection systems within the NFPA 805 defined Power Block were reviewed for inclusion in the reformatted "Required Automatic Suppression Systems" Table (LAR Table 4-8-1) and the "Required Automatic Fire Detection Systems" Table (LAR Table 4-8-2). The review included all Licensing Deviations, EEEE's and GL 86-10 evaluations that are credited in the HNP NFPA 805 LAR and Transition Report.

The issues noted in this question for fire zones 12-A-7-HV (first bullet) and 1-G-261 (third bullet) have been resolved in the Required Systems tables by noting the appropriate systems as "Required Systems."

The following discussion addresses the issue presented in the second bullet associated with fire zone 5-F-2-DEC in NLS-86-219. Upon further review of the fire zone, it was determined that there is no automatic fire detection installed in this fire zone and the notation of ionization detection in this fire zone in NLS-86-219 was in error. After reviewing the deviation, it was noted that this fire damper FD-73 is located in a barrier that is within the Fire Area 5-F-BAL envelope. This fire damper is in a barrier between two fire zones for which there is no fire rating requirement. Therefore, this damper should not be considered in the original list in NLS-86-219. Further review of the deviations noted that this same fire zone (5-F-2-DEC) was listed in NLS-84-471 as having ionization detection credited to support a discrepancy with Fire Door 3312. As discussed above, further review of the fire zone determined that there is no automatic fire detection installed in this fire zone and the notation of there being ionization detection in this fire zone in NLS-84-471 was in error. The basis for acceptance for this fire door should have been the negligible fire loading on both sides of the door and no SSA equipment within 20 feet of the door, which is a similar basis for several other fire doors included in NLS-84-471 and approved in the HNP SER. Considering the transition plan for HNP to NFPA 805, automatic fire detection for this fire zone would not be a requirement of Chapter 3 or 4 of NFPA 805. Therefore, the current situation, considering the errors noted and the justification provided, is considered acceptable without automatic fire detection installed in this fire zone.

c) *Please describe HNP's process for demonstrating compliance with the NFPA 805 Chapter 3 requirements for systems determined to be required (as denoted by one of the "Yes" criteria in Table 4-7 of the Harris Transition Report, which shows the approach to address the term "required" system per the NFPA 805 requirements), but which are not currently evaluated in an existing code compliance evaluation referenced in the appropriate B-1 Table element.*

The required Suppression and Detection Systems meet a standard of design, construction, maintenance, inspection and testing that is consistent with the applicable NFPA code(s). Since the majority of the required Systems at HNP have already been specifically reviewed for code compliance, a specific NFPA Code Compliance Review does not significantly demonstrate compliance. Only minor deviations from code requirements were found with the majority accepted "as-is" via engineering evaluation.

There is reasonable assurance of this conformance by virtue of the methodology normally used in the industry to control the design conformance, quality and ongoing performance of all NFPA code systems. It does not appear that expansion of these calculations for additional systems, many of which have been installed since original construction, would provide significant added benefit or safety beyond that currently in place. Original system design and installation was monitored by detailed specification development and adherence and internal quality assurance/control programs, along with review and approval by outside insurance underwriters.

The plant modification process controls what changes can be made to insure that the code requirements are maintained. Internal HNP programs such as the Engineering Program self-assessments and System Engineering monitoring and trending efforts provide continuous oversight of the systems to ensure their design and performance are maintained. These aspects, in combination with original plant construction and on-going system maintenance, provide assurance that the systems continue to meet the original NFPA code requirements and provide a suitable approach for demonstrating compliance with the NFPA 805 Chapter 3 requirements for systems determined to be required.

d) *There may be more than one suppression or detection system installed in a fire zone. Given that this may be the case, when Table 4-8 has a "Yes" in the "Suppression a*

Required System?" or "Detection a Required System?" field, does that indicate that all systems installed in the fire zone are required? If it does not, please list all installed systems and indicate which ones are required for each fire zone.

Response: For fire zones provided with more than one automatic suppression or detection system where that zone has a designation of "system required", the individual system(s) shown as required will be denoted in the supporting NFPA 805 Code Compliance calculation. This calculation will have an attached spreadsheet providing additional clarifying details for the required systems. There will be suitable "Comments" columns to provide the requested details along with any necessary clarifications for long term understanding of the basis for the required systems and for future clear recognition of which systems and/or system types are required. This code compliance calculation will be completed as part of the HNP NFPA 805 program implementation process as described in LAR Section 5.4.

e) *Where fire protection systems are required, HNP should document the reasons (i.e., all of the applicable "Yes" criteria from Table 4-7) that the systems are required for the fire zone.*

Response: This change has been noted in the reformatted LAR Tables 4-8-1 and 4-8-2.

f) *For fire zones where either a detection or suppression system is not installed, Table 4-8 contains a shaded space in the "Required System?" column. HNP should correct these columns to accurately document whether or not a system is required in the subject fire zone.*

Response: This change has been noted in the reformatted LAR Tables 4-8-1 and 4-8-2.

HNP RAI 3-71

Use of the Storm Drainage System for Suppression Water Runoff

Attachment C, "NEI 04-02 Table B-3 – Fire Area Transition," of the Harris Transition Report credits the use of the Storm Drainage System for suppression water runoff in the section titled "Fire Suppression Activities Effect on Nuclear Safety Performance Criteria" for the following fire areas:

1-A-BAL-A2	*1-A-BAL-B3*	*1-A-BAL-J*
1-A-BAL-A3	*1-A-BAL-B4*	*1-A-CSRA*
1-A-BAL-A4	*1-A-BAL-B5*	*1-A-CSRB*
1-A-BAL-B1	*1-A-BAL-C*	
1-A-BAL-B2	*1-A-BAL-D*	

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Please describe how the Storm Drainage System is used in this context. In the discussion, please also explain how the use of the Storm Drainage System meets the requirements for achieving the Radioactive Release Performance Criteria.

Response: For non-radioactive areas such as the switchgear rooms (1-A-SWGRA and 1-A-SWGRB), office areas in the Waste Processing Building or in out buildings (Diesel Building, Diesel Fuel Oil Storage Building, ESW Pumping and Screening Structures), the approach for firefighting is to bring hoses in from outside areas and allow the water to exit through the storm drains.

For areas in the Reactor Auxiliary Building, Containment, Fuel Handling Buildings, or Waste Process Buildings that are potentially radioactive and in which the suppression water may be contaminated, the water is handled by floor drains. The water is captured in tanks and processed as normal, with monitoring prior to release.

For the Turbine Building, water is collected and directed to the oily waste separator via a radiation monitor that on high alarm will shut the isolation valve and trip the pumps.

The Pre-Fire Plans state for the potentially contaminated areas floor drains are available.

Per DBD-317, "Water-Based Fire Suppression System," Rev. 5:

4.5.2 Water drainage from areas that may contain radioactivity should be collected, sampled, and analyzed before discharge to the environment. (ref. 6.8.3, Section C.5.a(14))

Compliance:

Water drained from areas having a potential for radioactive contamination will be collected, sampled and analyzed before discharge to the environment. (ref. 6.12.2, Section C.5.a(14))

Reference 6.8.3 is Branch Technical Position CMEB 9.5-1 (NUREG-0800), "Guidelines for Fire Protection for Nuclear Power Plants," Revision 3, July 1981

Reference 6.12.2 is Fire Protection Program Review to Branch Technical Position CMEB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants, Revision 3 to NRC Question 280.1 (Enclosure to CP&L Letter NLS-86-137 from S. R. Zimmerman (CP&L) to Harold R. Denton (NRC) dated May 7, 1986)

Per DBD-118, "Floor Drain Storage and Treatment System," Rev. 4:

1.0 The Floor Drain Storage and Treatment System (FDSTS) collects, stores, processes and monitors the following potentially radioactive liquid wastes for reuse or discharge: drainage from the Radioactive Floor Drain Systems, surges in the Equipment Drain Treatment Systems, and Laundry and Hot Shower Systems, decanting water from the Decanting Tanks, and condensate overflow from the Volume Reduction System.

2.2.8 Liquid discharge to the environment is automatically terminated and alarmed in the Main Control Room when predetermined radioactivity limits are exceeded. The Waste Monitor Tanks discharge is monitored by radiation monitor REM-3541 which sends a signal to close system isolation valve 7WL-D355-1 & 2 on detecting radioactivity above a preset level. The system is in compliance with Regulatory Guide 1.21 since the radiation monitor is located at the discharge point. In addition, an interlock system is provided to automatically isolate the liquid discharge in the event that dilution flow afforded by the cooling tower blowdown falls below a preset value.

HNP RAI 3-72

Control Room Delta Risk

Table G-2, "Disposition of Pre-Transition Operator Manual Actions," of the Harris Transition Report lists 10 recovery actions required for fires that occur in Fire Area 12-A-CR – Main Control Room. The use of these 10 recovery actions to achieve the nuclear safety performance criteria does not meet the deterministic requirements of NFPA 805 Section 4.2.3, "Deterministic Approach." However, Fire Area 12-A-CR credits the use of the performance-based approach in accordance with NFPA 805 Section 4.2.4, "Performance-Based Approach."
Since a Fire PRA is being utilized to assess the risk of variances from the deterministic requirements (VFDRs), NFPA 805 Section 4.2.4.2, "Use of Fire Risk Evaluation," applies to the assessment of the performance-based alternative. Accordingly, the unprotected cables associated with the components being addressed by the recovery actions required for Fire Area 12-A-CR should be considered VFDRs since they do not meet the deterministic requirements.

Although a Plant Change Evaluation may not be required because the existing deterministic licensing basis has approved the use of the operator manual actions that are being transitioned as recovery actions, a Fire Risk Evaluation in accordance with NFPA 805 Section 4.2.4.2 is required. The Fire Risk Evaluation performed in accordance with NFPA 805 Section 4.2.4.2 should compare the risk associated with implementation of the deterministic requirements with the risk associated with the proposed alternative (i.e., use of the recovery actions).

In addition, NFPA 805 Section 4.2.4 requires that when the use of recovery actions has resulted in the use of this approach (i.e., the performance-based approach), the additional risk presented *by their use shall be evaluated. Section G.5.3.2, "Results," of the Harris Transition Report states that: "Functional failures are not recovered in the Fire PRA by modeling the recovery actions for alternative shutdown fire scenarios. Therefore, the risk in the applicable areas provides a bounding assessment. Attachment Y[, "NFPA 805 Transition Risk Insights,"] provides the details of the fire area analysis."*

This passage appears to be stating that neither individual nor fire area delta risk calculations were performed to assess the additional risk of alternative shutdown recovery actions in Fire Area 12-A-CR. Using the bounding analysis approach described should result in the delta risk being set equal to the fire area risk. For the control room, this would result in the delta risk being equal to the full fire area fire risk (i.e., 4.48E-06 core damage frequency (CDF), 4.44E-07 large early release frequency (LERF)). This delta risk is not only the additional risk presented by the use of recovery actions, but also the delta risk of the associated VFDRs for not protecting the cables for those components associated with the recovery actions.

Given the above discussion, the licensee should treat the delta risk information as follows: 1) the delta risk should be added onto the risk change due to NFPA 805 transition as discussed in Section Y-2, "Risk Change Due to NFPA 805 Transition," of the Harris Transition Report; 2) the delta risk should be discussed in the Fire Area 12-A-CR section of Attachment Y, which currently states that the delta risk for the control room is zero; and 3) the control room risk should be included as the delta risk for VFDRs in the Fire Area 12-A-CR portion of Attachment Y. Accordingly, please address these concerns for Fire Area 12-A-CR.

Response: Recovery actions credited in the PRA for the main control room consist of those operator actions taken after abandonment that are not performed at the alternate shutdown panel. This risk is being estimated for Harris by using the entire risk due to main control room abandonment as a surrogate for this subset of actions. This approach produces a bounding value.

Abandonment was only credited in the main control room when it was determined that habitability thresholds would be reached prior to failures that could result in core damage. (i.e., the PRA did not credit recovery of equipment after it was determined failed due to the fire). Based on the Fire PRA analysis used for the LAR, the following control room scenarios include credit for control room abandonment actions:

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Panel	Scenario	Description	CDF	LERF
MCB-D1	1	Fire induced station blackout. Power recoverable via outside control post fire shutdown strategies.	4.75E-07	4.75E-08
x-MCR	1	Postulated fire at any location except panel MCB-A1 with failure to suppress before abandonment criteria exceeded. Including transient combustible contribution.	1.61E-08	1.61E-09
x-MCR	3	Postulated fire at MCB-D2. Results in loss of MCR Ventilation System.	8.30E-09	8.30E-10
		Total	4.99E-07	4.99E-08

Although Recovery Actions have also been identified in the "PIC" room (aka FC03 or 12-A-CRC1), no credit has been applied in the PRA. Additionally, because these recovery actions are not associated with VFDRs, no delta CDF has been determined in the baseline analysis. In order to determine the risk of these actions, the cables associated with these actions have been identified and this area reanalyzed assuming the associated cables are protected. This eliminates the need for the recovery actions. A delta CDF has been developed that bounds the risk of these actions. The delta CDF and delta LERF have been calculated as 5.09E-08/yr and 6.19E-08/yr respectively. The LERF value is obviously conservative, as it cannot realistically be larger than the CDF and should be significantly less than CDF. The previously submitted response to RAI 5-1 (reference SERIAL: HNP-09-084, dated August 13, 2009) discusses the LERF calculation issue in this compartment. Neglecting the conservatism in the calculation, the results can still be assessed as acceptable.

Adding the delta control room estimate for abandonment actions to the previous VFDRs results in the following table:

	dCDF	dLERF
Total VFDRs	1.13E-06	5.36E-08
MCR Recovery Actions	4.99E-07	4.99E-08
PIC Room Recovery Actions	5.09E-08	6.19E-08
Internal Events risk reduction for seal LOCA MOD	-1.71E-06	-9.00E-08
Total	-3.01E-08	7.54E-08

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HNP RAI 3-73

Suppression Systems Credited in Fire PRA But Not Classified as Required in Table 4-8

The American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) combined PRA standard includes requirements for the consideration of fire impacts to structural steel (SR FSS-F1). Large lubricating oil fires have been evaluated for the potential to impact structural steel in the HNP Turbine Building.

Attachment 8 of Calculation HNP-F/PSA-0079, "Harris Fire PRA – Quantification Calculation," Revision 1 states: "Fire scenarios involving significant oil hazards in the Turbine Building would be the only fires capable of causing significant damage. Dedicated water spray (deluge) suppression systems are provided near the large oil related fire hazards in the Turbine Building including the turbine lube oil reservoir, the condensate pumps, the steam generator feed pumps, the condensate booster pumps and the hydrogen seal oil unit. These suppression systems provide assurance that the fire affects remain localized to the hazard."

Attachment 8 of Calculation HNP-F/PSA-0079, Revision 1, goes on to say: "Going beyond the prototypical fire scenario described in the ANS Standard, a review of ignition sources (oil hazards) with the five highest heat release rates noted in the HNP Turbine Building Fire PRA analysis identified oil spills associated with the [Digital Electro-Hydraulic] (DEH) Pump Skid, and Condensate Booster Pumps as the largest contributors. In both cases the water spray suppression systems provided for the equipment, and the inherent venting capability of the open Turbine Building design will prevent catastrophic structural steel failure, and will limit worst case damage to the localized area(s) of the involved equipment."

The last sentence implies that the major oil hazard suppression systems in the HNP Turbine Building are credited as the means of addressing the risk for structural collapse of the building.

Section 4.8.4, "Required Systems and Features," of the Harris Transition Report provides the process used to identify those fire detection and suppression systems that are required to meet the nuclear safety performance criteria. Table 4-8 has been provided to document the results of this review. Table 4-8 includes some, but not all of the suppression systems currently installed in the HNP Turbine Building. For instance, Fire Zone 1-G-240 – Turbine Generator Building Basement, and Fire Zone 1-G-314 – Turbine Building Operating Floor, have installed suppression systems that are not required to meet the nuclear safety performance criteria.

This appears to be inconsistent with the statements made in calculation HNP-F/PSA-0079, Revision 1, that imply the major oil hazard suppression systems in the turbine building are credited to prevent catastrophic structural steel failure, which is a requirement of the ASME/ANS combined PRA standard. For example, based on the plant layout drawings observed during the Harris site audit, it appears that the condensate pumps are located in a pit on the 240 foot elevation of the HNP Turbine Building. The fire suppression system installed to protect the *condensate pumps appears to be credited in the Fire PRA to prevent a potential turbine building collapse, but is not in itself a required system. Note that one of the considerations used in determining the required systems is "By Performance Monitoring Group in Expert Panel."*

Accordingly, please either change the status of the fire suppression systems installed to protect the major oil hazards to "required" in Table 4-8, or provide a justification for classifying these suppression systems as not required.

Response: As described in RAI 3-73, and considering risk insights, the largest contributors to the five highest heat release rates noted in the HNP Turbine Building Fire PRA analysis were oil spills/fire events associated with the Digital Electro-Hydraulic (DEH) Pump Skid and Condensate Booster Pumps. In both cases, area fire suppression is provided. LAR Table 4-8-1 indicates "Required Systems" in the Turbine Generator Building to include those protecting the Condensate Booster Pumps, Steam Generator Feed Pumps, Turbine Lube Oil Reservoir and the area including the DEH Pump Skid.

HNP RAI 3-74

Use of the Performance-Based Approach with no Change Evaluation

NFPA 805 Section 2.2.8, "Performance-Based Approach," states:

> *The performance-based approach to satisfy the nuclear safety, radiation release, life safety, and property damage / business interruption performance criteria requires engineering analyses to evaluate whether the performance criteria are satisfied.*

Table 4-5, "Fire Area Compliance Summary," of the Harris Transition Report documents that Fire Areas 1-A-ACP – Auxiliary Control Panel Room, 12-A-CRC1 – Control Room Complex, and 12-A-HV&IR – Heating, Ventilating, and Instrument Repairs, Reactor Auxiliary Building, utilized the performance-based approach in accordance with NFPA 805 Section 4.2.4 but do not have associated Plant Change Evaluations to document the acceptability of this approach. The NRC staff reviewed the Fire Safety Analyses (FSAs) performed for these fire areas and noted that the Change Evaluation section for each of these associated calculations had been deleted during the latest revision.

The staff also noticed that: 1) Fire Area 1-A-ACP includes four VFDRs that require performance-based analyses to evaluate whether the performance criteria are satisfied; 2) Fire Area 12-A-CRC1 includes six VFDRs that require performance-based analyses to evaluate whether the performance criteria are satisfied; and 3) Fire Area 12-A-HV&IR includes one VFDR that requires a performance-based analysis to evaluate whether the performance criteria are satisfied.

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Accordingly, the licensee should either revise the analyses and NFPA 805 LAR for these fire areas to credit the deterministic approach in accordance with NFPA 805 Section 4.2.3 (including any required physical modifications), or provide justification that the existing engineering analyses adequately meet the NFPA 805 Section 4.2.4 performance-based approach. One method that is acceptable to the NRC staff is to perform the required engineering analysis demonstrating the ability to meet the nuclear safety performance criteria using either a qualitative or quantitative evaluation, and subsequently revise the NFPA 805 LAR to properly document the performance-based analysis.

Response: Fire Areas 1-A-ACP, 12-A-CRC1, and 12-A-HV&IR do require a change evaluation to document the acceptability of the Performance-Based approach. The calculations had been previously revised using an earlier revision of the VFDR list and were not updated prior to submittal of the LAR. Table 4-5 was inadvertently left unchanged. The calculations are currently in the process of being revised. The calculations will be revised and approved during implementation in accordance with LAR Section 4.5. Table 4-5 of the LAR has been corrected to document that change evaluations are required for these areas.

HNP RAI 4-1.1

Radioactive Release

a) <u>Reactor Containment Building Equipment Hatch</u>

The NRC staff finds that the qualitative justification provided in the August 28, 2009, HNP RAI response letter for part (e) of HNP RAI 4-1, regarding radioactive release from containment during low power and shut down conditions, as well as the information provided in Attachment E, "NEI 04-02 Table G-1 – Radioactive Release Transition," of the Harris Transition Report for FPP-012-01-CNMT, "Containment Building Fire Pre-Plan," regarding the Containment Equipment Hatch during non-power operational modes, is insufficient.

Please provide a detailed summary of the analysis / technical justification supporting an indefinite closure time for the containment hatch while fuel is off-loaded. Otherwise, provide a detailed summary of the analysis / technical justification that determines the necessary hatch closure time. The analysis / technical justification should take into account plant-specific equipment arrangement and containment geometry. Please also include a description of the key assumptions for the analysis / technical justification in the summary.

Response: Plant procedure OMM-031, "Implementation of Containment Closure," describes that containment openings are internal to the plant during non-power operations with the exception of the Containment Equipment Hatch. Closure of the Equipment Hatch for containment integrity during modes 5 & 6 is established via a containment closure plan with a specific closure time identified.

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During periods when the core is fully off-loaded to the Fuel Handling Building, the containment closure plan is in place to identify the work item, plan number, and penetration location as a tracking mechanism. While a specific closure time is not specified for the defueled condition, plant procedure CRC-851, "ODCM Software Instructions and Documentation," Section 10.12, directs that when the Containment Equipment Hatch is open, Operations should be requested to maintain ventilation such that the number of Containment exhaust fans is equal to or greater than the number of supply fans to minimize the potential for positive pressure inside the Containment Building that could lead to effluent flow from the Equipment Hatch.

Additionally, based on the volume of containment for collection of smoke, and location of the equipment hatch in relation to the top of containment (~150' below top of dome), the potential for smoke migration to lower elevations is not considered creditable prior to containment and monitoring actions being taken. Large ignition sources such as Reactor Coolant pumps and their associated oil supply were considered the largest contributor. Due to lack of large components such as these pumps/motors operating during this plant configuration, no ignition source could be identified. Additionally, with the heightened personnel attendance and monitoring of containment, the potential for fire hazards large enough to present a potential release is unlikely. Administrative controls for hot work and handling of transient combustibles during outages further enhance the prevention, detection and response elements of defense in depth for this area, ensuring the potential for radioactive release is minimized. Actions taken during the fire suppression activities are not expected to adversely affect these mitigating features and controls.

Furthermore, NEI 04-02, Revision 1 (as endorsed by NUREG 1.205), states in Section 4.3.4 that the nuclear safety goal, objectives and performance criteria all require the prevention of fuel cladding damage. As such, radiological release due to fuel damage should not require a separate examination since no such damage is assumed to occur without violating the basic requirements of NFPA 805. This effectively limits the source of radiation (release source term). Therefore, containment integrity should not require specific examination. This means the scope of the fire protection analyses need not be expanded to include all containment isolation valves. Additionally, if the reactor is de-fueled (core off loaded to the fuel handling building), the release source term has effectively been limited and containment integrity would not require specific examination.

Based on the above, radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) is expected to be as low as reasonably achievable and not exceed applicable 10 CFR Part 20 Limits. These details have been included in a revision to LAR Attachment E, Table G-1.

b) <u>Living Radioactive Release Program</u>

Please describe the configuration controls in place to ensure that the Radioactive Release Goals/Objectives/Performance Criteria will continue to be met considering the possibility that areas that were not contaminated at the time of the NFPA 805 transition review (and which were consequently screened out) may become contaminated in the future due to plant operational events. One example of this might be a steam generator tube leak that contaminates areas that are normally expected to be uncontaminated.

Response: Results of the Radioactive Release Review, Attachment E, Table G-1, will be incorporated in the NFPA 805 Code Compliance Calculation (HNP-M/BMRK-0011) during implementation, as described in LAR Section 5.4. Since this information will be contained in an Engineering Calculation, the process and results documenting conformance to NFPA 805 Radioactive Release Goals/Objectives/Performance Criteria will be maintained as part of the established Configuration Control Process. The screening criteria applied during the Radioactive Release reviews included plant areas with the "potential" for radioactive contamination (such as the Turbine Generator Building, resultant from a Steam Generator Tube Leak) and "screened in" these areas (ref. LAR Table 4-2).

HNP RAI 5-4.1

Qualification of Users

In response to HNP RAI 5-4, the licensee stated in the August 13, 2009, HNP RAI response letter that training guides had been established for personnel performing Fire Probabilistic Safety Assessment (PSA) and Fire Protection. The response further states that the qualification and training program at HNP is accredited by the Institute of Nuclear Power Operations (INPO).

Please elaborate on how the qualification and training program will ensure that personnel performing fire modeling in the future will meet the requirements of NFPA 805 Section 2.7.3.4, "Qualification of Users," with regard to being competent in that field and experienced in the application of fire modeling techniques and methods as they relate to nuclear power plants, nuclear power plant fire protection, and power plant operations.

Response: The accredited qualification and training program that is being established for the NFPA 805 transition will ensure that cognizant personnel who use and apply engineering analysis and numerical models (e.g., fire modeling techniques) are 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.

A specific Fire Modeling qualification is being developed to ensure that personnel performing fire modeling activities are competent in this area and have an understanding of the uses and limitations of the models. This qualification will be developed under the accredited ESP training process and the procedure controls under EGR-NGGC-0007, "Maintenance of Design Documents."

As discussed in LAR Section 5.4, implementation of the new program will include peer reviews, procedure changes, process updates and training to affected plant personnel to implement the NFPA 805 FP program. This will occur 180 days after NRC approval (issuance of the SER).

HNP RAI 5-14.1

Sensitivity to Cutting and Welding Fires

In the response to HNP RAI 5-14, the licensee provided a sensitivity analysis to estimate the risk contribution of transient fires due to cutting and welding by applying the ignition frequencies from NUREG/CR-6850, "EPRI/NRC-RES [Research] Fire PRA Methodology for Nuclear Power Facilities," for the frequency bins related to transient cutting and welding fires (bins 6, 24, and 36) to the general transient source scenarios (bins 3, 7, 25, and 37), and assuming a continuous fire watch. The results of this sensitivity analysis were presented in a table.

However, it is unclear what non-suppression probabilities were assumed in the sensitivity analysis when the frequencies for the bins related to transient cutting and welding fires (bins 6, 24, and 36) were substituted for those corresponding to the general transient source scenarios (bins 3, 7, 25 and 37). Since the sum of the ignition frequencies for bins 6, 24, and 36 (0.0228 per reactor-year) is comparable to that for bins 3, 7, 25, and 37 (0.0243 per reactor-year), any differences should result primarily from the choice of non-suppression probabilities. If the transient fire non-suppression probabilities from NUREG/CR-6850 were assumed, then these should bound those for cutting and welding in order for the sensitivity analysis to confirm the licensee conclusion of no significant plant impact due to a transient cutting and welding fire.

Accordingly, please clarify what non-suppression probabilities were employed for the sensitivity analysis and discuss if these bound the assumed non-suppression probabilities for cutting and welding so as to support the licensee's conclusion in the current response to HNP RAI 5-14.

Response: The sensitivity was performed to address the concern that transient fires due to cutting and welding (bins 6, 24, and 36) were not specifically addressed. For the sensitivity analysis, the previously identified transient fire scenarios were reanalyzed using the bin 6, 24 and 35 frequencies and the non-suppression probabilities for cutting and welding. These cutting and welding non-suppression probabilities are the most appropriate due to the fact that these fires are initiated by cutting and welding activities which always have a continuous fire watch present by procedure. Consistent with the treatment of other cutting and welding initiated fires evaluated in the HNP fire PRA, and the guidance in NUREG/CR-6850, a lambda of 0.19 was applied using the time to damage for the previously identified transient scenarios. The time to damage varies based on the target distance from the source.

The sensitivity analysis is not intended to bound the results, but to provide the best estimate of the risk due to these fires that were not quantified in the base calculation. The results demonstrate that the risk contribution from these bins is very small compared to the general transient fires and will not alter the previously identified risk insights.

HNP RAI 5-18.1

Screening Thresholds for Hot Gas Layers (HGLs)

One part of HNP RAI 5-18 addressed a statement from Section 5.8.3, "Detailed Qualitative Assessment (Step 3)," of calculation HNP-F/PSA-0079, Revision 1. The RAI noted that the screening values cited in the statement are based on HGL ignition frequencies in compartments for which the conditional core damage probability (CCDP) could be close to 1.0, or have the potential to increase to values close to 1.0 due to failures in the propagated compartment. Accordingly, the request was made to provide the basis for the "no further evaluation is performed" screening value of less than 1.0E-7 per year. In addition, the request asked the licensee to discuss the correspondence, if any, between the less than 1.0E-7 per year screening criterion and that for the less than 1.0E-6 per year screening criterion when considering the HGL. However, in the response the licensee appears to have addressed the use of "1.0E-7 per year rather than 1.0E-8 per year" as the screening criterion, instead of "1.0E-6 per year rather than 1.0E-7 per year," as requested.

Accordingly, please provide the basis for the "no further evaluation is performed" screening value of less than 1.0E-7 per year. In addition, discuss the correspondence, if any, between the less than 1.0E-7 per year screening criterion and that for the less than 1.0E-6 per year screening criterion.

Response: While NUREG/CR-6850 describes a process for screening, it does not provide a screening threshold. Therefore the initial screening threshold of 1.0E-7 for "no further analysis" is based on the use of screening probabilities. If the compartment does not pass the initial screening, a detailed review of the barriers and sources is performed. If no issues are identified, the barrier failure probabilities are qualitatively judged to be improved by an order of magnitude, effectively raising the screening threshold to 1.0E-06.

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HNP RAI 5-22.1

Spurious Actuation Probabilities

In the response to HNP RAI 5-22, the licensee discussed the use of a spurious actuation probability of 0.30 instead of 0.60 as the best estimate for failure mode probability in the ignition frequency calculations throughout Section 5.3, "Main Control Board Fire Initiating Events – Successful Fire Suppression," and Section 5.4, "Main Control Board Fire Initiating Events – Fire Suppression Failure Scenarios," of Attachment 4, "Main Control Room Analysis," to calculation HNP-F/PSA-0079, Revision 1, for the HNP Main Control Board (MCB) analysis.

Although justifying the validity of this spurious actuation probability for motor operated valves (MOVs), the licensee stated that the use of 0.30 is non-conservative for air operated valves (AOVs) with direct current (DC) solenoid controllers, and that this will be corrected when calculation HNP-F/PSA-0079 is updated. The licensee further responded that the conclusions of the NFPA 805 LAR are not expected to be adversely impacted because there are other offsetting conservatisms, a list of which was provided.

However, with respect to the use of the 0.30 spurious operation probability for AOVs with DC solenoid controllers instead of the appropriate value of 0.60, the NRC staff questions the claim that the conclusions of the NFPA 805 LAR would not be expected to be adversely impacted because of offsetting conservatisms. For example, relaxation of one of the cited conservatisms – no consideration of hot short duration – may not provide any reduction in overall hot short probability for DC-powered circuits. Additionally, relaxation of the cited conservatism of not crediting installation of incipient detection may not provide a sufficient offset because the effectiveness of incipient detection is highly dependent on the specific conditions of the installation, maintenance of the system, nature of the ignition sources, etc.

Accordingly, please provide a basis for the conclusion of no adverse impact from the non-conservative use of the 0.30 hot short probability for AOVs rather than 0.60, including a sensitivity analysis that is not dependent on "offsetting conservatisms."

Response: A 0.3 hot short probability with no recovery was applied in five sections of the MCB. Three (3) of those sections impact AOVs and account for 2.1% of the overall MCR risk. The dominant MCR risk contributor in the original analysis comes from a single MCB section that did not apply hot short probabilities (62.6%). Additionally, the original analysis of this MCB section did not credit the installation of VEWFDS, which provides a substantial risk reduction.

A sensitivity analysis has been performed by applying a 0.6 hot short probability (without recovery) in place of the original 0.3 for the AOVs and applying credit for the VEWFDS where applicable. This sensitivity shows an overall CDF decrease for the MCR from 4.63E-6/yr to 2.18E-6/yr. If the VEWFDS is not credited, the MCR risk increases by less than 10% and is still dominated by the previously mentioned MCB panel section (> 57%). Based on this data, no additional insights have been revealed.

HNP RAI 5-30.1

Uncertainty Analysis

In the response to HNP RAI 5-30, the licensee states that, with respect to the disposition for finding and observation (F&O) FSS-E3-1 in Attachment X, "Fire PRA Quality," of the Harris Transition Report, they "did not perform an overall statistical uncertainty analysis such as that typically performed for internal events because of software limitations with the methodology used to quantify the Fire PRA. The impact of this is not expected to influence the decisions related to this application."

While "software limitations" (i.e., the use of FRANC) may preclude an exact reproduction in the Fire PRA of the type of parametric uncertainty analysis performed for the internal events model of record, it should still be feasible to examine the effect of uncertainty on at least those human failure probabilities more likely to dominate the fire risk, even if this is accomplished through a means independent from that employed by using the internal events PRA software.

Accordingly, please provide a firmer basis for the conclusion that the uncertainties would not be expected to influence the decisions related to the NFPA 805 LAR.

Response: The finding for the referenced F&O states that, "No uncertainty analysis was performed." At the time of the NRC Staff Review, this was the case. The sources of uncertainty were subsequently addressed in Section 6.7 of the quantification calculation (HNP-F/PSA-0079) and additional uncertainty analysis was provided in the application calculation (HNP-F/PSA-0081). Several sensitivities have also been performed. These analyses clearly indicate that the majority of the uncertainty is in the conservative direction. An industry peer review was also conducted on the revised Harris fire PRA. The only outstanding item is the lack of an "overall uncertainty interval associated with the parameter uncertainties." However, the individual parameter uncertainties have been found acceptable.

The individual parameters in the fire PRA are identical to those in the internal events with the exception of certain HEPs and certain fire specific basic events (see RAI 5-36.1). The methodology for determining the uncertainty parameters for fire HEPs is the same as that used for the internal events HEPs. Therefore, the relative ranges of these uncertainties are the same as, or essentially equivalent to, those employed in the internal events PRA with respect to the best-estimate values.

Due to the format of results from the FRANC software, Progress Energy has not been able to complete propagation of the cutset results through the statistical analysis tool. However, because the formal model is essentially equivalent to the internal events model, the fire specific basic events are based on the data recommended by the preferred guidance and the balance of the uncertainty analysis and sensitivities indicate that most of the uncertainty is in the conservative direction, the lack of the overall uncertainty interval is not expected to influence the decisions related to this application.

The primary risk drivers in the decision process are total CDF/LERF and the delta CDF/LERF associated with the VFDRs. The quantitative analysis shows these metrics to be within acceptable thresholds for the 805 application. If removal of the conservatisms identified in the uncertainty analysis could be justified, the total risk would be reduced and the delta risks would be expected to be similar or lower. This is due to the reduced risk of individual scenarios and the tendency for uncertainties to cancel out during the delta risk calculation. Therefore, the decisions regarding the acceptability of the 805 transition would be expected to remain acceptable.

HNP RAI 5-31.1

Sensitivities on Reactor Coolant Pump (RCP) Seal Loss of Coolant Accidents (LOCAs) and Internal Events PRA Issues

HNP RAI 5-31 consisted of two items related to Table 15.1, "Amended Summary of Assumptions and Sources of Uncertainty in the HNP Fire PRA," of Attachment 15, "Identification of Sensitivity Analyses for HNP NFPA 805 Change Evaluations," to calculation HNP-F/PSA-0081, "Harris Fire PRA – Support for NFPA 805 Transition," Revision 1, which includes Assumptions 30, 31, and 32 regarding the fire risk model and cable selection. The licensee provided a response to both of these items.

While satisfied with the response to the part of item (1) addressing Assumption 32 because there are no VFDRs associated with the cables in question, the NRC staff still has the following concerns: 1) With regard to the part of item (1) addressing Assumption 30, the licensee cites use of the Westinghouse RCP Seal LOCA Model, but does not address its role in the expectation of having no masking effects. 2) With regard to item (2), the licensee appears to conclude that sensitivities performed for the internal events model adequately address sensitivities that could be considered for the Fire PRA, but does not elaborate.

Please provide additional discussion regarding the role of the Westinghouse RCP Seal LOCA Model in item (1) and the conclusion related to item (2), as identified above.

Response: Because the RCP seal LOCA model and success criteria are well established and based on models provided by the PWR Owners Group, it is relatively straight forward to identify and evaluate these in the PRA results. The potential conservatisms related to this RAI are considered acceptable since it would be difficult to justify less conservative assumptions.

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Based on these factors, no additional insights are envisioned that would impact the conclusions derived from the results.

Cable damage which could be postulated for induced tube ruptures has been considered with the modeling of induced tube ruptures as a potential consequence of certain transient sequences. All of the transient sequences were assessed and component/cable failures which might contribute to inducing a tube rupture are included, including any risk impact.

HNP RAI 5-33.1

Incipient Detection System

In the response to HNP RAI 5-33, the licensee provided an additional sensitivity analysis for Section 3.3.5.2, "Sensitivity Issue 7: Incipient Fire Detection in Low Voltage Cabinets," of Attachment 16, "Impact Assessment of Key Sensitivity Issues," to calculation HNP-F/PSA-0081, Revision 1. Three cases were considered: (1) Incipient Condition Exists (SI) = 0.001, Very Early Warning Fire Detection Systems (VEWFDS) Reliability (ID) = 0.005, Plant Response to Event (IP) = 0.001; (2) SI = 0.01, ID = 0.005, IP = 0.05; and (3) SI = 0.01, ID = 0.01, IP = 1.0; with the last case not being considered credible because of the use of the 1.0 value for IP.

While the NRC staff recognizes the licensee's reluctance to consider the most pessimistic (third) sensitivity case as credible, the staff also notes that in the previous case (second) the applicant was willing to consider sensitivities in the parameters SI and IP that increased their values from the base (first) case by factors of 10 and 50, respectively, but not to increase the value for parameter ID, which relates to the reliability of the VEWFDS.

For consistency with the other parameters, please consider expansion of at least the second case to increase the value of parameter ID to at least 0.05 (factor of 10 relative to base case). Also, discuss how the post-transition monitoring program will assure VEWFDS unreliability (parameter ID) will be kept below the level needed to assure the effectiveness being credited.

Response: Additional sensitivity results for the specific case requested and a case which increases each of the parameters by a factor of 10 have been included. Reference the original RAI response (SERIAL: HNP-09-094, dated October 09, 2009) for descriptions of all the terms.

SI	ID	IP	CDF (yr)	VFD delta CDF (yr)
0.001	0.005	0.001	3.06E-05	1.13E-06
0.01	0.005	0.05	3.28E-05	1.25E-06
0.01	0.05	0.01	3.92E-05	1.45E-06
0.01	0.05	0.05	3.99E-05	1.50E-06
0.01	0.01	1.0	4.89E-05	2.59E-06

In the fourth case above, virtually all of the change in delta risk for the VFDs is due to impacts in CSRB (increased 1.19E-07 to 1.30E-07) and SWGRB (increased 2.54E-07 to 1.30E-06).

The post-transition monitoring program will ensure that the VEWFDS unreliability will be kept below a level needed to assure the effectiveness being credited by maintaining the system according to both NFPA 72 and the manufactures' maintenance & testing schedule. Additionally, the system is self monitoring and will alert the plant to system faults or precursors to a fault. Based on the anticipated level of monitoring and maintenance, the manufacturer has provided reliability estimates that are better than those currently used in the PRA analysis.

HNP RAI 5-36.1

Uncertainties and Software Limitations

In the response to HNP RAI 5-36, the licensee states that, with respect to the disposition for F&O FSS-E3-1 in Attachment X of the Harris Transition Report, "the F&O relates to statistical uncertainties. The modeled basic events include a parameter for incorporating the error factors. Harris did not perform an overall statistical uncertainty analysis such as that typically performed for internal events because of software limitations with the methodology used to quantify the fire PRA. The impact of this is not expected to influence the decisions related to this application."

While "software limitations" (i.e., the use of FRANC) may preclude an exact reproduction in the Fire PRA of the type of uncertainty analysis performed for the internal events model, it should still be feasible to examine the effect of uncertainty using a surrogate method – perhaps a modified sensitivity analysis – in order to address potential uncertainties and support the conclusion that "the impact is not expected to influence the decisions related to this application."

Accordingly, please provide a firmer basis for the conclusion that the uncertainties would not be expected to influence the decisions related to the NFPA 805 LAR.

Response: The finding for the referenced F&O states that, "No uncertainty analysis was performed." At the time of the NRC Staff Review, this was the case. The sources of uncertainty were subsequently addressed in Section 6.7 of the quantification calculation (HNP-F/PSA-0079), and additional uncertainty analysis was provided in the application calculation (HNP-F/PSA-0081). Several sensitivities have also been performed. These analyses clearly indicate that the majority of the uncertainty is in the conservative direction. An industry peer review was also conducted on the revised Harris fire PRA. The only outstanding item is the lack of an "overall uncertainty interval associated with the parameter uncertainties". The individual parameter uncertainties have been found acceptable.

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The individual parameters in the fire PRA are identical to those in the internal events with the exception of certain HEPs (see RAI 5-30.1) and those listed below. The parameter values and uncertainty intervals for the following are all taken from NUREG/CR-6850:

Ignition Frequencies: Appendix C
Non-Suppression Probabilities: Appendix P
Hot Short Probabilities: Page 10-7

Due to the format of results from the FRANC software, Progress Energy has not been able to complete propagation of the cutset results through the statistical analysis tool. However, the formal model is essentially equivalent to the internal events model, the fire specific basic events are based on the data recommended by the preferred guidance and the balance of the uncertainty analysis and sensitivities indicate that most of the uncertainty is in the conservative direction. Therefore, the lack of the overall uncertainty interval is not expected to influence the decisions related to this application.

The primary risk drivers in the decision process are total CDF/LERF and the delta CDF/LERF associated with the VFDRs. The quantitative analysis shows these metrics to be within acceptable thresholds for the 805 application. If removal of the conservatisms identified in the uncertainty analysis could be justified, the total risk would be reduced and the delta risks would be expected to be similar or lower. This is due to the reduced risk of individual scenarios and the tendency for uncertainties to cancel out during the delta risk calculation. Based on this, the decisions regarding the acceptability of the 805 transition would be expected to remain acceptable.

HNP RAI 5-37.1

Going Forward Fire PRA Improvements

In the response to HNP RAI 5-37, the licensee states that, with respect to the disposition for F&Os FQ-F1-1, FQ-A4-02, FQ-D1-01, and FQ-F1-01 in Attachment X of the Harris Transition Report, "HNP will continue to be involved with the issues surrounding fire PRA development and the associated uncertainties. Improvements in the methods and data developed through industry and regulatory efforts will be incorporated in future revisions to the HNP Fire PRA. These improvements will be reflected in the 'going-forward' applications."

The NRC staff recognizes that these "improvements" would suffice to address the concern in HNP RAI 5-37 post-transition if such improvements are indeed available when the "going forward" Fire PRA is in effect for post-transition change evaluations.

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However, since this expectation may not necessarily be fulfilled, please discuss current plans to address the possibility of such improvements not being available at the anticipated time, as well as the impact this would have, if any, on post-transition change evaluations.

Response: Post transition change evaluations will use the same methods as the fire risk evaluations used for the LAR and transition. If a change evaluation requires the use of a new method that was not reviewed as a part of the baseline fire PRA or transition application, additional review and approval may be required. Change evaluations and other risk applications will be performed using the latest approved PRA. Open F&Os will be addressed in an on-going basis as needed to apply the risk insights. These situations will need to be addressed on a case by case basis.

HNP RAI 5-41

Impact of Plant Changes Not Yet in the PRA Model on Change Evaluations

In the NFPA 805 LAR descriptions of administrative controls for PRA model updates, there was no discussion of how plant changes not yet incorporated into the PRA model will be addressed in Plant Change Evaluations.

Accordingly, please describe the process for evaluation of such plant changes, including any screening criteria, expert panel consideration, or other disposition methods, when the changes are not yet included in the PRA model used to support a change evaluation.

Response: Plant Changes will be processed per current plant procedures which include EGR-NGGC-0003, "Design Review Requirements," EGR-NGGC-0005, "Engineering Change," and PRO-NGGC-0204, "Procedure Review and Approval." Plant changes, such as Engineering Change or Procedure change, would first go through a regulatory review to determine if the change involved a Chapter 3 requirement and to determine if a License Amendment Request is required or not. A preliminary risk review is then performed to determine if the change is trivial or not and if the change potentially impacts risk greater than minimally. If so, the change is then subjected to an initial evaluation screen for risk evaluation. Qualitative and Quantitative evaluations are performed and documented as required per each change. To determine whether the change is acceptable or not the change in CDF and LERF are calculated, when necessary, and DID and safety margins are verified as acceptable. The conclusion is then documented and input into the appropriate Progress Energy process for the initiating change.

When a Fire Protection Change Evaluation is being evaluated, plant changes that are not addressed in the latest plant specific PSA model need to be considered for impact on the proposed change to ensure that these un-incorporated changes do not adversely change conclusions relative to self approval. In addition, as appropriate, the change is identified as needing input into the next PRA model update. New procedure FIR-NGGC-0010, "FPP Change Process," outlined in LAR Attachment Z, discusses the change impact review process.

HNP RAI 5-42

Changes Made to the HNP Fire PRA Since the NRC Staff Audit

The February 2008 audit conducted by the NRC staff identified elements of the PRA standard which were satisfactorily met by the existing HNP Fire PRA. However, the staff identified that due to the incomplete status of major elements of the Fire PRA, the conclusion could not be reached that once the Fire PRA was complete these items would not be adversely impacted.

Therefore, the staff suggested that a full peer review of the HNP Fire PRA model be conducted when the Fire PRA was complete. The licensee declined to conduct such a review, and has used the NRC staff's disposition of acceptability for these elements in February 2008 to justify the technical adequacy of the HNP Fire PRA model. A focused scope Fire PRA review was conducted only for those items found deficient during the February 2008 audit.

In order to assist the NRC staff in reaching the conclusion that the HNP Fire PRA model continues to be acceptable, please identify the changes made to the Fire PRA model since the February 2008 audit and provide the basis for concluding that these changes do not impact those standard elements found acceptable during the NRC review.

Response: At the start of the pilot effort, there was no published standard for Fire PRA. It was determined that the best approach would be for the NRC to conduct their own Fire PRA reviews of the 805 pilot plants based on the latest available draft standard using the industry peer process. This review was held in February 2008. At that time most of the Harris Fire PRA products had been performed and documented, but the formal calculations had not been finalized. The NRC performed the reviews and provided grades and findings in the same format as a typical industry peer review. All elements were reviewed except for FSS-F, FSS-G, SF, and UNC.

Following the NRC staff review, the calculations were updated to address NRC findings in preparation for the follow-up industry peer review, held in April 2008. As with the NRC review, several of the products were updated and complete, however they were not formally issued as calculations. It should be noted that the process of performing a fire PRA is very iterative. The initial analyses are more heavily based on conservatively bounding assumptions and the initial results may not support the intended application without additional analysis. The risk insights derived from the initial results are used to focus additional, more refined analysis, to remove uncertainty of the potential risk on the most significant scenarios. This process can be repeated multiple times before the conclusions are finalized. These iterations do not imply that new methods are being applied that would require additional peer review. Specific examples of these updates include completion of additional walk downs, expanded application of fire modeling insights, additional application of hot short probabilities and modeling additional instrumentation to support human events.

Based on the results of the NRC review and changes to the fire PRA, the scope of the industry peer review included those items not previously reviewed by the NRC and items that were determined to be "not met" or graded as Category I. The quantification results and documentation were also included, which references most of the other products as inputs. In addition, the review specifically addressed SRs related to findings generated during the NRC review. The findings and results for both the NRC and the industry reviews are discussed in Section X of the LAR.

The fire PRA calculations were updated to address the industry peer review comments and were formally issued prior to being used to support the license application.

The tables below include the revision logs from each of the Harris Fire PRA calculations. As discussed above, in most cases the Revision 0 calculations included changes that address the specific peer review findings. Changes for subsequent revisions are specifically listed. These changes consist primarily of documentation enhancements, data updates and expanded use of more detailed assessments of individual scenarios for the risk significant sources. Review of the changes does not indicate that there are any methodology changes that would necessitate the need for another peer review.

HNP-F/PSA-0071, Fire Ignition Frequency Calculation

Rev. #	Date	Revision Summary (list of ECs incorporated)
0	10/06/2006	Initial issue of this calculation. It defines compartments, identifies ignitions sources, and determines compartment frequencies from the Harris Fire PRA.
1	04/04/2008	Revised Global Boundaries, re-named Compartments, added new Ignition Sources, resolved Revision 0 NRC review comments, revised Ignition Source counts for electrical cabinets based on FAQ-16, 17, 18.
2	09/08/2008	This revision incorporated resolutions to the Facts and Observations identified during the NRC's HNP FPRA peer review including IGN-A5-1, IGN-A9-1, IGN-B5-1, PP-B2-1, PP-B2-2, and PP-C3-1 (HNP-F/PSA-0083 "HNP Fire PRA Reviews"). Removed oil ignition sources. Incorporated NCR 283136. Revised ignition frequency values. Removed sources FC41_S1607 and FC41_S1608 because they were duplicates of sources FC41_S0866 and FC41_S0824. Added a compartment mapping table (Attachment 18). Added Fire Zone Data (Attachment 19). Separated the fixed and transient ignition frequencies into two separate tables (addition of Attachment 20). Added Attachment 21 to describe the contents of Fire Ignition Sources.mdb. Revised Section 4.1 to clarify the calculation outputs.

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Revision summary of calculation HNP-F/PSA-0071 discusses boundary changes. The calculation as presented to the NRC excluded areas from analysis based on initial screening as described in Attachment 2 of the calculation. The change to the calculation in regards to the global boundaries is documentation only. Attachment 2 is unchanged in the current revision.

HNP-F/PSA-0077, Component Selection and Fire-Induced Model Calculation

Rev. #		Revision Summary (list of ECs incorporated)
0	05/21/2008	Original Calculation. No EC's are incorporated.

HNP-F/PSA-0075, Harris Fire PRA – Human Reliability Analysis

Rev. #		Revision Summary (list of ECs incorporated)
0	04/03/2008	Initial issue of this calculation. Documents the Human Reliability Analysis performed for the Harris Fire PRA.
1	11/05/2008	Incorporated changes in accordance with the following F&Os: HRA-B2-1, HRA-B3-1, HRA-C1-1, HRA-C1-4, HRA-C1-6, HRA-C1-7, and ES-C1-1. No changes were made to any of the attachments of this calculation and no changes were made affecting the technical content of this calculation.

HNP-F/PSA-0078, Harris Fire PRA – Scoping Walkdown Calculation (and Fire Scenario Data)

Rev. #		Revision Summary (list of ECs incorporated)
0	06/10/2008	• Initial issue of this calculation.
1	11/06/2008	• General rewrite of calculation. • Established a baseline database for fixed, transient, and oil ignition sources and source-to-target information to be used in further quantification processes. • Relocated ignition source and target set databases from HNP-F/PSA-0079 revision 0 to this calculation's database in order to obtain control of the data specifically recorded in the walkdown and fire modeling process. • Updated walkdown sheets and database containing walkdown data with oil and transient ignition sources added to resolve HNP-F/PSA-0079 Attachment 24 open items. Included tables for disposition of potential oil sources and verification of oil quantities. • Revised/added sheets to Attachment 3 – Transient Source Walkdown Sheets, in response to attachment 24 of HNP-F/PSA-0079 rev. 0. Also, added transient source information for TB-240. • Established data to be used for building protected cable lists for further quantification processes. • Translated heat release limit calculations to determine hot gas layer scenarios for each compartment based on calculation HNP-M/MECH-1128.

		• Eliminated former attachment 5 and replaced it with reference to database. Renumbered other attachments as appropriate.
2	03/31/2009	• Moved Section 3.4 to attachment 26. • Revised section 3.4 • Updated Database, Fire Scenario Data.mdb, table "z-0078R1 SOURCE". The "Target_Set" field was updated for sources including: Cable fires cause by cutting and welding, transient fires caused by cutting and welding, "Exception Screen" sources, Sources with no targets, and FC03_S0096, FC17_S0401, and FC17_S0407. "Temp_Nearest_Target", "Temp_Nearest_Tray", and "z-0078R1 Source-Target Short" were updated to reflect new target sets. • Updated Section 3.4.1, Ignition Source Data, with definitions of Target_Set field including new Target Sets: "No Targets" (NT) and "Exception Screen" (ES) • Updated Cutting and Welding methodology description, section 3.5.6, for treatment of Transient fires caused by cutting and welding, and process to capture targets for cable fires caused by cutting and welding. • Updated attachment 3, Transient Ignition Source Walkdowns. Pages 214, 249, 250, and 254 were deleted, and the table of contents was updated. Deleted pages, 249, 250, and 254, were oil ignition source walkdowns, and the same walkdown sheets are located in Attachment 4. Page 214 is no longer applicable to the document. • Removed conduit 12763J from Table [Cable_Mod_B] since EC 68660 will not protect this conduit • Moved historical database change information from revision 1 to attachment 26 • Added attachment 27 to document revision 2 data changes.

HNP-F/PSA-0079, Harris Fire PRA – Quantification Calculation

Rev. #		Revision Summary (list of ECs incorporated)
0	06/12/2008	Initial issue.
1	04/06/2009	Incorporate updated inputs. Incorporate updated/revised quantification process.

The process change referred to in the Revision 1 change of calculation HNP-F/PSA-0079 was to allow integration of each individual fire compartment spreadsheets into one spreadsheet. This process change allowed the use of a single integrated spreadsheet to calculate and summarize results. The methods for determining the fire CDF and LERF were not changed.

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HNP RAI 5-43

Inter-Cable and Three-Phase Hot Short Assumptions

a) *Inter-cable and three-phase hot shorts were not considered for having the potential to lead to containment bypass and the possibility of a large early release from containment. The licensee indicates that the issue has been resolved by evaluation of each potential containment release pathway for multiple spurious operations, and consideration by an expert panel of the potential consequences of more than two spurious operations. Although the corresponding supporting requirements (SRs) (Cable Selection (CS)-A7 and CS-A8) were reviewed by the Focused Scope Peer Review Team and found to be acceptable, it is not clear that this disposition addresses inter-cable as opposed to intra-cable hot shorts.*

   Accordingly, please provide a clarifying description of the above process, including a discussion of whether or not the two types of hot shorts were considered and, if not, why this was deemed acceptable during the Focused Scope Peer Review.

b) *The Focused Scope Peer Review finding associated with SR Circuit Failure Analysis (CF)-A1 noted that contributions to hot short probability do not necessarily include both intra- and inter-cable hot shorts. The licensee indicates that no change to the Fire PRA method was made based on the inclusion of conservatisms in the hot short probabilities.*

   Please provide a discussion of these conservatisms in the hot short probabilities and the basis for not changing the Fire PRA method, including appropriate justification.

Responses:

a) Within the safe shutdown analysis, components whose spurious operation could lead to an intersystem LOCA were classified as high-low pressure interfaces and cable to cable shorts were considered. Cable to cable shorts were also considered in the Fire PRA.

   The Fire PRA identified eight (8) containment bypasses (1CP-1, 1CP-3, 1CP-4, 1CP-5, 1CP-6, 1CP-7, 1CP-9, and 1CP-10), six of which have had their circuits analyzed. 1CP-1 and 1CP-7 were not analyzed since they are locked by locking 1IA-1054-I1 and 1IA-1044-I1 respectively and disconnecting the instrument air line between the isolation valve and the actuator during normal operation. During normal operations, 1CP-4 and 1CP-10 have power removed from the valve operators via key lock switch CS-2691.1SA in ARP-4A-SA.

   These components are all air-operated with solenoid pilot valves. On a loss of power to the solenoid or a loss of air, the valves are designed to fail closed, which is the required position for the Fire PRA.

A review of the circuit indicates that a three-phase hot short is not applicable as they are 120V AC solenoids. A simple intra-cable hot short can spuriously open the valves and was considered in the PRA.

b) The hot short probabilities were assigned to a sub-set of the most critical PRA components. For the majority of PRA components, the hot short probability was assumed to be one (1) and no additional circuit analysis was performed. For those that did have fault probabilities assigned, the worst-case (inter vs. intra) probability number was assigned. For example, in the case of the valves above, the likelihood of an inter-cable hot short causing the spurious opening of any of the valves would be 0.06 (thermoset cable, no CPT), whereas the likelihood of an intra-cable hot short causing the spurious opening would be 0.6. If the applicable circuits had been analyzed for their fault probability, a value of 0.6 would have been assigned since this is the more conservative of the two numbers. The valve would not be susceptible to an intra-cable and an inter-cable short at the same time. Since HNP has very few single conductor cables, all cables were assumed to be multi-conductor and all were assumed to be routed in tray unless they were known to be in dedicated conduit throughout their run or for the specific scenario in question.

For those cases where both intra- and inter-cable hot shorts were possible, 0.66 (0.33 in circuits with a CPT) would be the preferred value from NUREG/CR-6850. It was determined for these cases that the applied value of 0.60 was acceptable based on the following:

- Per Table 10-2 in NUREG/CR-6850, the uncertainty for an intra-cable short in a multiconductor cable is from 0.2 to 1.0, which envelopes the omission of the potential 0.06 increase to the assigned value of 0.6.
- The hot short probabilities in NUREG/CR-6850 are not adjusted for thermoset vs. thermoplastic cable. The thermoset cables used at HNP would be expected to be at the lower end of the distribution.
- Unless cables were known to be in dedicated conduit, they were assumed to be in trays. It is likely that many were in conduit, which would allow the use of 0.15 instead of 0.6. This conservatism also offsets the omission of the potential 0.06 increase.

Since the fault probabilities for multi-conductor cables routed in conduit are lower by an approximate factor of four, the assumption that the subject cables were always in tray is conservative. In the example for the valves above, if the subject cables were routed in conduit for the scenario of concern, the probability of an intra-cable hot short resulting in the spurious actuation would be 0.15 instead of 0.6.

Note that for inter-cable shorts between multi-conductor cables, NUREG/CR-6850 provides a range of values. In this case, the best estimate range provided is 0.02 to 0.1 (thermoset cable, no CPT, multiconductor to multiconductor), and HNP chose the center of this range, 0.06. It is also noted that while the fire testing conducted by the industry demonstrated that cables with thermoplastic insulation were more susceptible to inter-cable faults, the probabilities assigned in NUREG/CR-6850 for both intra-cable and inter-cable shorts are the same for thermoset cables as for thermoplastic.

If the circuits were not identified for additional analysis to assign a fault probability as outlined in NUREG-6850, then the assumed fault probability in the Fire PRA would remain at 1.0. With the conservative fault probability of 1.0 used for most spurious actuations, it was determined acceptable not to add the fault probabilities of the two failure modes per NUREG-6850.

For circuits that are not specifically discussed in NUREG-6850, such as low-voltage shielded DC control circuits, a fault probability for hot shorts of 1.0 was assigned. This is also considered conservative since the source would have to be a similar low voltage cable of the proper polarity and associated with the same DC source.

HNP RAI 5-44

Justification of SRs that Are Either Met or Capability Category (CC)-I

NRC Staff Review Findings associated with the following SRs, as dispositioned by the Focused Scope Peer Review Team to be satisfied at either the Met or CC-I levels, were deemed acceptable by the licensee:

(a) *Fire Scenario Selection and Analysis (FSS)-D7: based on a review of the licensee's calculations and the treatment of fire detection system unavailability. It is not clear why an outlier performance by a detection system might not result in a significant risk increase for HNP, which is the basis for CC-II.*

(b) *FSS-D9: based on no treatment of the potential for smoke damage to equipment credited in the Fire PRA. It is not clear why there is any plant-specific basis to disregard smoke damage when the PRA consensus standard includes this supporting requirement. Consequently, the licensee's assertion that the risk associated with smoke damage to equipment credited in the Fire PRA is not significant has not been justified.*

(c) *Fire Ignition Frequency (IGN)-A4: based on a lack of consideration of plant-specific experience for fire events required for CC-II. It is not clear that failure to consider plant-specific fire experience is conservative, and so the justification is not adequate.*

(d) *Fire Risk Quantification (FQ)-F2 (note that no Finding was identified): based on completing the referenced requirements of the internal events PRA standard to document any non-applicability and identify the non-applicable requirements. No basis for not meeting the internal events PRA standard requirements is provided.*

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Given the above discussion, please provide the bases for concluding that the SRs associated with these Findings are acceptable at the Met or CC-I levels.

Responses:

(a) FSS-D7 states that, "In crediting fire detection and suppression systems, USE generic estimates of total system unavailability provided that (a) the credited system is installed and maintained in accordance with applicable codes and standards (b) the credited system is in fully operable state during plant operation and (c) the system has not experienced outlier behavior relative to system unavailability."

The Harris fire PRA currently applies the generic unavailability values provided in NUREG/CR-6850. Actual system performance is being reviewed in preparation for implementation of the monitoring program and the generic values appear to be slightly conservative.

Fire suppression and detection systems provided at HNP have been installed in accordance with applicable NFPA Codes and Standards as documented in detail in LAR Attachment A (NEI 04-02, Table B-1). Procedure FPP-013, "Fire Protection- Minimum Requirements, Mitigating Actions and Surveillance Requirements," describes and controls fire protection and suppression systems, including compensatory actions when applicable, ensuring credited systems are operable during plant operations. Additionally, system availability is monitored as part of the NFPA 805 Monitoring Program as described in LAR Section 4.6.

(b) FSS-D9 states, "EVALUATE the potential for smoke damage to FPRA equipment on a qualitative basis and INCORPORATE the results of this assessment into the definition of fire scenario target sets." The footnotes continue with, "Fire scenarios that assume widespread damage (e.g., damage across an entire physical analysis unit) would generally capture potential smoke damage within the limits of the assumed fire damage (e.g., assuming the loss of all equipment in a physical analysis unit given a fire, as might be employed during early stages of a screening analysis)."

HNP utilizes smoke purge ventilation systems for major plant areas such as Switchgear Rooms, Electrical Protection Rooms, Rod Control Cabinets Rooms and Stairs B, Control Room and Cable Vault areas. Smoke purge capabilities are further detailed in the Fire Safety Analyses by individual plant fire area. For all other plant fire areas, smoke management, in addition to smoke purge ventilation, is provided through the use of portable smoke ejectors by the fire brigade, as described in Fire Pre-Plans for individual plant fire areas/zones. By minimizing the time of exposure and extent of smoke migration, damage to FPRA related equipment is likewise reasonably expected to be minimized.

(c) The Harris Ignition Frequency calculation reviewed and documented plant experience, concluding that there were no outliers. The majority of the fires were associated with hot work and none were considered to be in the potentially challenging category as described by NUREG/CR-6850 Appendix C. There were three noteworthy fires for bin 15, none of which resulted in external damage. Two were extinguished with CO2 or dry chemical and the other did not require any suppression. Since a Bayesian update of the generic data using zero plant events would reduce the generic frequencies results, the currently applied values were considered conservative. Based on the current analysis, this SR (IGN-A4) meets Category II.

(d) Attachment W of the LAR provides the assessment of the Internal Events PRA model quality. Since the models were upgraded to meet quality requirements or to incorporate findings from the previous peer review or the gap assessment, a limited scope peer review was performed to address those elements. If an SR did not meet Category II, F&Os were developed. Attachment W provides the resultant F&Os with the dispositions or justifications to support the 805 application. Based on these results, the internal events PRA model was determined to be adequate for use in this application.

HNP RAI 5-45

Exposed Structural Steel

For CC-I/II, SR FSS-F1 states: "DETERMINE if any locations within the Fire PRA global analysis boundary meet both of the following: (a) exposed structural steel is present; (b) a high-hazard fire source is present in that location AND, if such locations are identified, SELECT one or more fire scenario(s) that could damage, including collapse, the exposed structural steel for each identified location."

The licensee indicates that it identified and assessed qualitatively the most limiting location and fire scenario, which was the turbine building and a large turbine lube oil fire, thereby implying that if the most limiting fire scenario would not damage structural steel, there would be no need to examine lesser fire scenarios in other locations.

Accordingly, please discuss whether the evaluation for potential damage to exposed structural steel was performed for ALL locations meeting the two requirements stated above, and if not, please justify the decision to only assess the most limiting fire scenario and location.

Response: Calculation HNP-F/PSA-0079, Attachment 8, "Treatment of Exposed Structural Steel," states:

> Structures at HNP were reviewed to assess the potential for large oil fires capable of causing failure of structural steel. With the exception of the Turbine Building, there are no large oil hazards capable of sustaining fire temperatures for structural damage. The Turbine Generator Building would be the only structure at HNP which

satisfies the above criteria, both from the exposed structural steel and a high hazard fire source perspective.

Therefore, the evaluation for potential damage to exposed structural steel was performed for all locations identified as meeting both (a) exposed structural steel is present; (b) a high-hazard fire source is present in that location.

HNP RAI 5-46

SRs Dispositioned Without a Focused Scope Peer Review

The findings from the February 2008 NRC staff audit associated with SRs FQ-E1, FQ-F1, and FQ-F2 have been dispositioned as now satisfied, but the Focused Scope Peer Review Team apparently did not review these items, or the review confirmed that the items were valid but corrective action was not complete.

Since these items were not resolved and subsequently reviewed by the Focused Scope Peer Review Team, please provide additional technical details of the specific changes made to resolve the issues involved with SRs FQ-E1, FQ-F1, and FQ-F2.

Response: The associated findings were reviewed at the Focused Peer Review, but could not be closed at that time because the results and documentation had not been finalized. The methods and draft results were used to assess the SRs to the extent possible and the Peer Review Report stated that the "... results were generally available which satisfy the SR requirements, but not documented in a final format."

These SRs are documentation requirements. The final quantification and documentation was completed prior to submittal of the LAR and provides documented values for CDF, LERF, a list ranking significant contributors, assumptions, sources of uncertainty, and other items consistent with the SRs.

HNP RAI 5-47

Cable Routing for Instrumentation

Findings associated with SRs Equipment Selection (ES)-C1, Post-Fire Human Reliability Analysis (HRA)-B3, and HRA-C1 deal with the assumption of independent cable routing for redundant instrumentation associated with operator manual actions. It is not clear from the resolution of these items whether this remains an assumption, or whether additional cable routing data was obtained.

Accordingly, please describe how this issue is being addressed, either by assuming independent cable routing, or by verification of cable routing, and provide the basis for why the assumption is valid (if still applicable).

Response: The fire model submitted in the LAR includes both human action with instrumentation with verified cable routing and instruments with assumed routing. The justification for human actions credited without circuit analysis and routing is documented in HNP-F/PSA-0077, Attachment 4. The general justification used is:

> This required action has redundancy and diversity in instrumentation. Multiple channels for indication are divisionally separated and thus should not share common raceways. Additional diverse indications are available that should mitigate effects of confusing indications. Based on this, the subject instrumentation does not need to be added to the Fire PRA Component List.

As a general practice, divisionally separated redundant instrument channels are not routed in the same cable trays or conduits. Instruments penetrating containment would also generally be routed through their associated train penetration rooms. Several of the assumed routings are associated with instruments credited in the safe shutdown analysis, requiring protection of these instruments from a fire impacting both divisions. Additionally, the diverse instrument requirement was added to provide margin by including additional parameters with greater likelihood of independent routing. Therefore, the assumption of independent cable routing for diverse and redundant, divisionally separated instruments is reasonable.

HNP RAI 5-48

Seismic – Fire Analysis

Attachment 7, "Seismic – Fire Analysis," to calculation HNP-F/PSA-0079, Revision 1, describes the capability to provide manual fire fighting capability to standpipes and hose stations by aligning the Emergency Service Water System to a portion of the Fire Protection System such that the Containment, Auxiliary Building and Fuel Handling Building may be able to support two 75 gallon per minute hose stations for local fire fighting.

The discussion in the calculation does not address how a potential fire in the Diesel Generator Building (fire areas 1-D-DGA, 1-D-DGB, 1-D-DTA, and 1-D-DTB) or Emergency Service Water Intake Structure (fire areas 12-I-ESWPA and 12-I-ESWPB) would be addressed after a Safe Shutdown Earthquake. Following an earthquake, it would appear that continued operation of Emergency Diesel Generators and Emergency Service Water Pumps may be important to risk.

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Please provide a discussion of the impact of a post-seismic fire in the following fire areas: 1-D-DGA, 1-D-DGB, 1-D-DTA, 1-D-DTB, 12-I-ESWPA, and 12-I-ESWPB. In the discussion, provide either a qualitative or quantitative assessment of the risk impact of a fire in these areas.

Response: NLS-86-315 contained the following discussion:

"The Company requests approval of a deviation from the requirements to provide-SSE hose stations in the following plant areas:

Plant Location:
a) Diesel Generator Building
b) Diesel Fuel Oil Storage Building
c) Emergency Service Water Intake Structure

Fire Area:
a) I-D-DGA, I-D-DGB, I-D-DTA, 1-D-DTB
b) 12-0-TA, 12-0-TB, 1-0-PA, 1-0-PB, 5-0-BAL
c) 12-I-ESWPA, 12-I-ESWPB

SSA Area:
a) FADDGA, FADDGB, FADDTA, FADDTB
b) FCOTKA, FCOTKB, FAOPA, FAOPB, FPOBAL
c) FCIESA, FCIESB

Other safe shutdown equipment within the SSA area:

a) Diesel Generator, Diesel Generator Day Tanks, and Diesel Control Panels
b) Diesel Generator Fuel Oil Transfer Pumps, Diesel Generator Storage Tanks
c) Emergency Service Water Pumps and Controls

The Company considers this deviation justified because:

- the above redundant safe shutdown equipment is separated from each other by three-hour rated barriers, which are Seismic Class I structures,
- these areas are provided with non-seismic fire protection systems, and
- the combustible loading in these areas is considered low, except in the case of the diesel day tank and storage tank area where the enclosures are Seismic Class I or ASME Section III.

CONCLUSION

Based on the fire protection provided and described above, CP&L believes that a commensurate level of protection has been provided in lieu of additional SSE hose stations as described in Section C.6,dU) of NUREG-0800."

The NRC accepted this configuration in Supplement 3 to the Safety Evaluation Report (NUREG 1038) as follows:

> *"On the basis of its review of the applicant's justification, which is based on the separation of the redundant safe-shutdown equipment located in the diesel generator building, the diesel fuel oil storage building, and the emergency service water (ESW) intake structure by seismic Category I 3-hour fire rated barriers, and the provision of alternative means of manual firefighting, the staff concludes that the standpipe system is acceptable. The lack of seismically qualified hose stations in the diesel generator and fuel oil storage buildings and the emergency service water (ESW) intake structure is an acceptable deviation from Section C.6.c of BTP CMEB 9.5-1."*

As noted above, any fire that would occur in any of these areas would be isolated by the seismic three-hour fire barriers from the redundant safe shutdown path. The buildings in question are remotely located with regards to other power block structures, so a fire would not be expected to involve any other structure. In addition, the equipment in these areas that are capable of causing a challenging fire, such as large combustible liquid quantities, high voltage electrical panels, are themselves seismically qualified, which reduces the overall probability they would be subject to a seismically induced fire.

HNP RAI 6-1.1

Monitoring Program Performance Criteria

In the response to HNP RAI 6-1 dated August 13, 2009, item number 3 states that acceptable levels of availability, reliability and performance criteria are based on Fire PRA insights and accepted industry guidance such as Electric Power Research Institute (EPRI) Technical Report 1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide." In addition, item number 4 states that "unacceptable levels of availability, reliability and performance will be triggered by the established action levels."

This response corresponds well with the requirement in NFPA 805 Section 2.6.1, "Availability, Reliability, and Performance Levels," which states that "acceptable levels of availability, reliability and performance shall be established." However, the October 9, 2009, revised NFPA 805 LAR states: "The performance criteria used should be availability, reliability or condition monitoring, as appropriate."

There appears to be a change in intent with this wording change. The NRC staff understands that performance with regard to fire protection systems and equipment should relate to measurement of physical attributes that demonstrate the ability to functionally deliver some needed aspect in order to meet the nuclear safety performance criteria. For example, if the

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
RESPONSE TO SECOND REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE LICENSE AMENDMENT REQUEST TO ADOPT
NFPA 805, "PERFORMANCE-BASED STANDARD FOR FIRE
PROTECTION FOR LIGHT WATER REACTOR GENERATING PLANTS"

required component was a pump, performance would mean the ability to meet certain developed head and flow criteria. Changing the wording to condition monitoring changes the focus to address such things as vibration monitoring and thermographic monitoring for a pump, and thrust and/or torque measurement (MOVATS, VOTES, VIPER, etc.) for a valve. Not all of these activities would necessarily verify the required performance aspects of the component.

Accordingly, please provide an explanation of how condition monitoring fully meets the intent of the requirement in NFPA 805 Section 2.6.1 to include performance in the monitoring program.

Response: Acceptable levels of availability, reliability and performance criteria are based primarily on FPRA insights and accepted industry guidance such as *EPRI/NMAC Technical Report 1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide."* The method that will be used is a database called Plant Equipment Reliability Management Information Tool (PERMIT). PERMIT is used for Component Analysis and System Monitoring and is the tool used for documenting FPRA scoping, safety significant fire compartments, performance criteria, and performance monitoring data.

There was no intent to change the process of performance monitoring applied to those systems where reliability and availability can be reasonably monitored. The wording in the LAR, Section 4.6.2, "Overview of Post-Transition NFPA 805 Monitoring Program" has been revised to the earlier terminology.