Enclosure 3 to SERIAL: HNP-08-061

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT TO ADOPT NFPA 805 PERFORMANCEBASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTOR GENERATING PLANTS (2001 EDITION)

TRANSITION REPORT
(Redacted Version)

Progress Energy
Shearon Harris Nuclear Power Plant
Docket 50-400

Transition to 10 CFR 50.48(c) - NFPA 805

***Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants,* 2001 Edition**

Progress Energy

Transition Report

Revision 0

May 29, 2008

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Executive Summary

Progress Energy determined to transition the Shearon Harris Nuclear Power Plant (HNP) fire protection licensing basis to a performance-based alternative based upon National Fire Protection Association 805 (NFPA 805) per 10 CFR 50.48(c) in February 2005. HNP is one of two Pilot Plants for the transition process.

The transition process consisted of a review and update of HNP documentation, including the development of a Fire Probabilistic Risk Assessment (PRA) using NUREG/CR 6850 as guidance. The transition project was implemented using a comprehensive project plan and a number of technical meetings with the Nuclear Regulatory Commission (NRC) staff as part of the Pilot Plant process. This Transition Report summarizes the transition process and results. This Transition Report contains information:

- Required by 10 CFR 50.48(c).
- Recommended by guidance document Nuclear Energy Institute (NEI) 04-02 Revision 1 and appropriate Frequently Asked Questions (FAQs).
- Recommended by guidance document Regulatory Guide 1.205.

Section 4 of the Transition Report provides a summary of compliance with NFPA 805 requirements, addressing the following major topics:

- Fundamental Fire Protection Program Elements and Minimum Design Requirements
- Nuclear Safety Performance Criteria Transition Review
- Non-Power Operational Modes
- Radioactive Release Performance Criteria
- Change Evaluations
- Monitoring Program
- Program Documentation, Configuration Control, and Quality Assurance

Regulatory evaluations are included in Section 5 of the transition report and associated attachments, including:

- License condition changes,
- Changes to technical specifications, orders, and exemptions, and
- Significant hazards and environmental considerations.

The attachments to the Transition Report include detail to support the transition process and results.

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) has adopted an alternative rule for fire protection requirements at nuclear power plants, 10 CFR 50.48(c). Progress Energy has implemented the process for transitioning from its current fire protection licensing basis for the Shearon Harris Nuclear Power Plant (HNP) to compliance with the new requirements. This document describes the transition process applied by Progress Energy for HNP and the results that demonstrate compliance with the new requirements.

1.1 Background

1.1.1 NFPA 805 – Requirements and Guidance

On July 16, 2004 the Nuclear Regulatory Commission amended 10 CFR Part 50.48, *Fire Protection*, to add a new subsection, 10 CFR 50.48(c), that established acceptable fire protection requirements. The change to 10 CFR 50.48 endorses, with exceptions, the National Fire Protection Association's 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition (NFPA 805), as an alternative for demonstrating compliance with 10 CFR 50.48 Section (b) and Section (f).

Compliance with this rule may be adopted as an acceptable alternative method for complying with either 10 CFR 50.48 (b), for plants licensed to operate before January 1, 1979, or the fire protection license conditions for plants licensed to operate after January 1, 1979, or 10 CFR 50.48 (f), plants shutdown in accordance with 10 CFR 50.82(a)(1).

The Nuclear Energy Institute (NEI) developed NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c), to assist licensees in adopting NFPA 805 and make the transition from their current fire protection licensing basis to one based on NFPA 805. The NRC issued a Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, which endorses NEI 04-02, in May 2006.

A depiction of the primary document relationships is shown below:

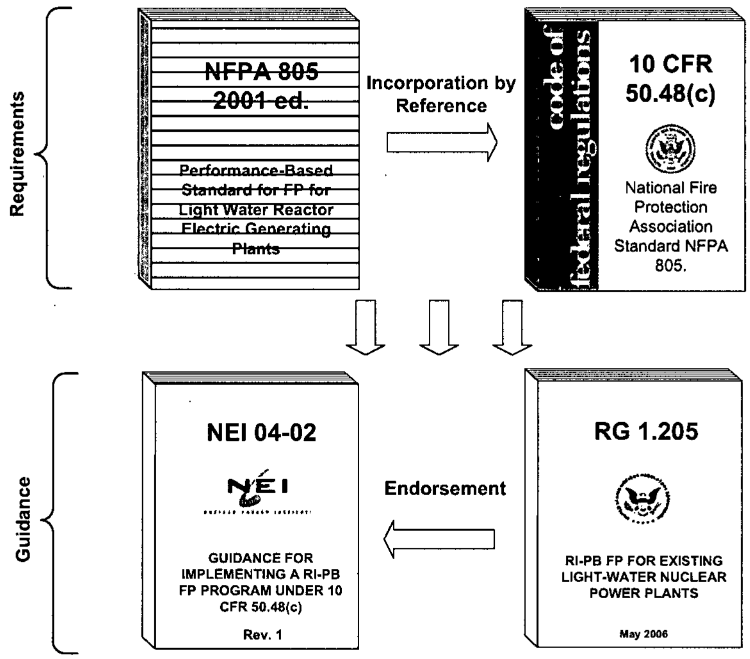

Figure 1-1 NFPA 805 Transition – Implementation Requirements/Guidance

1.1.2 HNP Transition to 10 CFR 50.48(c)

1.1.2.1 Start of Transition

HNP decided to transition its fire protection licensing basis to the performance-based alternative in 10 CFR 50.48(c). A letter of intent was submitted by Progress Energy to the NRC on June 10, 2005 (ML051720404) for HNP to adopt NFPA 805 in accordance with 10 CFR 50.48(c). This letter of intent also addressed other Progress Energy plants (Brunswick Steam Electric Plant, H.B. Robinson Steam Electric Plant Unit No. 2, and Crystal River Unit 3 Nuclear Generating Plant). The letter of intent requested three years of enforcement discretion and proposed that HNP be considered a Pilot Plant for the NFPA 805 transition process.

The NRC responded to Progress Energy on September 19, 2005 (ML052140391). In the response, the NRC agreed that HNP should be an NFPA 805 Transition Pilot Plant. The NRC also sent a letter to Progress Energy on April 29, 2007, granting a third year of enforcement discretion (ML070590625).

1.1.2.2 HNP Transition Process

The HNP NFPA 805 transition was conducted as part of a fleet Fire Protection Initiatives Project for each Progress Energy nuclear site. The Fire Protection Initiatives Project included high level activities to:

- Complete Safe Shutdown Analysis Reconstitution (activities started in 2003)
- Develop Fire PRAs using NUREG/CR-6850, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, as guidance and revise the Internal Events PRA to support the Fire PRAs
- Transition to 10 CFR 50.48(c)/NFPA 805

The project was implemented using a comprehensive project plan and individual procedures/instructions for individual scopes of work. These procedures/instructions (e.g., Progress Energy Project Instruction "FPIP" series procedures referenced in this report) were developed for the purposes of NFPA 805 transition. Appropriate technical content from these procedures were and will be incorporated into technical documents and configuration control procedures, as required.

1.1.2.3 NFPA 805 Pilot Plant Summary

The HNP NFPA 805 transition underwent a series of reviews and observation meetings as part of the Transition Pilot Plant process, with the following goals:

- Increase communication between the NRC and transitioning licensees
- Develop transition lesson learned reports from observation visits
- Improve the NFPA 805 Regulatory Guide and Inspection Procedures
- Develop License Amendment Request and Safety Evaluation Report templates

A summary of the major Pilot Plant activity is shown in Table 1-1:

Table 1-1 NFPA 805 Pilot Observation Meeting Summary

Date	Location	Summary
11/7/05-11/11/05	Charlotte, NC	Pilot Observation Meeting (ML060250033, ML060250034)
3/27/06—3/30/06	Raleigh, NC	Pilot Observation Meeting (ML061500468, ML061520285)
10/16/06-10/19/06	Seneca, SC Oconee	Pilot Observation Meeting (ML070280007, ML070320285)
11/6/06-11/8/06	Raleigh, NC	Pilot Observation Meeting (ML063330521, ML070820251, ML063310386, ML071210207, ML071060267)
3/6/07-3/8/07	Raleigh, NC	Pilot Observation Meeting (ML070950030, ML070960489, ML071160447)
5/30/07-6/1/07	Raleigh/Apex, NC	Pilot Observation Meeting / Public Meeting (ML071930362, ML071930339)
7/10/07-7/13/07	Seneca, SC Oconee	Pilot Observation Meeting / Public Meeting (ML072270014, ML072610448, ML072610455, ML072140380)
8/6/07-8/9/07	Bethesda, MD	Pilot Observation Meeting / Public Meeting (ML072830064, ML072890127, ML072910745)
11/5/07-11/8/07	Atlanta, GA NRC Region II Offices	Pilot Observation Meeting / Public Meeting (ML073321171, ML073270905)
12/7/07	Rockville, MD NRC Headquarters	Pilot Observation Meeting / Public Meeting – FAQ 07-0040, Non-Power Operations (ML073241052)
12/12/07	Washington, DC NEI Headquarters	Pilot Observation Meeting / Public Meeting – Fire PRA Human Reliability Analysis (HRA) and Operator Manual Action Reconciliation (ML073371166)

Table 1-1 NFPA 805 Pilot Observation Meeting Summary

Date	Location	Summary
1/7/08-1/8/08	Raleigh, NC	Pilot Observation Meeting / Public Meeting ; Review of LAR/Transition Report detail. (ML080450128, ML080450058)
4/15/08-4/16/08	Charlotte, NC	Pilot Observation Meeting / Public Meeting ; Review of LAR/Transition Report detail, UFSAR, MSO and OMA resolution. (ADAMS Reference Pending)

In addition to the Pilot Plant Observation Meetings, NEI established an NFPA 805 Task Force, to ensure the successful implementation of NEI 04-02. The Task Force provides support to plants transitioning to 10 CFR 50.48(c), risk-informed performance based regulatory framework. This includes support for the resolution of issues that may surface during the pilot process. The Task Force holds monthly meetings to communicate status of the Pilot Plant activities, provide examples of Pilot Plant transition products, and to provide non-pilot transitioning plants a forum to resolve/discuss technical issues arising from their transitions.

1.2 Purpose

The purpose of the HNP Transition Report is as follows:

(1) Describe the process implemented by Progress Energy to transition the HNP fire protection program to demonstrate compliance with the requirements in 10 CFR 50.48(c);

(2) Summarize the results of HNP's transition process;

(3) Explain the bases for Progress Energy's conclusions that the HNP fire protection program, with certain modifications, complies with those requirements; and

(4) To describe the new HNP fire protection licensing basis.

2.0 OVERVIEW OF EXISTING FIRE PROTECTION PROGRAM

The HNP license condition 2.F states:

> *"F. Fire Protection Program (Section 9.5.1)*
>
> *Carolina Power & Light Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility as amended and as approved in the Safety Evaluation Report (SER) dated November 1983 (and supplements 1 through 4), and the Safety Evaluation dated January 12, 1987, subject to the following provision below. The licensees may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire."*

3.0 TRANSITION PROCESS

3.1 Background

The process for transitioning from compliance with the current fire protection licensing basis to the new requirements is described in general in Section 4.0 of NEI 04-02. It contains the following steps: (1) licensee determination to transition the licensing basis and devote the necessary resources to it; (2) submit a Letter of Intent to the NRC stating the licensee's intention to transition the licensing basis in accordance with a tentative schedule; (3) conduct the transition process to determine the extent to which the current fire protection licensing basis supports compliance with the new requirements and the extent to which additional analyses, plant and program changes, and alternative methods and analytical approaches are needed; (4) file a License Amendment Request (LAR); (5) complete transition activities that can be completed prior to the receipt of the License Amendment; (6) receive License Amendment; and (7) complete implementation of the new licensing basis.

3.2 NFPA 805 Process

Section 2.2 of NFPA 805 establishes the general process for demonstrating compliance with NFPA 805. This process is illustrated below in Figure 3-1. It shows that except for the fundamental fire protection requirements, compliance can be achieved on a fire area basis either by deterministic or performance-based methods. (The NRC permits licensees to use performance-based methods to comply with the fundamental fire protection requirements but those applications must be approved through the NRC's license amendment process, as discussed above.) HNP implemented this process by first determining the extent to which its current fire protection program supported findings of deterministic compliance with the requirements in NFPA 805. Risk-informed, performance-based methods were then applied to the requirements for which deterministic compliance could not be shown.

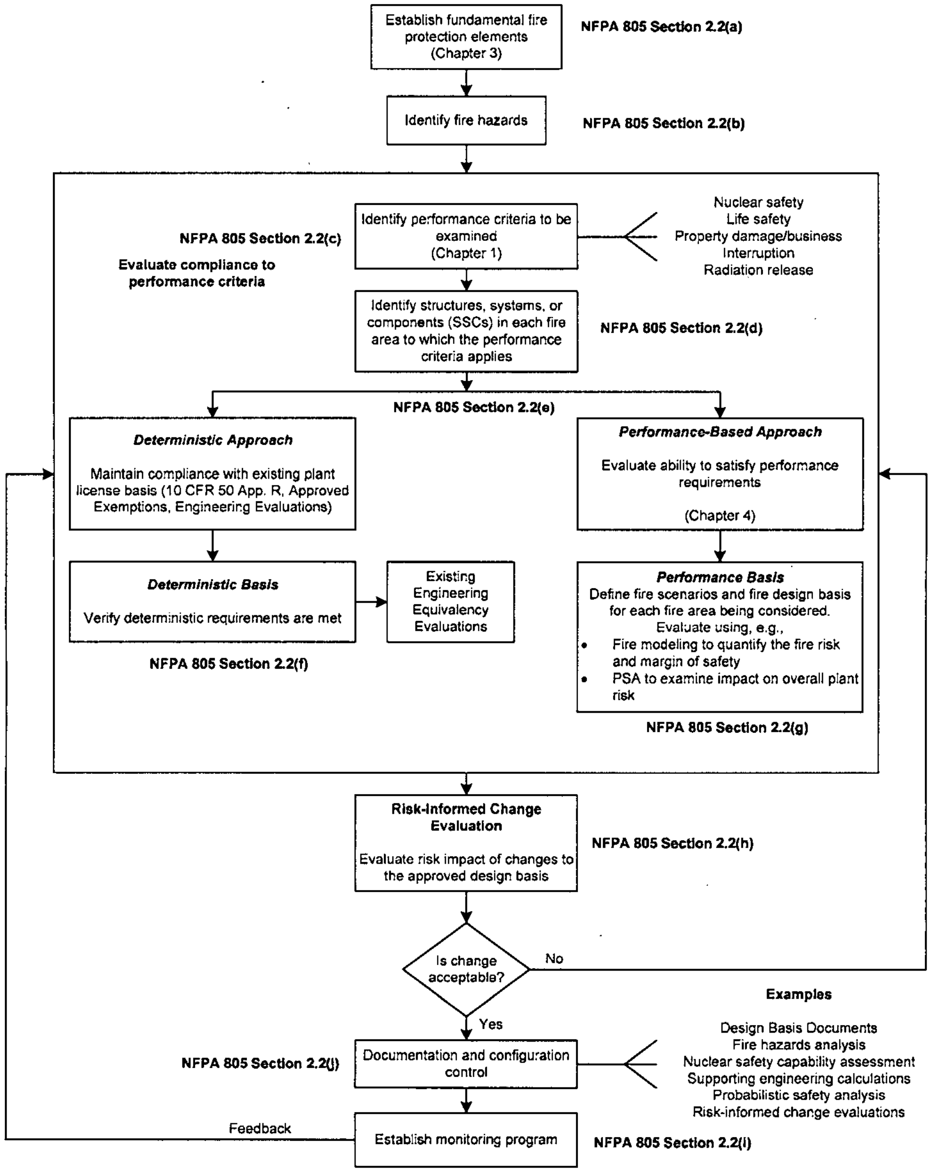

Figure 3-1 NFPA 805 Process [NEI 04-02 Figure 3-1 based on Figure 2-2 of NFPA 805][1]

3.3 NEI 04-02 – NFPA 805 Transition Process

NFPA 805 contains technical processes and requirements for a risk-informed, performance-based fire protection program. NEI 04-02 was developed to provide guidance on the overall process (programmatic, technical, and licensing) of the transition from a traditional deterministic fire protection licensing basis to a new one based upon NFPA 805, as shown below in Figure 3-2.

[1] Note: 10 CFR 50.48(c) does not endorse Life Safety and Plant Damage / Business Interruption goals, objectives and criteria

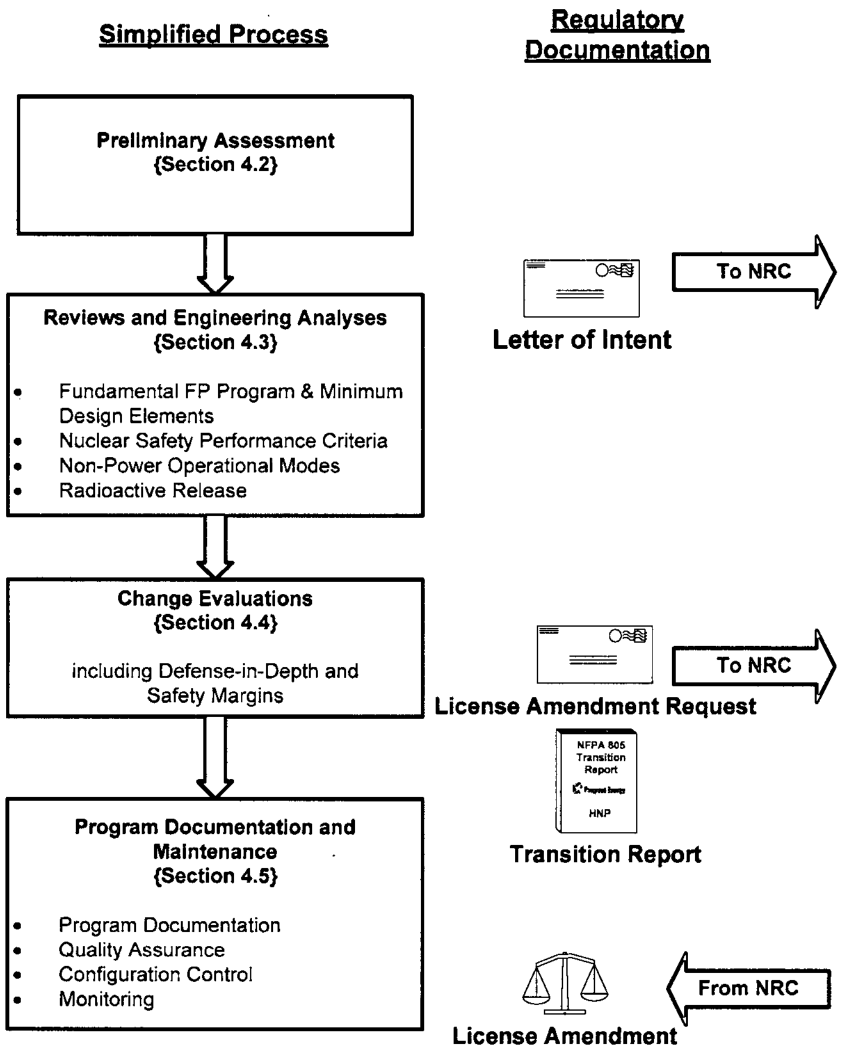

Figure 3-2 Implementing the New Licensing Basis [NEI 04-02 Figure 3-3]

Section 4.0 of NEI 04-02 describes the detailed process for assessing a fire protection program for the extent to which it complies with NFPA 805, as shown below in Figure 3-3. HNP conducted the detailed evaluation processes by establishing teams comprised of knowledgeable plant personnel. The assessment processes used by these teams and the results of the assessments are discussed in detail below.

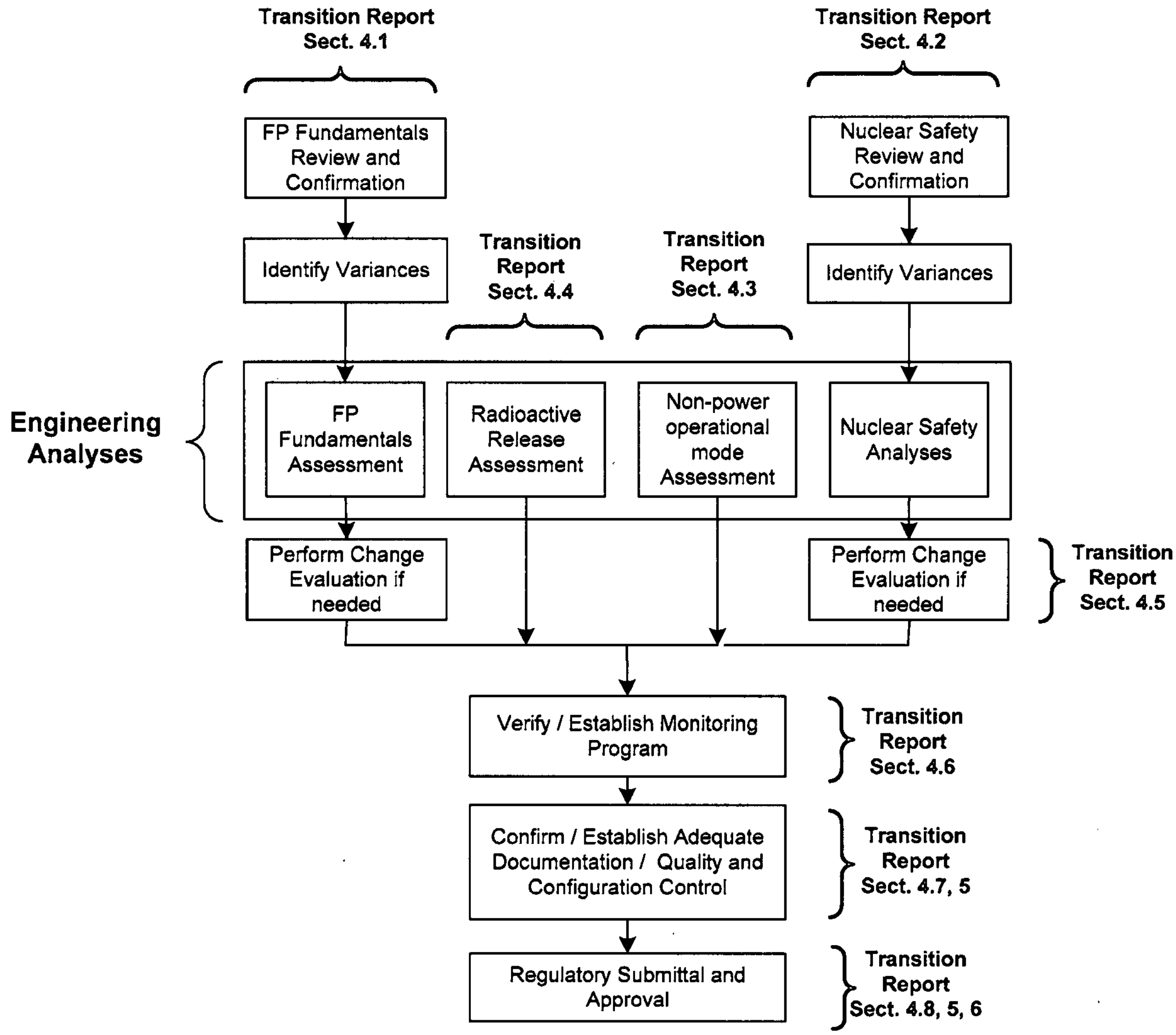

Figure 3-3 Transition Process (Simplified) [based on NEI 04-02 Figure 4-1]

3.4 NEI 04-02 Frequently Asked Questions (FAQs)

The NRC staff has worked with NEI and two Pilot Plants (Oconee Nuclear Station and Harris Nuclear Plant) to define the licensing process for transitioning to a new licensing basis under 10 CFR 50.48(c) and NFPA 805. Both the NRC and the industry recognized the need for additional clarifications to the guidance provided in Regulatory Guide 1.205 and NEI 04-02, Revision 1. The NFPA 805 Frequently Asked Question (FAQ) Process was jointly developed by NEI and NRC to facilitate timely clarifications of NRC positions. This process is described in a letter from the NRC dated July 12, 2006, to NEI (ML061660105) and in RIS 2007-19, Process For Communicating Clarifications Of Staff Positions Provided In Regulatory Guide 1.205 Concerning Issues Identified During The Pilot Application of NFPA 805, dated August 20, 2007.

Under the FAQ Process, transition issues were submitted to the NEI NFPA 805 Task Force for review, and subsequently presented to the NRC during public FAQ meetings. Once an acceptable FAQ was submitted to the NRC, the NRC staff issued a memorandum to file to indicate that the revised FAQ was acceptable. NEI 04-02 will be revised to incorporate the approved FAQs. Final closure will occur when Regulatory Guide 1.205, which endorses the new revision of NEI 04-02, is approved by the NRC.

Attachment H contains the FAQs that were used to clarify the guidance in Regulatory Guide 1.205 and NEI 04-02 and in the preparation of this License Amendment Request.

4.0 COMPLIANCE WITH NFPA 805 REQUIREMENTS

4.1 Fundamental Fire Protection Program Elements and Minimum Design Requirements

The Fundamental Fire Protection Program and Design Elements are established in Chapter 3 of NFPA 805. Section 4.3.1 of NEI 04-02 sets out a systematic process for determining the extent to which the current licensing basis meets these criteria and for identifying the fire protection program changes that would be necessary for complete compliance. Appendix B-1 of NEI 04-02 provides guidance on documenting the comparison of the fire protection program fundamentals of NFPA 805 Chapter 3 to the current approved fire protection program.

4.1.1 Overview of Evaluation Process

The comparison of the HNP fire protection program to NFPA 805 Chapter 3 (NEI 04-02 Table B-1) was performed using the methodology contained in Progress Energy Project Instruction FPIP-0120, NFPA 805 Chapter 3 Fundamental Transition, and the guidance contained in FAQ 07-0036, Incorporation of Pilot Plant Lessons Learned – NEI 04-02 Table B-1. The methodology depicted in Figure 4-1 is outlined below.

Each section and subsection of NFPA 805 Chapter 3 was reviewed against the current fire protection program. For each section/subsection the appropriate compliance statement was determined:

- o Complies
- o Complies with Clarification
- o Complies Via Previous NRC Approval
- o Complies with Use of Existing Engineering Equivalency Evaluations (EEEEs)
- o License Amendment Required

Complies - For those sections/subsections determined to comply with NFPA 805:

- An implementing reference was provided for the compliance statement in the Reference Document Field.
    - For administrative requirements such as required procedures, control of combustibles and ignition sources, fire brigade requirements, etc., a reference was provided to the site or corporate procedure that provides the control required by the program. The reference was made to the highest level document that satisfies the requirement. For example, the requirement for ensuring the plant has procedures for inspection and maintenance of systems would reference the program procedure that establishes this requirement, rather than a list of all testing and inspection procedures.
    - For system requirements such as water supply, automatic detection and suppression, manual suppression, fire extinguishers, and fire barriers, the following was provided:
        - For active systems (including extinguishers), a reference to the code compliance evaluation with a summary of evaluated deviations, the edition of code and a statement of the ability of the system(s) to meet its functional requirement.

- For passive systems, a reference to specifications, design documents or test reports that demonstrate compliance with the systems.

Complies with Clarification - For those sections/subsections determined to comply with NFPA 805 with clarification:

- An implementing reference was provided for the compliance statement in the Reference Document Field.
- The clarification was provided in the compliance basis field.

Complies Via Previous NRC Approval - For those sections/subsections determined to comply via previous NRC approval:

- Appropriate excerpts from submittals regarding the issue for which previous approval is being claimed were included in the reference document field.
- Appropriate excerpts from the NRC documents that provide the formal approval of the fire protection system/feature were included.
- An implementing reference was provided for the compliance statement in the Reference Document Field.

Complies with Use of Existing Engineering Equivalency Evaluations (EEEEs) - For those sections/subsections determined to comply with the use of engineering equivalency evaluations:

- An implementing reference was provided for the compliance statement in the Reference Document Field.
- A summary of the bases of acceptability of the engineering evaluation was provided in the compliance basis field.

License Amendment Required - For those sections/subsections determined to need a license amendment request:

- An implementing reference was provided for the compliance statement in the Reference Document Field.
- A summary of the bases of acceptability of the engineering evaluation was provided in the compliance basis field.

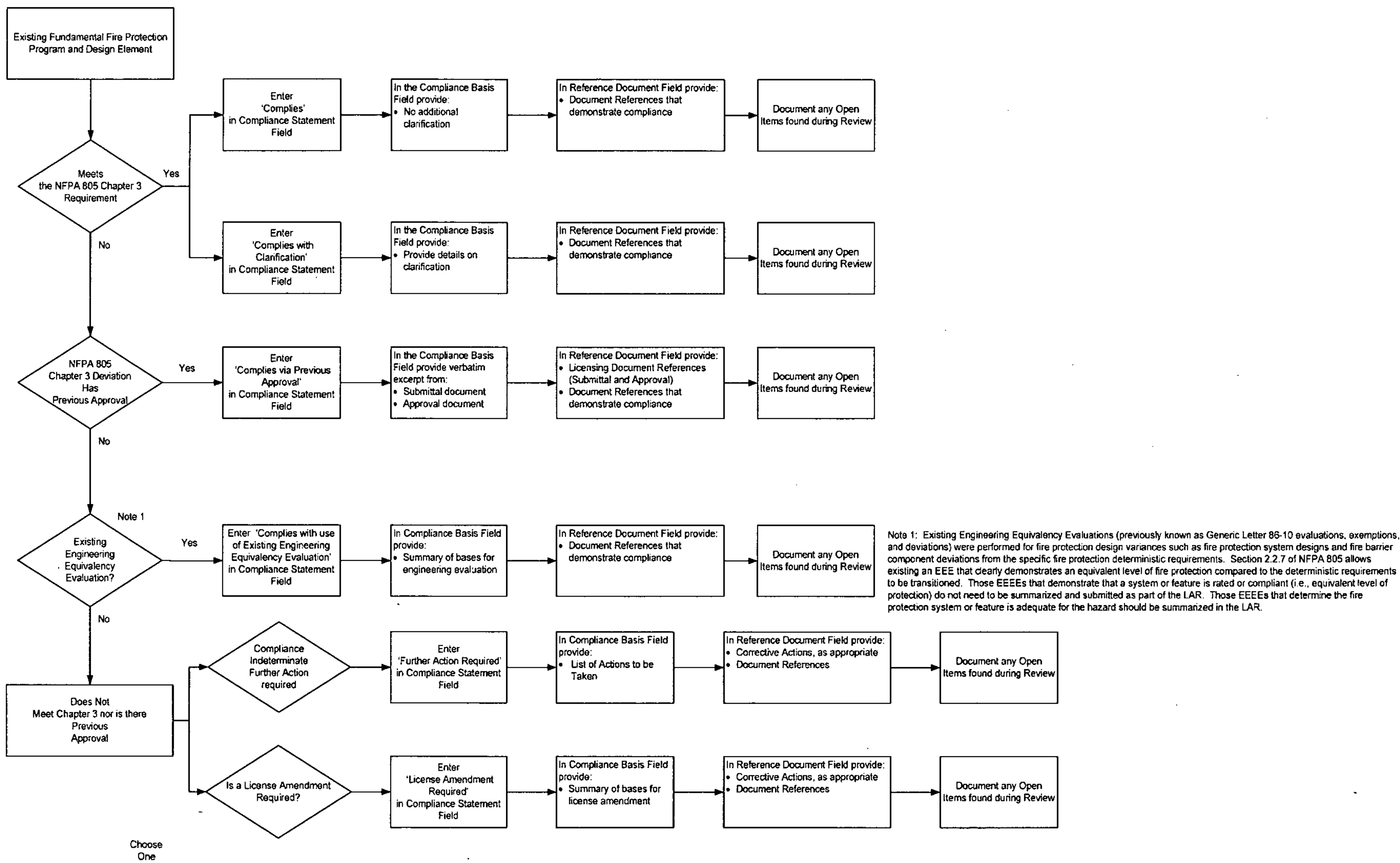

Figure 4-1 - Fundamental Program and Design Elements Transition Process [based on NEI 04-02 Figure 4-2/FAQ 07-0036]

4.1.2 Results of the Evaluation Process

4.1.2.1 NFPA 805 Chapter 3 Requirements Met or Previously Approved by the NRC

Results of the review of NFPA 805 Chapter 3 requirements are included in Attachment A to the Transition Report. References to the document(s) that justify that position are also included in Attachment A.

4.1.2.2 NFPA 805 Chapter 3 Requirements Not Previously Approved by NRC

The following sections of NFPA 805 Chapter 3 are not specifically met nor do previous NRC approvals of alternatives exist:

- 3.5.16 – Approval is requested for non-Fire Protection related uses of the fire protection water supply. Under normal circumstances there would be no water going to this system as makeup. Also, controls and communications are in place to ensure fire protection water demand can be secured immediately if a fire occurs.
- 3.6.5 – Approval is requested for the HNP cross-connect to Emergency Service Water (ESW) which can degrade the ESW function. Only one train is declared inoperable while cross-connected. This is to assist in extinguishing a fire as a temporary measure. The system would be restored to its normal condition when the fire was extinguished. This change is to maintain the current plant operational flexibility.

The specific deviation and a discussion of its equivalence to the deterministic requirements of NFPA 805 Chapter 3 are provided in Attachment L of the Transition Report.

4.1.2.3 NFPA 805 Chapter 3 Requirements Requiring Clarification of Prior NRC Approval

There are no sections of NFPA 805 Chapter 3 that require clarification of prior NRC approval.

4.1.3 Definition of Power Block

The definition of "Power Block" and "Plant" as referenced in NFPA 805, Chapter 3 was clarified in FAQ 06-0019. "Power Block" and "Plant" refer to structures that have equipment required for nuclear plant operations, such as containment, auxiliary building, service building, control building, fuel building, radioactive waste, water treatment, turbine building, and intake structures, or structures that are identified in the facility's current license basis (CLB).

The "Power Block" structures for HNP are identified in Attachment I.

4.2 Nuclear Safety Performance Criteria Transition Review

The Nuclear Safety Performance Criteria are established in Section 1.5 of NFPA 805. Chapter 4 of NFPA 805 provides the methodology to determine the fire protection systems and features required to achieve the performance criteria outlined in Section 1.5. Section 4.3.2 of NEI 04-02 sets out a systematic process for determining the extent to which the current licensing basis meets these criteria and for identifying the fire protection program changes that would be necessary for complete compliance. Appendix B-2 of NEI 04-02 provides guidance on documenting the transition of Nuclear Safety Capability Assessment Methodology and the fire area compliance strategies.

4.2.1 Nuclear Safety Capability Assessment Methodology Review

4.2.1.1 Overview of Evaluation Process

The comparison of the HNP fire protection program to NEI 00-01, Guidance for Post-Fire Safe Shutdown Circuit Analysis, Revision 1, Chapter 3 (NEI 04-02 Table B-2) was performed using the methodology contained in Progress Energy Project Instruction FPIP-0127, NFPA 805 Nuclear Safety Capability Assessment Transition Review, and the guidance contained in FAQ 07-0039, Lessons Learned – NEI 04-02 B-2 and B-3 Tables. The methodology steps depicted in Figure 4-2 are outlined below.

Step 1 – Documentation was assembled, including industry and plant-specific information.

Step 2 - Applicability of NEI 00-01 sections was determined and documented. The NFPA 805 2.4.2 sections were correlated to the corresponding section of NEI 00-01 Chapter 3. Based upon the content of the NEI 00-01 methodology statements, the applicability of the sections was determined.

Step 3 – A comparison of plant-specific safe shutdown methodology to applicable sections of NEI 00-01 was performed. A determination was made of the consequences of the failure to maintain strict alignment with the guidance in NEI 00-01. The alignment with the NEI guidance was determined and the basis for the alignment statement was provided.

Step 4 – Open Items associated with the review of the NEI 00-01 guidance were documented.

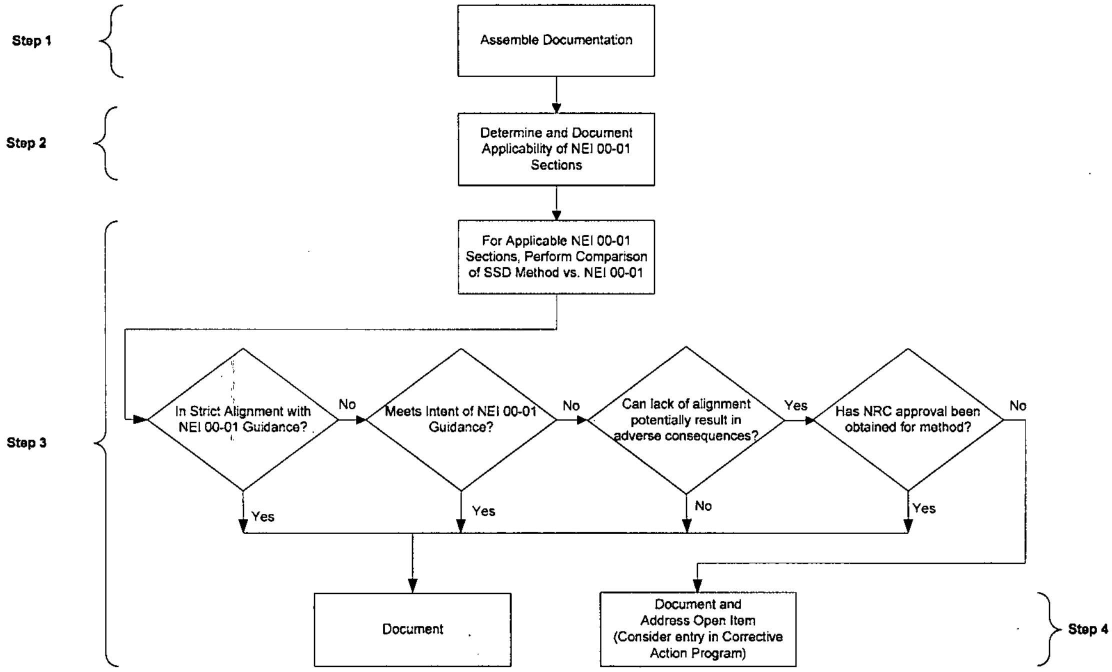

Figure 4-2 – Summary of Nuclear Safety Methodology Review Process (FAQ 07-0039)

4.2.1.2 Results from Evaluation Process

The specific results of the HNP Nuclear Safety Capability Assessment Methodology Review are included in Attachment B. The HNP methodology aligns with the guidance in NEI 00-01.

In addition, the NEI 00-01 methodology has been supplemented by the guidance for analyzing Multiple Spurious Operations (MSO) (FAQ 07-0038) and the guidance for evaluating the additional risk associated with the use of Recovery Actions as compliance strategies (FAQ 07-0030). Attachment F contains the details of the MSO Methodology. Attachment G contains the Recovery Action Methodology.

4.2.2 Fire Area-by-Fire Area Transition

4.2.2.1 Overview of Evaluation Process

The Fire Area-by-Fire Area Transition of the HNP fire protection program (NEI 04-02 Table B-3) was performed using the methodology contained in Progress Energy Project Instruction FPIP-0127, NFPA 805 Nuclear Safety Capability Assessment Transition Review, and the guidance contained in FAQ 07-0039, Lessons Learned – NEI 04-02 Table B-2 and Table B-3. The methodology for performing the Fire Area-by-Fire Area Transition, depicted in Figure 4-3, is outlined below.

Step 1 – Documentation Gathering - Documentation was assembled, including industry and plant-specific fire area analysis analytical and licensing basis documents.

Step 2 – Performance Goal Assessment - Accomplishment of the NFPA 805 performance goals was assessed and documented in summary level form for the selected fire area.

Step 3 –Fire Area Licensing Action Review - A review of the licensing aspects of the selected fire area was performed and documented.

Step 4 – Engineering Evaluation Review - A review of appropriate engineering evaluations to determine and assess the basis for acceptability was performed. The purpose of the evaluation and the results of the review were documented.

Step 5 – Documentation - The results of the review were documented. This step included documenting Fire Protection Systems and Features Determination, Fire Suppression Activities, and Open Items/Change Evaluations.

NRC requirements and guidance (e.g., 10CFR 50.48(c) and Regulatory Guide 1.205) address information to be included in this LAR/Transition Report. An open item tracking mechanism was put in place during transition to ensure those items were appropriately captured in this report.

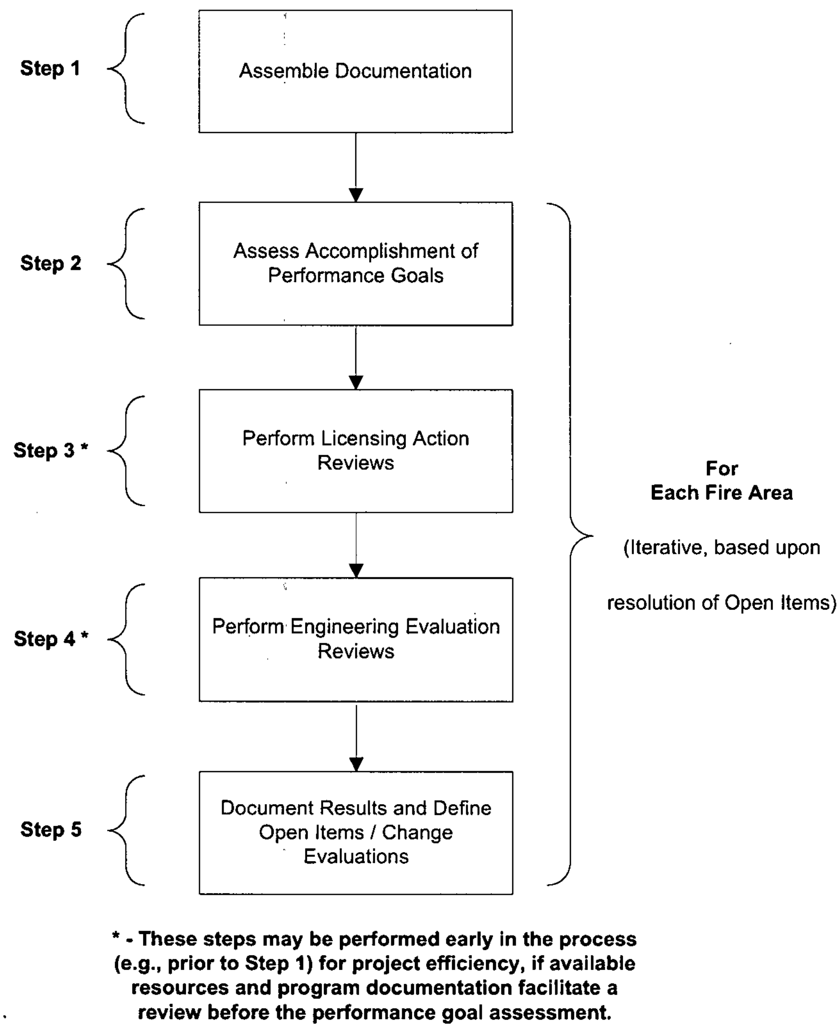

Figure 4-3 – Summary of Fire Area-by-Fire Area Review (FAQ 07-0039)

4.2.2.2 Results of the Evaluation Process

4.2.2.2.1 Results of the Existing Engineering Equivalency Evaluation Review

The Existing Engineering Equivalency Evaluation (EEEE) review was performed using the methodology contained in Progress Energy Project Instruction FPIP-0125, Transition of Existing Engineering Equivalency Evaluations, and the guidance contained in FAQ 07-0033, Transition of Existing Engineering Equivalency Evaluations. The methodology for performing the EEEE review included the following:

- Determination that the EEEE is not based solely on quantitative risk evaluations,
- Determination that the EEEE is an appropriate use of an engineering equivalency evaluation,
- Determination that the EEEE is of appropriate quality,
- Determination that the standard license condition is met,
- Determination that the evaluation reflects the plant as-built condition, and
- Determination that the EEEE is technically adequate.

Attachment J of the Transition Report contains the results of the EEEE review.

In accordance with the guidance provided in Regulatory Guide 1.205, Revision 0, Regulatory Position C.3.2.4 and FAQ 07-0033, EEEEs identified in Attachment J as "Include in LAR/TR – 'Yes'" are performance-based evaluations and as such are included in the LAR.

4.2.2.2.2 Results of the Licensing Action Review

The existing licensing action review was performed using the methodology contained in Progress Energy Project Instruction FPIP-0127, NFPA 805 Nuclear Safety Capability Assessment Transition Review. The methodology for the licensing action review included the following:

- Determination of the basis of acceptability.
- Determination that the basis of acceptability was still valid.

The licensing actions for HNP that were reviewed consisted of Deviation Requests and the associated Safety Evaluation Report and Supplements. Since HNP is a post-1979 plant, 10 CFR 50 Appendix R exemptions were not necessary and are therefore not applicable to HNP.

The licensing action review resulted in the identification of licensing actions that would be transitioned to the new 10 CFR 50.48(c) licensing basis and those that would no longer be necessary. Attachment K of the Transition Report contains the results of the Licensing Action Review.

As discussed in Attachment M, upon issuance of the new 10 CFR 50.48(c) license condition, the current HNP fire protection license condition will be superseded. It is HNP's understanding that implicit in the superseding of the current license condition, all prior fire protection program Safety Evaluation Reports and commitments will be superseded in their entirety.

4.2.2.3 Results of the Fire Area-by-Fire Area Review

Attachment C contains the results of the Fire Area Transition review (NEI 04-02 Table B-3). The NEI 04-02 Table B-3 includes the following summary level information for each fire area:

- Regulatory Basis – Both pre and post-transition regulatory bases are included.
- Performance Goal Summary – An overview of the method of accomplishment of each of the performance criteria in NFPA 805 Section 1.5 is provided.
- Reference Documents – Specific References to Operating Procedures and the Nuclear Safety Capability Assessment Documents are provided.
- Licensing Actions – Specific References to licensing actions whose content will remain part of the post-transition licensing basis and the Basis for Acceptability of that Licensing Action.
- Engineering Equivalency Evaluations – Specific References to Engineering Equivalency Evaluations that rely on determinations of adequate for the hazard that will remain part of the post-transition licensing basis and the Basis for Acceptability of that Engineering Equivalency Evaluation.
- Open Items – Specific references such as modifications, change evaluations, or procedural changes that are required to be included in the LAR. Refer to Section 4.5 for a discussion of the change evaluations performed during the transition process.

4.3 Non-Power Operational Modes

4.3.1 Overview of Evaluation Process

The current industry approach for evaluating risk during shutdown conditions involves qualitative and/or quantitative assessments and is based on NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management, as documented in NEI 04-02 FAQ 07-0040, Non-Power Operations Clarification. The strategy for controls/protection of equipment during Non-Power Operational (NPO) modes for plants adopting NFPA 805, is a combination of the normal fire protection program defense-in-depth actions and additional risk-informed steps based on the availability of systems and equipment needed to support Key Safety Functions (KSFs) and whether or not the plant is in a Higher Risk Evolution (HRE). The goal (as depicted in Figure 4-5) is to ensure that contingency plans are implemented when the plant is in an HRE and there is the possibility of losing a KSF due to fire. Additional controls/measures are evaluated during NPO modes where the risk is intrinsically high[2]. During low risk periods normal risk management controls and fire prevention/protection processes and procedures will be utilized.

The process to demonstrate that the nuclear safety performance criteria are met during non-power modes of operations involves the following steps:

1. Review existing Outage Management Processes
2. Identify Components/Cables
    a. Review plant systems to determine success paths that support each of the defense-in-depth KSFs, and then
    b. Identify cables required for the selected components and then determine their routing
3. Perform Fire Area Assessments (identify pinch points)
4. Manage risk associated with fire-induced vulnerabilities during the outage

These steps are described in the sections below, and the process is depicted on Figures 4-4 and 4-5.

4.3.1.1 Review Existing Outage Management Processes

To begin the process of assessing the fire protection plan for NPO modes, discussions were held between the Probabilistic Risk Assessment (PRA) Staff, Fire Protection, and the Outage

[2] According to Section 1.3.1, "Nuclear Safety Goal," of NFPA 805, "[t]he nuclear safety goal shall be to provide *reasonable assurance* that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition." As stated, this does not mandate a fire risk evaluation comparable to what would be expected during full power. Therefore, it is recognized that, for non-power operations, a "risk-informed" approach has been developed which addresses what is believed to be (and evidenced through the referenced studies) the most risk-significant Plant Operational States (POS) during non-power operations when including considerations of fire effects, namely total loss of a KSF. As such, these are expected to account for most, if not all, POSs that can be considered "intrinsically high" when considering fire effects." This approach, while compliant with 10 CFR 50.48(c), does not constitute a complete surrogate for a non-power risk evaluation since, under plant-specific conditions (believed to be relatively rare), there may be non-power POSs where less than total loss of a KSF (e.g., a reduction in the availability of credited paths ["redundancy decrease"] such that at least one path still remains), including consideration of fire effects, could result in a risk-significant contribution.

Management staff to determine the best way to integrate NFPA 805 fire protection aspects into existing Outage Management Processes.

Included in this review was a definition of what was considered an HRE. The HRE definition considered the following:

- Time to boil
- Reactor coolant system and fuel pool inventory
- Decay heat removal capability

In accordance with NUMARC 91-06, activities that may impact KSFs are limited and strictly controlled during HREs or infrequently performed evolutions.[3]

4.3.1.2 Identify Components/Cables

The identification of systems and components to be included in this NPO Review began with the identification of the Plant Operational States (POSs) that needed to be considered. The following discussion identifies the various operational states that a plant goes through during NPO, and which ones are the most risk significant. The definitions of the following simplified POSs are contained in NRC Inspection Manual IM0609, Appendix G, Attachment 2, Phase 2 Significance Determination Process Template for PWR During Shutdown, and are included here for use in reading Table 4-1.

<u>Pressurized Water Reactor (PWR) [IM0609, Appendix G Attachment 2]</u>

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

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

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

Disposition of the POSs (to determine which POSs required the identification of systems and components to support KSF) is provided in Table 4-1. For other non-power conditions (e.g., PWR Mode 3), normal fire protection program controls, processes and procedures will be used.

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[3] According to Section 1.3.1, "Nuclear Safety Goal," of NFPA 805, "[t]he nuclear safety goal shall be to provide *reasonable assurance* that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition." As stated, this does not mandate a fire risk evaluation comparable to what would be expected during full power. Therefore, it is recognized that, for non-power operations, a "risk-informed" approach has been developed which addresses what is believed to be (and evidenced through the referenced studies) the most risk-significant POSs during non-power operations when including considerations of fire effects, namely <u>total</u> loss of a KSF. As such, these are expected to account for most, if not all, POSs that can be considered "intrinsically high" when considering fire effects." This approach, while compliant with 10 CFR 50.48(c), does not constitute a complete surrogate for a non-power risk evaluation since, under plant-specific conditions (believed to be relatively rare), there may be non-power POSs where less than total loss of a KSF (e.g., a reduction in the availability of credited paths ["redundancy decrease"] such that at least one path still remains), including consideration of fire effects, could result in a risk-significant contribution.

Table 4-1 - PWR POS Disposition For Equipment Selection

POS/Configuration	Disposition	Discussion
POS 1 with SG Heat Removal Available	No additional reviews required under NEI 04-02, Section 4.3.3 (as modified by FAQ 07-0040) based upon previous risk reviews. Provide normal fire protection and prevention practices apply)	In this POS, if SGs are available in addition to RHR, significant redundancy and diversity exists for heat removal. Just having inventory in the SGs can provide substantial passive heat removal, providing additional time to recover other heat removal methods. Inventory control is not generally challenged during this POS.
POS 1 with SG Heat Removal Unavailable [Consider limiting to configurations where time to boil is less than 2 hours and/or RCS level is being changed]	Perform actions per NEI 04-02, Section 4.3.3 (as modified by FAQ 07-0040).	Without SG Heat Removal capability, heat removal is limited to RHR and potentially bleed and feed. RCS pressurization on loss of heat removal could render RHR unavailable due to high pressure. Activities in this POS often involve changing RCS level. During RCS level changes, the likelihood of loss of inventory control is higher, challenging the inventory control safety function.
POS 2	Perform actions per NEI 04-02, Section 4.3.3 (as modified by FAQ 07-0040).	This is the generally the highest risk configuration/POS for a PWR. Due to low inventory, times to core boil are low, typically on the order of 2 hours or less.
POS 3	Evaluate potential RCS drain paths that could be affected by fire	During this POS, substantial inventory exists to cope with an extended loss of active heat removal. Times to boil are often on the order of 16 or more hours. However, fire induced RCS draindown events can reduce margins substantially.

After identifying the POSs that require additional equipment evaluation for inclusion in the NPO review, the following steps were performed:

- Existing plant outage processes (outage management and outage risk assessments) to determine KSFs that support the POSs of concern were reviewed.
- Equipment relied upon to provide KSFs, including support functions, during the POSs to be evaluated, were determined.
- Equipment credited for achieving these KSFs were compared against the equipment credited for nuclear safety, including a comparison of required equipment positions/functions.
- For equipment not already credited (or credited in a different way, e.g., on versus off, open versus closed, etc.), the circuits were analyzed in accordance with the nuclear safety methodology. Cables to be included in the NPO review were identified.
- For cables that are not already credited in the nuclear safety capability assessment, routing of the cables was determined.

4.3.1.3 Perform Fire Area Assessments (identify pinch points)

Locations were identified where:

- Fires may cause damage to the equipment (and cabling) credited above, or
- KSFs are achieved solely by crediting recovery actions, (e.g., alignment of gravity feed).

4.3.1.4 Manage risk associated with fire-induced vulnerabilities during the outage

The management of risk associated with fire-induced vulnerabilities during NPO varies based on whether or not the plant is in an HRE as follows:

- During those NPO evolutions where risk is relatively low:

  The normal fire protection program defense-in-depth actions are credited for addressing the risk impact of those fires that potentially impact one or more trains of equipment that provide a KSF required during NPO modes, but would not be expected to cause the total loss of that KSF. The following actions are considered to be adequate to address minor losses of system capability or redundancy:

  - Control of Ignition Sources
    - Hot Work (cutting, welding and/or grinding)
    - Temporary Electrical Installations
    - Electric portable space heaters
  - Control of Combustibles
    - Transient fire hazards
    - Modifications
    - Flammable and Combustible liquids and gases
  - Compensatory Actions for fire protection system impairments
    - Openings in fire barriers
    - Inoperable fire detectors or detection systems
    - Inoperable fire suppression systems
  - Housekeeping

- During those NPO evolutions that are defined as HREs:

  Additional fire protection defense-in-depth measures are taken during HREs by:

  - Managing risk in fire areas that contain known pinch points.
  - Managing risk in fire areas where pinch points may arise because of equipment taken out of service.

  NUMARC 91-06 discusses the development of outage plans and schedules. A key element of that process is to ensure the KSFs perform as needed during the various outage evolutions. During outage planning, the NPO Fire Area Assessment is reviewed to identify areas of single-point KSF vulnerability during HREs to develop any needed contingency plans/actions. For those areas, combinations of the following options are implemented to reduce fire risk, depending upon the significance of the potential damage:

  - Prohibition or limitation of hot work in fire areas during periods of increased vulnerability
  - Verification of operable detection and/or suppression in the vulnerable areas
  - Prohibition or limitation of combustible materials in fire areas during periods of increased vulnerability
  - Plant lineup modifications (removing power from equipment once it is placed in its desired position)

- Provision of additional fire patrols at periodic intervals or other appropriate compensatory measures (such as surveillance cameras) during increased vulnerability
- Use of recovery actions to mitigate potential losses of KSFs
- Identification and monitoring in-situ ignition sources for “fire precursors” (e.g., equipment temperatures)

In addition, for KSF equipment removed from service during the HREs the impact is evaluated based on KSF equipment status and the NPO Fire Area Assessment to develop needed contingency plans/actions.

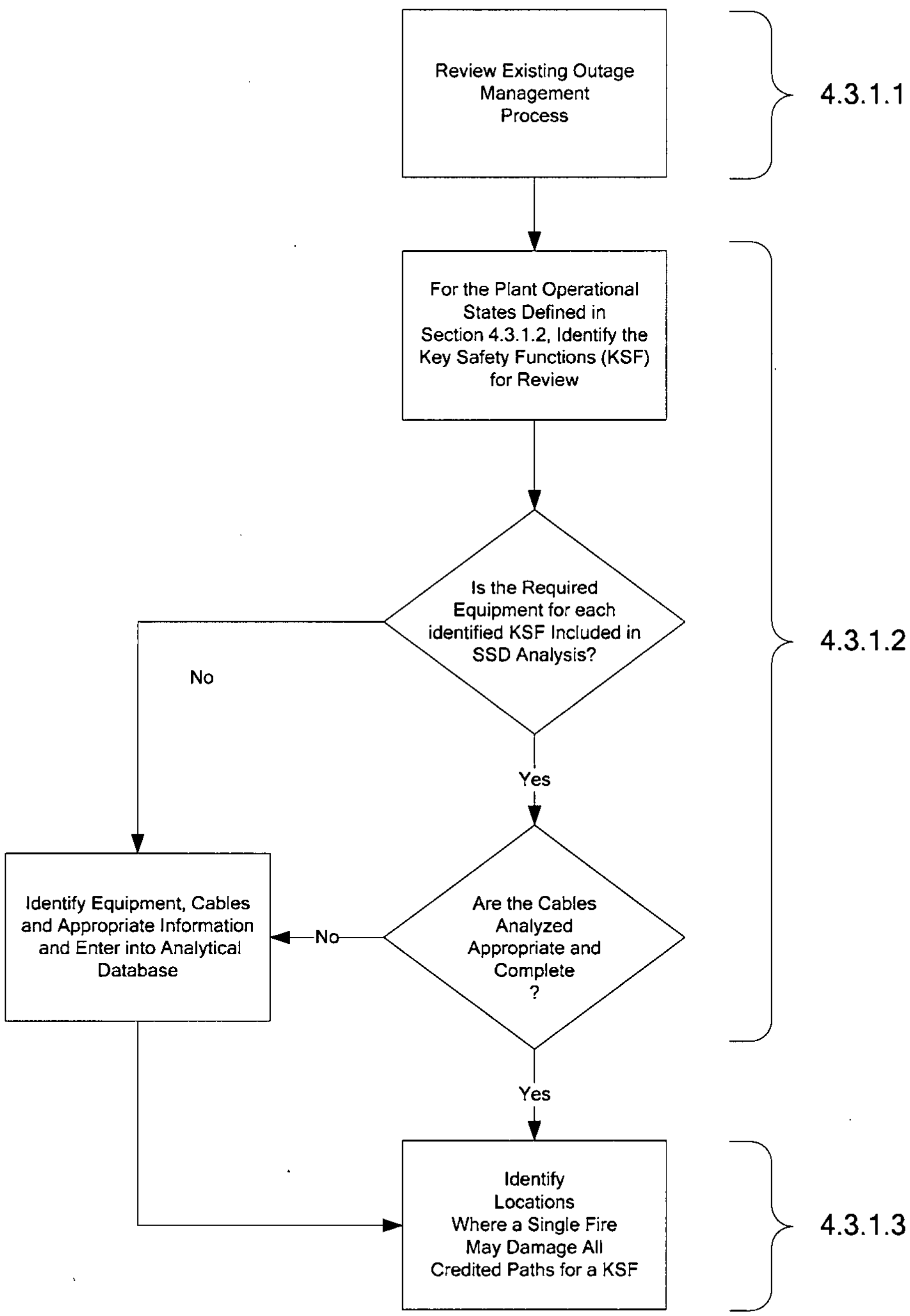

Figure 4-4 Review POSs, KSFs, Components, and Cables and Identify Pinch Points

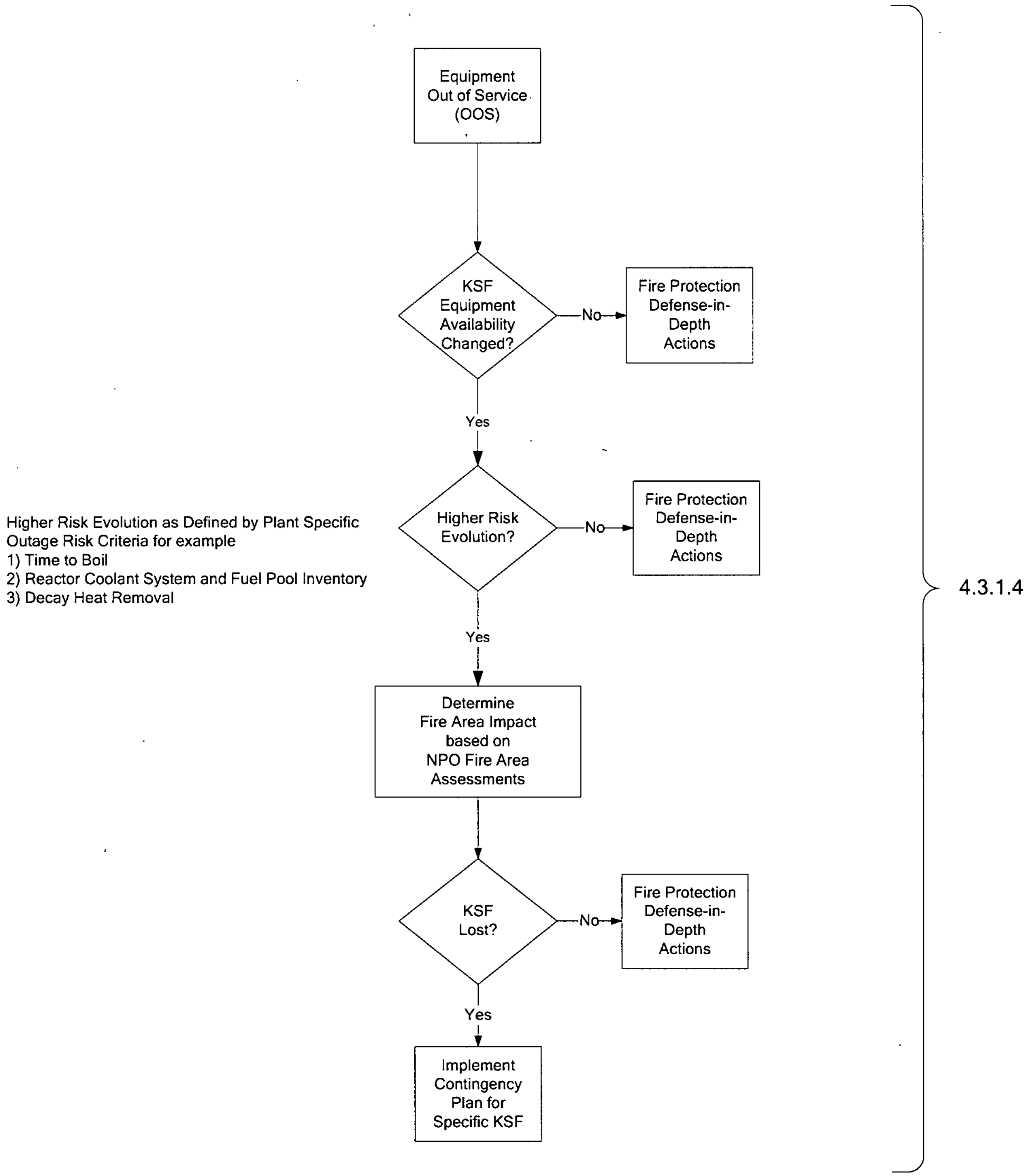

Figure 4-5 Manage Risks

4.3.2 Results of the Evaluation Process

The NPO fire area reviews conservatively assumed that the entire contents of a fire area would be lost. These reviews identified that there are fire areas where a single fire could result in a loss of all credited paths for a given KSF (i.e. pinch point). The review also identified that there are certain fire areas that are vulnerable to a loss of a KSF if certain system trains or components are taken out of service during a non-power operational mode and a fire were to occur. Locations where a fire might cause damage to equipment are identified in the HNP-E/ELEC-0002, NFPA 805 Transition – Non-Power Operational Modes Review.

Approximately 20 generic pinch points were identified during the performance of the NPO fire area reviews. In order to preclude or mitigate these KSF failures, a number of strategies were developed. These strategies include planned revisions to plant shutdown and abnormal operating procedures. These planned procedural revisions make changes to plant equipment and electrical system line-ups as the plant is brought to cold shutdown conditions, and were made to preserve the KSF. Plant operational procedures were also revised to include recovery actions for those instances where operator actions would be necessary to ensure that a specific KSF can be maintained. Specific procedures to be updated are identified in the HNP-E/ELEC-0002, NFPA 805 Transition – Non-Power Operational Modes Review.

To address concerns associated with equipment being taken out of service during NPO modes, and the potential for a concurrent fire, the HNP outage management procedure (OMP-003) is being revised to provide instructions that will assist in mitigating the effects of a fire if one were to occur. This procedure revision provides guidelines for actions to be taken in specific fire areas when components or system trains are taken out of service. For those fire areas where the credited KSF system or equipment has been taken out of service the following guidelines will be included in the outage management procedure:

- Prohibition or limitation of hot work,
- Prohibition or limitation of combustible materials, and/or
- Establishment of additional fire watches as appropriate.

Utilizing the above outlined approaches to alleviate the identified "pinch points" precluded the need to utilize fire modeling in order to resolve a KSF concern.

4.4 Radioactive Release Performance Criteria

4.4.1 Overview of Evaluation Process

The review of the HNP fire protection program against NFPA 805 requirements for fire suppression related radioactive release (NEI 04-02 Table G-1) was performed using the methodology contained in Progress Energy Project Instruction FPIP-0121, Radiological Release Reviews During Fire Fighting Operations. The methodology steps are outlined below.

Step 1 - Fire Pre-Plan review. The site fire pre-plans for locations that have the potential for radiological contamination were reviewed. The review was conducted by an expert panel to ensure specific steps are included for containment and monitoring of potentially contaminated materials.

Step 2 - Fire Brigade Training Plan review. The site fire brigade training materials were reviewed by an expert panel to ensure specific steps are included for dealing specifically with containment and monitoring of potentially contaminated materials and monitoring of potentially contaminated fire suppression products following a fire event.

Step 3 – Engineering Controls. During the expert panel review process, it was determined whether or not Engineering Controls could be established to minimize the release of radioactive materials (e.g. smoke and/or contaminated water).

Step 4 – Documentation – The result of the review were documented.

4.4.2 Results of the Evaluation Process

The HNP radiological release review determined if the current Fire Protection program is compliant with the requirements of NFPA 805, NEI 04-02 and Regulatory Guide 1.205. The review determined that radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) would be as low as reasonable achievable and would not exceed applicable 10 CFR, Part 20 limits. The site specific review of associated fire event and fire suppression related radioactive release is summarized in Attachment E of this document, the NEI 04-02 Table G-1. During implementation a Fire Pre-plan will be developed for outside yard areas to address Radioactive Materials Areas (RMAs) and sea-land type container storage as documented in the G-1 Table open items list.

The generic fire pre-plan outline identifies typical fixed radiological hazards for each area. All HNP fire pre-plans were screened for applicability. Fire pre-plans that address areas where there is no possibility of radiological hazards were screened out from further review. The full list of Fire pre-plans is provided below.

Table 4-2 HNP Pre-Plans

Fire Pre-Plan	Status
FPP-12-01-CNMT Containment	Reviewed
FPP-12-02-RAB190-216	Reviewed
FPP-12-02-RAB236	Reviewed
FPP-12-02-RAB261	Reviewed
FPP-12-02-RAB286	Reviewed

Table 4-2 HNP Pre-Plans

Fire Pre-Plan	Status
FPP-12-02-RAB305324	Reviewed
FPP-12-03-FHB	Reviewed
FPP-12-04-DGB	Screen Out
FPP-12-05-DFOS	Screen Out
FPP-12-06-WPB	Reviewed
FPP-12-07-TB	Reviewed
FPP-12-08-SEC	Screen Out
FPP-12-09-LAF	Screen Out

HNP has transitioned its fire brigade and Site Incident Commander lesson plans into NFPA 600 compliant lessons. These lesson plans were under development at the time this review was conducted. Attributes included within the new NFPA 600 lesson plans to address the Radioactive Release objectives ensure the materials deal specifically with the containment and monitoring of potentially contaminated fire suppression agents and products of combustion have been incorporated.

4.5 Change Evaluations

4.5.1 Fire PRA Development and Acceptance

A Fire PRA model was developed for HNP using the guidance provided in NUREG/CR-6850. The resulting model was reviewed against the requirements of ANSI/ANS-58.23-2007, American National Standard - Fire PRA Methodology, as supplemented by additional elements in a developing 'Combined Standard' (ASME RA-S-2002, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (and 2007 addenda ASME RA-Sc-2007, Appendix A)). As noted in Regulatory Guide 1.205, the review of the Fire PRA was conducted by the NRC, such that a separate peer review would not be required. Additionally, HNP elected to obtain a formal industry review of the Fire PRA model.

4.5.1.1 Internal Events PRA

The HNP base internal events PRA was the starting point for the final fire PRA. The internal events PRA was modified to capture the effects of fire both as an initiator of an event and the subsequent potential failure modes for affected circuits or individual targets.

The HNP internal events PRA had a peer review performed in June 2002 in accordance with guidance in NEI-00-02, Industry PRA Peer Review Process. All of the Facts and Observations (F & Os) were resolved. In 2006 a gap assessment of the HNP Internal Events PRA was performed to determine the scope of work required to ensure the HNP Internal Events PRA meets Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 1. After completion of the required work, a focused peer review was performed as required by the ASME RA-S-2002, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (and 2007 addenda ASME RA-Sc-2007, Appendix A). The results of this focused peer review were 11 findings. All but 2 of the findings have been completely resolved, both of which have been adequately resolved for use in the Fire PRA application.

4.5.1.2 Fire PRA – NRC Staff Review and Industry Peer Review

In February 2008, as part of the NRC fire PRA staff review, the NRC staff reviewed the resolution of F & Os from the 2002 NEI-00-02 peer review, the 2006 gap assessment review and the subsequent 2007 Focused Peer Review. The NRC staff had no findings from that review. At the same time the NRC staff also reviewed the fire PRA in order to provide feedback on the progress to date of the HNP fire PRA development. The NRC fire PRA staff review, which was required by Regulatory Guide 1.205, noted that there were a number of incomplete high level and supporting requirements.

Following the NRC staff review, additional work was completed on the fire PRA. In April 2008, a team of industry experts in risk and fire PRA focused on those areas the NRC previously identified as not completed, had findings or were assigned a Category I quality level. This limited scope peer review team included two individuals who were authors of the fire PRA guidance document NUREG/CR-6850. The industry review also included three NRC observers. The results of the limited peer review identified that most of the NRC identified findings were resolved and a few new findings were identified. The conclusion of the industry peer review report was that "the Harris Nuclear Plant Fire PRA substantially meets the ANSI/ANS fire PRA standard at capability category II or better. The HNP fire PRA meets capability category II or better for 87% of the applicable supporting requirements. The outstanding issues primarily pertain to completion of the final quantification and completing the documentation."

The NRC staff review and the follow-up industry peer review compared the fire PRA against requirements of ASME/ANS-RA-S-2007, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Application, Combined PRA Standard (Draft). Other attributes that both the NRC and industry peers reviewed include the PRA maintenance and update requirements to ensure the PRA will continue to reflect the HNP plant as designed and operated.

In addition to the above reviews, as part of the pilot process the Progress Energy PRA staff made a number of presentations to the NRC staff on the treatment of various technical elements as well as provided copies of preliminary results. This included items such as the determination of fire frequency, the method of electrical cabinet counting and plant partitioning. Another area discussed with the staff and subsequently incorporated in the final PRA are the Human Reliability Analysis (HRA) screening method.

The NRC issued results of the NRC Fire PRA Staff Review on March 10, 2008 (ML080650403). The identification and resolution of the high level findings from the NRC Fire PRA Staff Review and Fire PRA industry peer review process are summarized below per NRC guidance (Regulatory Guide 1.205, Revision 0, Regulatory Position C.4.3 and NEI 04-02, Rev. 1, Section 5.1.3).

Table 4-3 HNP Fire PRA Staff Review –Findings

SR	Topic	Status
PP-B2-1	Justification for non-rated partition boundaries insufficient	Closed, per industry peer review. Replaced by new finding PP-B2-01
PP-B2-2	Insufficient justification/documentation for "rooms" within Fire Compartments for hot gas layer.	Closed, per industry peer review
PP-C3-1	Adequate justification required for non-rated barriers and the use of rooms as partitioning physical analysis units.	Closed, per industry peer review. Replaced by new finding PP-B2-01
ES-A6-1	LERF not considered	Closed, per industry peer review
ES-B4-1	Containment bypass other than ISLOCA has not been addressed.	Closed, per industry peer review
ES-C1-1	Redundant and diverse indication for operator actions	Additional documentation and circuit routing was performed but the documentation was inconsistent. No effect on HRA analysis results.
ES-D1-1	Difficulty linking cable to components to scenarios	Closed, per industry peer review
CS-A3-1	Power supply cable (MCC) not modeled in PRA	Closed, per industry peer review
CS-A4-1	Same issue as above	Closed, per industry peer review
CS-A7-1	Inter-cable hot shorts for CTMT Bypass not documented	Closed, per industry peer review
CS-A8-1	Three phase shorts for CTMT Bypass not documented	Closed, per industry peer review

Table 4-3 HNP Fire PRA Staff Review –Findings

SR	Topic	Status
CS-C4-1	Configuration control for electrical coordination study	Closed, per industry peer review. Replaced by new finding MU-A1-01
PRM-A1-1	No conditional fire LERF model developed	Closed, per industry peer review
PRM-A2-1	CLERP not quantified for fire scenarios	Closed, per industry peer review
PRM-B1-1	New minor LERF contributors not investigated	Closed, per industry peer review
FSS-A2-1	Lower damage threshold for equipment not directly in fire zone of influence not considered	Closed, per industry peer review
FSS-B1-1	Control Room abandonment for loss or degraded functions not documented	Closed, per industry peer review
FSS-C5-1	Damage criteria for solid state components not considered	Closed, per industry peer review
FSS-D1-1	ZOI and hot gas layer calculations appear very conservative and no detailed fire modeling done	Closed, per industry peer review
FSS-E3-1	No Uncertainty analysis performed	Complete to CAT 1 as listed on Table 4-5 below
FSS-F-1	Structural steel not evaluated	Complete to at least CAT 1. CAT 1 limitations are listed on Table 4-5 below
FSS-G-1	Multi-compartment analysis	Closed, per industry peer review
FSS-H8-1	Document methodology for multi-compartment analysis	Closed, per industry peer review
FSS-H9-1	Uncertainties	Closed, per industry peer review
IGN-A5-1	Use of reactor year/critical year	Complete. Frequencies are adjusted for the fraction of time the plant is at-power.
CF-B1-1	Circuit failure analysis documentation not in calculation	Closed, per industry peer review
HRA-B2-1	Negative effect of Fire HFE not in model	Closed, per industry peer review
HRA-B3-1	Fire induced operator actions due to instrumentation failure evaluation not complete	Closed, per industry peer review
HRA-C1-3	Documentation of significant Human Error Probability (HEPs) determination	Due to the fire quantification methodology and different component failure probabilities due to how the various component circuits react to a fire, the compiling of a single file to search for the significant HEPs using the normal PRA methods is not possible until new software tools are developed. This has no effect on the HEP results.

Table 4-3 HNP Fire PRA Staff Review –Findings

SR	Topic	Status
HRA-C1-4	Instrumentation for HEPs used and conflict resolution	Closed, per industry peer review
HRA-C1-5	Additional T-H analysis for new operator actions	Closed, per industry peer review
HRA-C1-6	Revision of HFEs due to fire and consistency	The comparison and check for consistency was performed, but needs improved documented in calculation. This has no effect on the HFE results
SF-1	Seismic and fire interaction	Closed, per industry peer review
FQ-A4-1	Fire sequence identification	Closed, per industry peer review
FQ-D1-1	LERF not quantified	Closed, per industry peer review
FQ-E1-1	LERF not quantified	Closed, per industry peer review
FQ-E1-2	Determination of significant sequences	Important sequences are identified by ignition source.
FQ-F1-1	Documentation of CDF and LERF analysis	Items m and b are complete. Items e, f, g, and l, all require additional documentation. No effect on analysis CDF/LERF results.
FQ-F1-2	Significant contributors to fire CDF	HNP Fire PSA Quantification calculation has been issued which identifies significant sources of fire CDF. No further work for this item.
FQ-F1-3	Assumptions and uncertainty	Statistical uncertainty was not performed. No effect on results.
FQ-F2-1	Assess PRA to standard for reference internal events requirements	9 of 11 internal Peer review findings complete. The remaining two have no effect on fire delta CDFs or delta LERF.
UNC-1	Uncertainty and assumptions	Closed, per industry peer review
MUD-B4-1	Procedure lacks reference to PRA combined standard (draft)	Completed, procedure revision has been issued.

HNP decided to perform an additional industry peer review, which started on April 21, 2008. The additional industry peer review resulted in six findings. Those not resolved are listed in Table 4-4:

Table 4-4 HNP Fire PRA Industry Peer Review – Findings

SR	Topic	Status
PP-B2-01	Plant partitioning using non-rated fair barriers needed additional information, (sealing material, etc) and need to be consistently described. Use judgment of smoke and fire migration in the Multi-compartment analysis.	Determined to be documentation only, no effect on CDF or plant partitioning. Determined no effect on results required to support the LAR.

ES-C1-01	Attachment 2 of calculation HNP-F/PSA-0077 conflicts with cable routing data base, in that instrumentation cables were actually routed and failures added to fault tree.	Documentation to be revised. Determined no effect on results required to support the LAR.
FSS-F3-01	Turbine building fire needs discussion of accident scenario.	Completed. Discussion added and calculation issued.
CF-A1-01	The intra and inter cable failure probabilities are not added	The difference of not adding these failure probabilities are within the uncertainty values provided in the industry guidance. In addition there is other evidence that the failure rates are over conservative, hence no impact on CDF and adjustments will be made after industry guidance is clarified.
SF-A1-01	Seismic analysis changes since IPEEE need to be addressed	Analysis was updated and issued.
FQ-A4-02	Quantification does not meet Category II for all internal events standard SRs.	See Category I discussions in section 4.5
FQ-D1-01	Need to perform uncertainty analysis on LERF	No effect on CDF/LERF results, Related to UNC-A1-01
FQ-F1-01	Final CDF and LERF not documented	Completed, the calculation has been issued.
UNC-A1-01	Statistical uncertainty not completed	No effect on CDF/LERF results,
MU-A1-01	Add the breaker coordination calculation as a reference to PRA maintenance procedure.	Completed, procedure revision has been issued.
MU-B4-01	The ASME/ANS standard has been issued and procedure needs to reference the issued standard rather than draft.	Completed, procedure revision has been issued.

A limited number of AMSE/ANS areas were identified by both the NRC and the industry peer review team as meeting Category I only requirements. These are listed below with the planned disposition.

Table 4-5 HNP Fire PRA Category I Summary

SR	Topic	Status
FSS-D7	Verification that the systems have not experienced outlier behavior.	Cat I is acceptable. A check of system unavailability indicates that HNP it is not an outlier.
FSS-D9	Smoke damage to equipment not considered	Cat I is acceptable.
FSS-E3	Use of uncertainty parameters for generic fire frequencies.	Cat I is acceptable. HNP is using the numbers provided by the NRC in NUREG/CR-6850.

Table 4-5 HNP Fire PRA Category I Summary

SR	Topic	Status
FSS-F2	Exposed structural steel column collapse	Cat I is acceptable. The qualitative failure criteria are sufficient to evaluate risk.
FSS-F3	Risk assessment from exposed structural steel	Cat I is acceptable. The qualitative assessment is sufficient to evaluate risk.
FSS-H5	Documentation of fire modeling results – parameter uncertainty distributions	Cat I is acceptable. The parameter uncertainty distributions would not impact the values used for making decisions
FSS-H6	Documentation of fire modeling results – statistical models	Cat I is acceptable. Plant specific updates of fire parameters were not done (FSS-F2), so no need to describe statistical basis
IGN-A4-1	Bayesian plant specific update from the data in NUREG/CR-6850.	Cat I is acceptable.

4.5.2 NFPA 805 Risk-Informed, Performance-Based Change Evaluation Process

The risk-informed performance based change evaluations are based upon the HNP Fire PRA, Revision 0. This initial version of the HNP PRA has some follow-up items associated with verification of selected data inputs, changes in inputs not quantified and unincorporated methodology enhancements. Based upon a qualitative impact review of these items, it is expected that the baseline fire CDF in this submittal is representative of CDF when these items are closed, as well as the importance measures and the delta CDFs provided.

Risk-informed, performance-based change evaluations were performed as part of the HNP NFPA 805 transition. Change evaluations were developed using Progress Energy Project Instruction FPIP-0128, NFPA 805 Change Evaluations. This instruction is based upon the industry guidance in primary documents NFPA 805, NEI 04-02 Revision 1, and Regulatory Guide 1.205, Revision 0.

Table 4-6 – Change Evaluation – Guidance Summary Table

Document	Section(s)	Topic
NFPA 805	2.2(h), 2.2.9, 2.4.4, 4.2.4, A.2.2(h), A.2.4.4, D.5	Change Evaluation Risk of Recovery Actions (4.2.4)
NEI 04-02 Revision 1	4.4, 5.3, Appendix B, Appendix I, Appendix J	Change Evaluation, Change Evaluation Forms (App. I)
Reg. Guide 1.205 (May 2006)	C.2.2, C.2.3, C.3.2	LAR reporting requirements (C.2.2) Risk of operator manual actions (C.2.3) Change Evaluations (C.3.2) Circuit Analysis (C.3.3) PSA Peer Review (C.4.3)

The Plant Change Process consists of the following subtasks:

- Defining the Change
- Preliminary Risk Review
- Risk Evaluation
- Acceptability Determination

4.5.2.1 Defining the Change

The Change Evaluation process started with definition of the change or altered condition to be examined (i.e., variance from the deterministic requirements) and the baseline configuration as defined by the Licensing Basis (i.e., current pre-transition licensing basis).

4.5.2.2 Preliminary Risk Review

The NFPA 805 change process has a step to perform this review to evaluate minor program changes. This step was not utilized for the transition process as the variances from the deterministic requirements were deemed not to be minor.

4.5.2.3 Risk Evaluation

For changes that were not determined to be minor, the changes were assessed using risk-informed, performance-based techniques (including, but not limited to fire modeling and PRA). The risk evaluations, depending upon the nature of the change, were performed as either limiting or bounding fire modeling/fire risk analyses or detailed integrated analyses.

4.5.2.4 Acceptability Determination

The risk evaluation of plant changes, as appropriate, were measured quantitatively for acceptability using the change in Core Damage Frequency (ΔCDF) and the change in Large Early Release Frequency (ΔLERF) criteria from Section 5.3.5 of NEI 04-02 and Regulatory Guide 1.205. The ΔLERF is estimated using the ratio of Fire LERF to Fire CDF. The results of the acceptability determination were documented in calculations. An evaluation to ensure maintenance of defense-in-depth and safety margins was also performed.

4.5.3 NFPA 805 Risk-Informed, Performance-Based Change Evaluation Results

HNP's pre-transition post-fire safe shutdown analysis revalidation efforts and the NFPA 805 transition project activities have identified a number of variances from the pre-transition fire protection licensing basis. A number of plant and programmatic changes address these variances and reduce risk.

In addition, the changes associated with transition to NFPA 805 have been assessed for impact on fire protection defense-in-depth and safety margin. Defense-in-depth and safety margins are maintained.

Following completion of transition activities and planned modifications and program changes, the plant will be compliant with 10 CFR 50.48(c).

4.5.4 Risk Change Due to NFPA 805 Transition

In accordance with the guidance in Regulatory Position C.2.2 of Regulatory Guide 1.205, Revision 0, the total risk change associated with pre-transition fire protection program variances that will meet the NFPA 805 performance-based approach (via the change evaluation process) were evaluated. Upon completion of plant modifications (see section 4.8.3) the total change in risk associated with the transition to NFPA 805 will be consistent with the acceptance guidelines in Regulatory Guide 1.174. Progress Energy has committed to be in compliance with 10 CFR 50.48(c) by the end of Refueling Outage 16, currently scheduled for November 5, 2010.

4.6 Monitoring Program

The Monitoring Program will be implemented after the LAR submittal as part of the fire protection program transition to NFPA 805. In order to assess the impact of the transition to NFPA 805 on the current monitoring program, the HNP fire protection program documentation hierarchy, maintenance program process/procedures and plant change processes will be reviewed. Sections 4.5.3 and 5.2 of the NEI 04-02 Implementing Guidance will be used during the review process, and that process is described below.

4.6.1 Overview of NFPA 805 Requirements and NEI 04-02 Guidance on the Existing Monitoring Program

Section 2.6 of NFPA 805 states:

> *"A monitoring program shall be established to ensure that the availability and reliability of the fire protection systems and features are maintained and to assess the performance of the fire protection program in meeting the performance criteria. Monitoring shall ensure that the assumptions in the engineering analysis remain valid."*

The intent of the monitoring transition effort will be to confirm (or modify as necessary) the adequacy of the existing surveillance, testing, maintenance, compensatory measures, and oversight processes for transition to NFPA 805. This review will consider the following:

1. The adequacy of the scope of systems and equipment within existing plant programs (i.e., are the necessary fire protection program systems and features included).

2. The performance criteria for the availability and reliability of fire protection systems and features relied on to demonstrate compliance.

3. The adequacy of the plant corrective action program in determining causes of equipment and programmatic failures and in minimizing their recurrence.

A Project Instruction was developed to outline the review methodology process that will be used. The process is based on NUMARC 93-01, Industry Guideline For Monitoring the Effectiveness of Maintenance at Nuclear Power Plants. The Project Instruction will be converted to a procedure to support program implementation and the review methodology is described below. This process and the proposed plan for monitoring implementation were discussed at the April 2008 Pilot Observation Meeting.

4.6.2 Overview of Post-Transition NFPA 805 Monitoring Program

The following process will be established for monitoring post-transition:

A NFPA 805 Monitoring Expert Panel will be used to:

- Determine the scope of SSCs and programmatic elements to monitor.
- Establish levels of availability, reliability, or other criteria for those elements that require monitoring.

A flowchart of the overall process for NFPA 805 monitoring implementation is shown in Figure 4-6. The four main phases of monitoring process are described below:

- Phase 1 - Scoping
- Phase 2 - Establishing Risk Criteria
- Phase 3 - Risk Determination
- Phase 4 - Monitoring Implementation

Phase 1 - Scoping

Phase 1 of the process will determine the scope of the NFPA 805 monitoring program. In order to meet the NFPA 805 requirements for monitoring, three basic categories are established:

- Monitoring of Fire Protection Program Structures, Systems, and Components
- Monitoring of Fire Protection Programmatic Elements
- Monitoring of Key Assumptions in Engineering Analyses

Phase 2 - Establishing Risk Criteria

Phase 2 of the process will establish risk significant criteria for SSCs and programmatic elements within the NFPA 805 monitoring scope. The Fire PRA is the primary tool used to establish risk significant criteria. Only certain SSCs/fire protection program elements are amenable to risk measurement in Fire PRA.

Another aspect of risk criteria is establishing performance criteria. These performance criteria will be established for items within the NFPA 805 monitoring scope, regardless of their ability to be measured using risk significant criteria. The performance criteria used should be availability, reliability, or condition monitoring, as appropriate.

Phase 3 - Risk Determination

Phase 3 will consist of utilizing the Fire PRA, or other processes, as appropriate, to determine the risk significant SSCs/fire protection program elements using the criteria established in Phase 2.

Phase 4 – Monitoring Implementation

Phase 4 is the full implementation of the monitoring program, once the scope and criteria are established in previous phases. The implementation includes the assessment of performance against the established criteria. Follow on steps include refinement of performance goals and criteria, analysis of situations where goals are not met, and addressing items appropriately via the corrective action program.

The fire protection program monitoring scope is not included in the scope of Maintenance Rule. For convenience, the Maintenance Rule monitoring process will be used to facilitate use of existing programs and processes (e.g., use of tools such as tracking databases).

As part of the monitoring program a fire protection assessment (audit) shall be performed at least once per 36 months using an outside (external to Progress Energy) qualified fire protection engineer meeting Society of Fire Protection Engineers (SFPE) member grade qualifications. This assessment is currently performed by the Progress Energy QA organization.

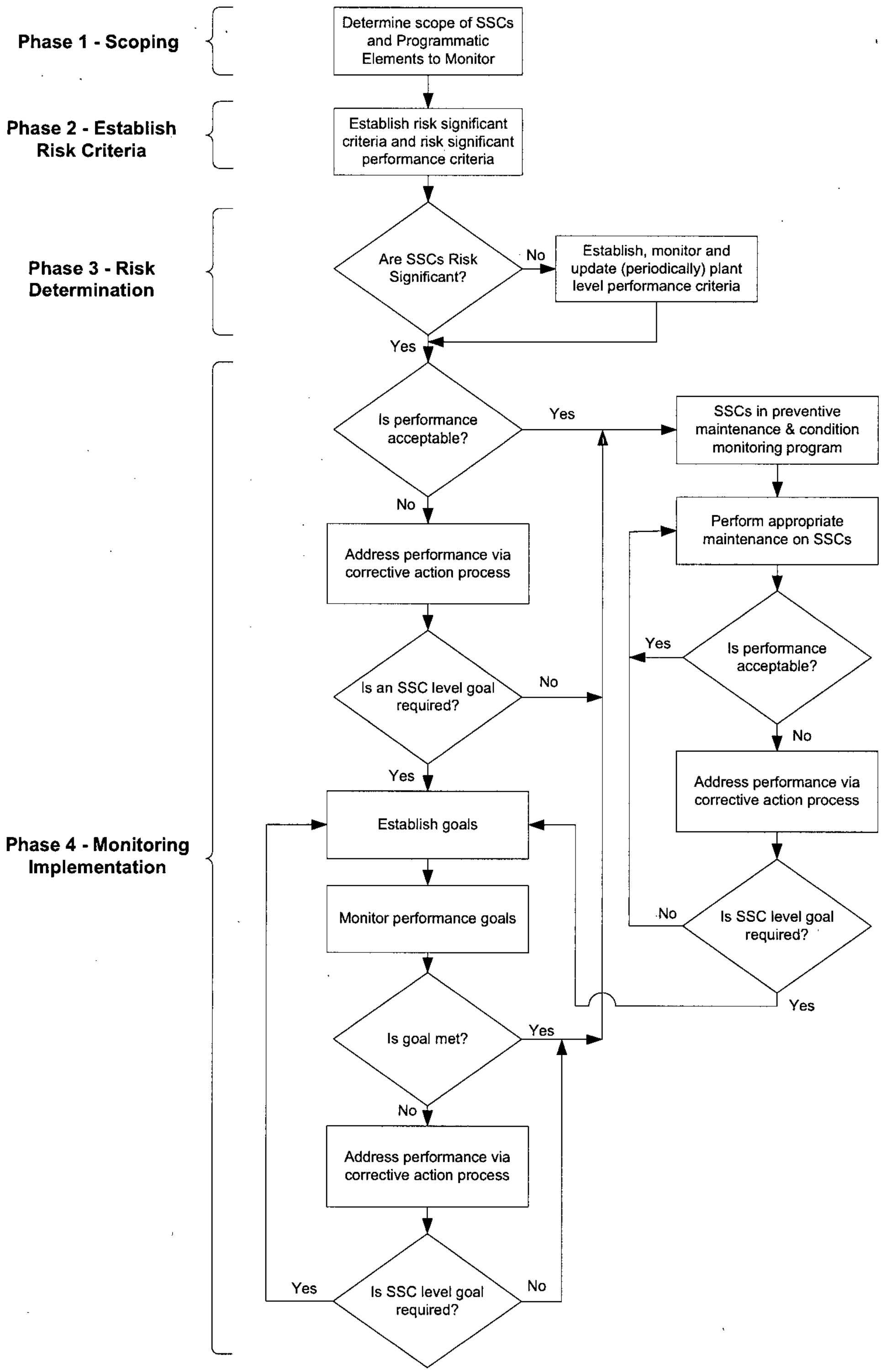

Figure 4-6 – Post-Transition NFPA 805 Monitoring Program

4.7 Program Documentation, Configuration Control, and Quality Assurance

4.7.1 Compliance with Documentation Requirements in Section 2.7.1 of NFPA 805

In accordance with the requirements and guidance in NFPA 805 Section 2.7.1 and NEI 04-02, Revision 1, HNP has documented analyses to support compliance with 10 CFR 50.48(c). The analyses and calculations have been performed in accordance with Progress Energy's processes for ensuring assumptions are clearly defined, that results be easily understood, that results be clearly and consistently described, and that sufficient detail be provided to allow future review of the entire analyses.

Documentation associated with 10 CFR 50.48(c) will be maintained for the life of the plant and organized to facilitate review for accuracy and adequacy.

The HNP "Fire Protection Program Design Basis Document" concept described in Section 2.7.1.2 of NFPA 805 and necessary supporting documentation described in Section 2.7.1.3 of NFPA 805 have been developed as part of transition to 10 CFR 50.48(c) and will be issued for use as part of program implementation following receipt of the license amendment. Appropriate cross references have been established to supporting documents as required by Progress Energy processes.

A diagram depicting the planned post-transition documentation relationships is shown below in Figure 4-7.

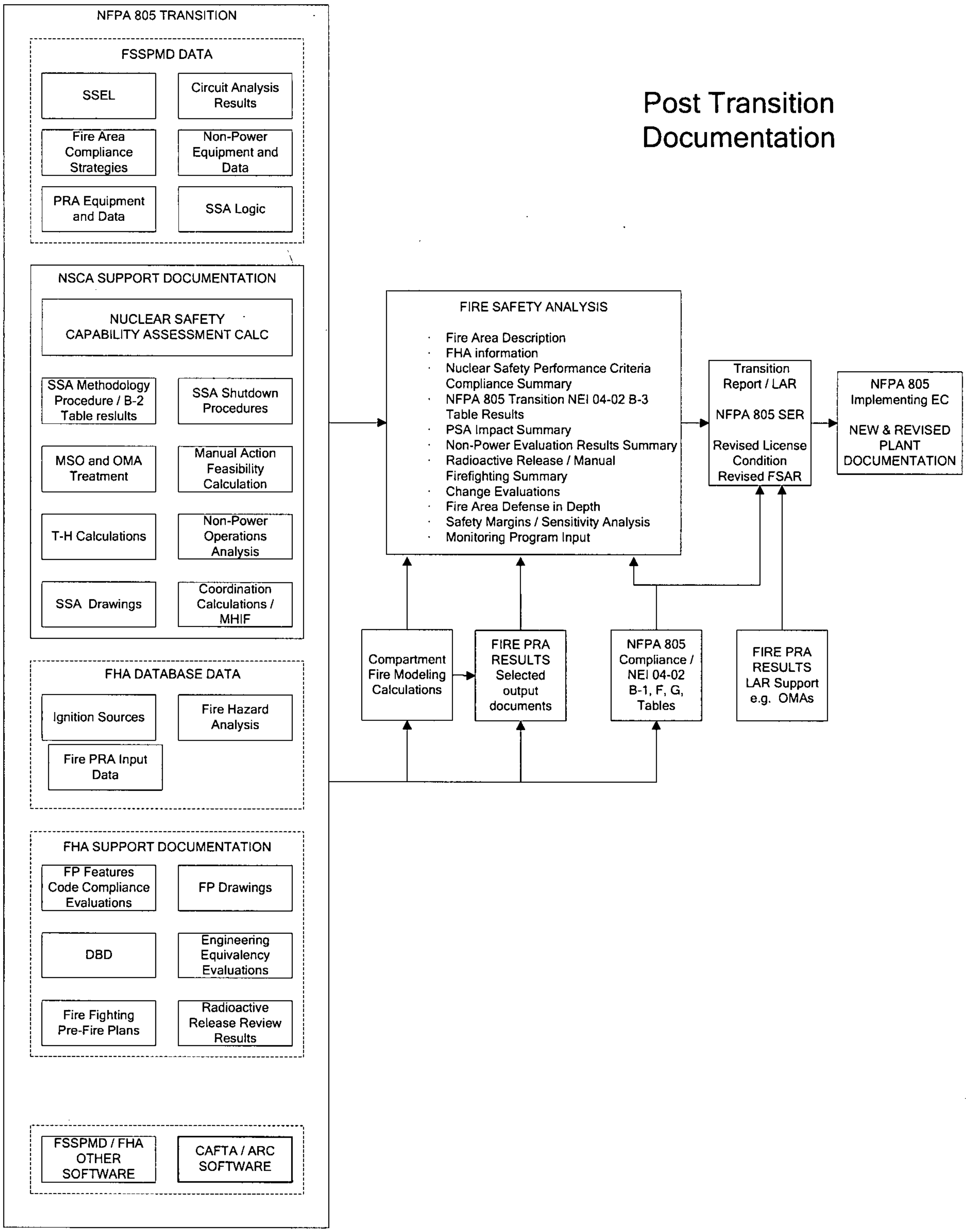

Figure 4-7 NFPA 805 Transition – Planned Post-Transition Documentation Relationships

4.7.2 Compliance with Configuration Control Requirements in Section 2.7.2 of NFPA 805

Program documentation established, revised, or utilized in support of compliance with 10 CFR 50.48(c) is subject to Progress Energy configuration control processes that meet the requirements of Section 2.7.2 of NFPA 805. This includes the appropriate procedures and processes for ensuring that changes impacting the fire protection program are reviewed for impact.

4.7.3 Compliance with Quality Requirements in Section 2.7.3 of NFPA 805

During the transition to 10 CFR 50.48(c), HNP performed work in accordance with the quality requirements of Section 2.7.3 of NFPA 805. Post-transition quality requirements from NFPA 805 that are not currently part of the Progress Energy processes will be revised to include any additional requirements.

NFPA 805 Section 2.7.3.1 - Review – Analyses, calculations, and evaluations performed in support of compliance with 10 CFR 50.48(c) have been and will be performed in accordance with Progress Energy procedures that require independent review.

NFPA 805 Section 2.7.3.2 – Verification and Validation – Calculational models and numerical methods used in support of compliance with 10 CFR 50.48(c) have been and will be verified and validated as required by Section 2.7.3.2 of NFPA 805.

NFPA 805 Section 2.7.3.3 – Limitations of Use - Engineering methods and numerical models used in support of compliance with 10 CFR 50.48(c) were used and will be used appropriately as required by Section 2.7.3.3 of NFPA 805.

NFPA 805 Section 2.7.3.4 – Qualification of Users – Cognizant personnel who used and applied engineering analysis and numerical methods that support compliance with 10 CFR 50.48(c) were competent and experienced as required by Section 2.7.3.4 of NFPA 805. This requirement was met and will continue to be met by adherence to Progress Energy procedures and project management of contractor support staff.

NFPA 805 Section 2.7.3.5 – Uncertainty Analysis – Uncertainty analyses have been and will be performed as required by 2.7.3.5 of NFPA 805. This is of particular interest in the performance of fire modeling and Fire PRA development. These analyses have addressed uncertainty.

4.8 Summary of Engineering Analysis Results

4.8.1 Results of the Fire Area-by-Fire Area Review

Attachment C contains the results of the Fire Area Transition review (NEI 04-02 Table B-3). The Table B-3 includes the following summary level information for each fire area:

- Regulatory Basis – Both pre and post-transition regulatory bases are included.
- Performance Goal Summary – An overview of the method of accomplishment of each of the performance criteria in NFPA 805 Section 1.5 is provided.
- Reference Documents – Specific References to Operating Procedures and the Nuclear Safety Capability Assessment.
- Licensing Actions – Specific References to Deviation Requests that will remain part of the post-transition licensing basis and the Basis for Acceptability of that Licensing Action.
- Engineering Equivalency Evaluations – Specific References to Engineering Equivalency Evaluations that rely on determinations of Adequate for the Hazard that will remain part of the post-transition licensing basis and the Basis for Acceptability of that Engineering Equivalency Evaluation.

A higher level summary is provided below in Table 4-7. The table provides the following information:

- Fire Area: Fire Area Identifier
- Area Description: Fire Area Description
- FSA Reference: Reference to the HNP Fire Safety Analysis for the Fire Area
- NFPA 805 Regulatory Basis (Post-Transition): Post-transition NFPA 805 Chapter 4 reference for the Fire Area
- Change Evaluations: Reference to Change Evaluation (HNP Fire Safety Analysis Calculation), Documentation (Yes/No) of a change evaluation for the fire area
- Licensing Actions or EEEEs Transitioning? (Yes/No) Variances from NFPA 805 Chapter 3
- Suppression Required? (Yes/No): Confirmation of requirement for fire suppression in the Fire Area based on NFPA 805
  - Systems required to meet Chapter 4 deterministic compliance – summarized in Table 4-7
  - Systems required to meet Chapter 4 performance-based compliance (including systems credited for defense-in-depth) – summarized in Change Evaluations
  - High Safety Significant Fire SSCs as determined by the fire protection monitoring program – to be determined during Transition Implementation
- Detection Required? (Yes/No): Confirmation of requirement for fire detection in the Fire Area on NFPA 805. This includes:
  - Systems required to meet Chapter 4 deterministic compliance – summarized in Table 4-7

- Systems required to meet Chapter 4 performance-based compliance (including systems credited for defense-in-depth) – summarized in Change Evaluations
- High Safety Significant Fire SSCs as determined by the fire protection monitoring program – to be determined during Transition Implementation

Table 4-7 Fire Area Compliance Summary

Fire Area	Area Description	FSA Reference (HNP-M/MECH-)	NFPA 805 4.2 NSCA Reg. Basis (Post-Transition)	Change Evaluations (Yes/No)	Licensing Actions or EEEEs Transitioning? (Yes/No)	Suppression Req'd? (Yes/No)	Detection Req'd? (Yes/No)
12-A-BAL	Reactor Auxiliary Building Units 1 And 2 Balance	1106	4.2.3.2, 4.2.3.2	Yes	Yes	No	Yes
12-A-CR	Control Room, Reactor Auxiliary Building	1127	4.2.4	Yes	Yes	Yes	Yes
12-A-CRC1	Control Room Complex	1126	4.2.4	Yes	Yes	No	Yes
12-A-HVIR	Heating, Ventilating, and Instrument Repairs, Reactor Auxiliary Building	1125	4.2.4	No	Yes	No	No
12-I-ESWPA	Emergency Service Water Pump 1A	1173	4.2.3.2	No	Yes	No	No
12-I-ESWPB	Emergency Service Water Pump 1B	1174	4.2.3.2	No	Yes	No	No
12-O-TA	Diesel Fuel Oil Storage Tank A	1175	4.2.3.2	No	Yes	No	No
12-O-TB	Diesel Fuel Oil Storage Tank B	1176	4.2.3.2	No	Yes	No	No
1-A-ACP	Auxiliary Control Panel Room	1124	4.2.3.2, 4.2.3.3.a, 4.2.4	Yes	Yes	No	No
1-A-BAL-A	Reactor Auxiliary Building Unit 1 – Lower Elevations	1116	4.2.3.2, 4.2.3.3.a, 4.2.3.3.b, 4.2.3.3.c, 4.2.4	Yes	Yes	Yes	Yes
1-A-BAL-B	Reactor Auxiliary Building Unit 1 – Upper Elevations	1117	4.2.3.2, 4.2.3.3.b, 4.2.3.3.c, 4.2.4	Yes	Yes	Yes	Yes
1-A-BAL-C	Reactor Auxiliary Building Unit 1 – Analysis Area C	1105	4.2.3.2, 4.2.3.3.a	Yes	Yes	No	Yes
1-A-BAL-D	Reactor Auxiliary Building Unit 1, Analysis Area D	1118	4.2.3.2	No	No	No	No
1-A-BAL-E	Reactor Auxiliary Building Unit 1 – Analysis Area E	1107	4.2.3.2, 4.2.3.3.a, 4.2.4	Yes	Yes	No	No
1-A-BAL-F	Reactor Auxiliary Building Unit 1 – Analysis Area F	1108	4.2.3.2	No	No	No	No
1-A-BAL-G	Reactor Auxiliary Building Unit 1 – Analysis Area G	1177	4.2.3.2	No	No	No	No
1-A-BAL-H	Reactor Auxiliary Building Unit 1 – Analysis Area H	1178	4.2.3.2	No	No	No	No
1-A-BAL-J	Reactor Auxiliary Building Unit 1 – Analysis Area J	1109	4.2.4	Yes	Yes	No	No

Table 4-7 Fire Area Compliance Summary

Fire Area	Area Description	FSA Reference (HNP-M/MECH-)	NFPA 805 4.2 NSCA Reg. Basis (Post-Transition)	Change Evaluations (Yes/No)	Licensing Actions or EEEEs Transitioning? (Yes/No)	Suppression Req'd? (Yes/No)	Detection Req'd? (Yes/No)
1-A-BAL-K	Reactor Auxiliary Building Unit 1 – Analysis Area K	1179	4.2.3.2, 4.2.4	Yes	Yes	No	No
1-A-BATA	Battery Room A	1110	4.2.3.2	No	No	No	No
1-A-BATB	Battery Room B	1111	4.2.3.2	No	No	No	No
1-A-CSRA	Cable Spreading Room A	1119	4.2.3.2, 4.2.3.3.c, 4.2.4	Yes	Yes	Yes	Yes
1-A-CSRB	Cable Spreading Room B	1120	4.2.4	Yes	Yes	Yes	Yes
1-A-EPA	Electrical Penetration Area A	1112	4.2.3.2, 4.2.3.3.c, 4.2.4	Yes	No	Yes	Yes
1-A-EPB	Electrical Penetration Area B'	1113	4.2.3.2, 4.2.4	Yes	Yes	Yes	Yes
1-A-SWGRA	Switchgear Room A	1122	4.2.3.2, 4.2.3.3.a, 4.2.4	Yes	Yes	No	Yes
1-A-SWGRB	Switchgear Room B	1123	4.2.4	Yes	Yes	No	Yes
1-C	Containment Building, All Levels	1121	4.2.3.2	No	Yes	Yes	Yes
1-D-DGA	Diesel Generator 1A	1180	4.2.3.2	No	Yes	No	No
1-D-DGB	Diesel Generator 1B	1181	4.2.3.2	No	Yes	No	No
1-D-DTA	Diesel Generator Fuel Oil Day Tank A Enclosure	1182	4.2.3.2	No	Yes	No	No
1-D-DTB	Diesel Generator Fuel Oil Day Tank B Enclosure	1183	4.2.3.2	No	Yes	No	No
1-G	Turbine Generator Building	1115	4.2.3.2	No	Yes	No	No
1-O-PA	Diesel Oil Pump Room A	1184	4.2.3.2	No	Yes	No	No
1-O-PB	Diesel Oil Pump Room B	1185	4.2.3.2	No	Yes	No	No
5-F-BAL	Fuel Handling Building Balance of Areas	1186	4.2.3.2	No	Yes	No	No
5-F-CHF	Fuel Handling Building Emergency Exhaust	1187	4.2.3.2	No	Yes	No	No
5-F-FPP	Fuel Handling Building Fuel Pool Heat Exchangers	1188	4.2.3.2	No	Yes	No	No
5-O-BAL	Diesel Fuel Oil Storage Area Balance	1189	4.2.3.2	No	Yes	No	No
5-S-BAL	Auxiliary Reservoir Screening Structure	1190	4.2.3.2	No	Yes	No	No
5-W-BAL	Waste Processing Building	1114	4.2.3.2	No	Yes	No	No
FPYARD	Outside Yard	1191	4.2.3.2	No	Yes	No	No

4.8.2 Supplemental Information – Generic Issue Resolution

4.8.2.1 Fire Induced Multiple Spurious Operations Resolution

NEI 04-02 suggests that a licensee submit a summary of its approach for addressing potential fire-induced multiple spurious operations (MSOs) for NRC review and approval. As a minimum, NEI 04-02 suggests that the summary contain sufficient information relevant to methods, tools, and acceptance criteria used to enable the staff to determine the acceptability of the licensee's methodology. Attachment F of this document contains the methodology that Progress Energy is submitting for NRC approval.

4.8.2.2 Operator Manual Actions – Transition to Recovery Actions

The acceptability of post-fire operator manual actions developed into an industry issue in 2001. Inspection activity, subsequent meetings, and interpretations evolved into a plan for rulemaking in SECY 03-0100, Rulemaking Plan on Post-Fire Operator Manual Actions, dated June 17, 2003. The NRC withdrew its proposed rulemaking related to post-fire operator manual actions in 71 FR 11169 on March 6, 2006 as outlined in SECY 06-0010, Withdraw Proposed Rulemaking - Fire Protection Program Post-Fire Operator Manual Actions. NRC RIS 2006-10, Regulatory Expectations with Appendix R Paragraph III.G.2 Operator Manual Actions, was issued to provide clarification on the topic.

NEI 04-02 FAQ 06-0012, Determining Operator Manual Actions that Require a Change Evaluation During Transition, was developed to provide clarification on allowable operator manual actions and to define the scope of the associated risk-informed, performance-based change evaluations. FAQ 06-0012 includes a binning process to determine if post-fire operator manual actions are allowed under the pre-transition licensing basis. FAQ 06-0012, Revision 5 was accepted by the NRC via closure memo dated January 24, 2008 (ML072340368).

Additional Risk of Recovery Actions

Section 4.2.4 of NFPA 805 discusses the performance-based approach for meeting the nuclear safety performance criteria and includes the following statement:

"When the use of recovery actions has resulted in the use of this approach, the additional risk presented by their use shall be evaluated."

HNP Transition Process

HNP reviewed and documented pre-transition operator manual actions using the process from FAQ 06-0012. Operator manual actions categorized as "Bin H" per FAQ 06-0012 were evaluated for acceptability using the risk-informed, performance-based change evaluation process. In addition, the additional risk associated with recovery actions per Section 4.2.4 of NFPA 805 was evaluated using the Fire PRA. Refer to Attachment G for additional detail regarding the transition of operator manual actions.

4.8.2.3 Hemyc and MT Electrical Raceway Fire Barrier Systems

HNP Hemyc and MT Resolution Plans

On April 10, 2006, the NRC issued Generic Letter 2006-03, Potentially Nonconforming Hemyc and MT Fire Barrier Configurations, which requested licensees to provide information regarding the use of Hemyc and MT fire barriers:

- A statement on whether Hemyc or MT fire barrier material is used at their nuclear power plants and whether it is relied on for separation and/or safe shutdown purposes in accordance with the licensing basis, including whether Hemyc or MT is credited in other analyses (e.g., exemptions, license amendments, Generic Letter 86-10 analyses),
- A description of the controls that were used to ensure that other fire barrier types relied on for separation of redundant trains located in a single fire area are capable of providing the necessary level of protection,
- The extent of the installation (e.g., linear feet of wrap, areas installed, systems protected),
- Whether the Hemyc and/or MT installed in their plants is conforming with their licensing basis in light of recent findings, and if these recent findings do not apply, why not,
- The compensatory measures that have been implemented to provide protection and maintain the safe shutdown function of affected areas of the plant in light of the recent findings associated with Hemyc and MT installations, including evaluations to support the addressees' conclusions, and
- A description of, and implementation schedules for corrective actions, including a description of any licensing actions or exemption requests needed to support changes to the plant licensing basis.

HNP responded on June 9, 2006 to the Generic Letter concerns. HNP stated in part based on the fact that fire barrier issues are integrated with the NFPA 805 transition, the final resolution date for the Hemyc and MT fire barrier issue may extend beyond the License Amendment Request approval to allow for modifications for any applications not found acceptable during the NFPA 805 transition process.

The following provides a brief outline of the long-term resolution plan for Hemyc and MT fire barriers at HNP. This plan will consist of the following three (3) phases:

Phase One - Establishment of the ERFBS fire protective worth by plant-specific fire testing and evaluation of installed configurations,

Phase Two - Evaluation of the acceptability of the credited applications using the NFPA 805 change process, and

Phase Three - Resolution (e.g., upgrade) of remaining credited applications which were not found acceptable per NFPA 805.

The third phase of the project will fully address any HNP applications of the Hemyc and MT/ERFBS which were identified by the evaluation as not providing adequate protection for the required redundant SSD circuits.

Current Status

Phase One – All plant specific testing for Hemyc and MT is complete. Fire Barrier worth has been established for the HNP applications of Hemyc and MT that are credited to meet the Nuclear Safety Performance Criteria of NFPA 805 Chapter 4. Additionally, some applications of Hemyc and MT identified within the Generic Letter 60 Day Response (Reference 10) Attachment 2 were determined to not be required to meet the Nuclear Safety Performance Criteria of Chapter 4. Phase One is complete.

Phase Two – The HNP risk-informed, performance-based change process, discussed in Section 4.5 of the Transition Report, was used to assess the acceptability of the Hemyc and MT installations credited to meet the Nuclear Safety Performance Criteria requirements of Chapter 4. Phase Two is complete pending NRC approval of this LAR/TR.

Phase Three – The phase three change process will consist of modifying the Hemyc and MT applications to meet the fire barrier worth credited in the phase two evaluations. The following provides a summary of the Phase Three actions for the Hemyc and MT identified in the Generic Letter 60 Day Response Attachment 2:

- MT installations in Fire Areas 5-O-BAL, FPYARD, 1-A-ACP, 1-A-BAL-A (A2), and 1-A-BAL-D are not credited to meet the Nuclear Safety Performance criteria of NFPA 805 Chapter 4. Compensatory measures as described in the HNP Generic Letter 60 Day Response will remain in effect as long as the barriers are credited under the current license basis.
- MT installations in Fire Areas 1-A-BAL-C, 1-A-BAL-B, 1-A-BAL-A (A1), 12-A-BAL, and 1-A-SWGRB will be modified to meet the fire barrier worth rating credited within the risk-informed, performance-based change process. Compensatory measures as described in the HNP Generic Letter 60 Day Response will remain in effect until the modifications are completed and NRC issuance of the SER for transition to NFPA 805.
- Hemyc installations in Fire Areas 1-A-BAL-A (A4) are not credited to meet the Nuclear Safety Performance criteria of Chapter 4. Compensatory measures as described in the HNP Generic Letter 60 Day Response will remain in effect as long as the barriers are credited under the current license basis.
- The remaining Hemyc installations identified within Generic Letter 60 Day Response Attachment 2 will be modified to meet the fire barrier worth rating credited within the risk-informed, performance-based change process. Compensatory measures as described in the HNP Generic Letter 60 Day Response will remain in effect until the modifications are completed and NRC issuance of the SER for transition to NFPA 805.

4.8.3 Plant Modifications

The modifications necessary to support the new licensing basis will be complete by the end of Refueling Outage 16, currently scheduled for November 5, 2010.

Planned modifications to comply with NFPA 805:

Fire Prevention
- Install high efficiency incipient fire detection

Fire Suppression
- Install automatic fire suppression

Separation of Safe Shutdown Plant Equipment and Cables
- Upgrade Hemyc and MT wrap
- Reroute cables to provide additional separation

Attachment S includes a description of modifications already completed to date to meet NFPA 805 requirements.

5.0 REGULATORY EVALUATION

5.1 Introduction – 10 CFR 50.48

On July 16, 2004, the Nuclear Regulatory Commission amended 10 CFR Part 50.48, Fire Protection, to add a new subsection, 10 CFR 50.48(c), that established acceptable fire protection requirements. The change to 10 CFR 50.48 endorses, with exceptions, the National Fire Protection Association's 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition (NFPA 805), as an alternative for demonstrating compliance with 10 CFR 50.48 Section (b) and Section (f).

The adoption of 10 CFR 50.48(c) by HNP does not obviate the need to comply with 10 CFR 50.48(a) and 10 CFR 50, Appendix A, General Design Criterion (GDC) 3, Fire Protection. The NRC addressed the overall adequacy of the regulations during the promulgation of 10 CFR 50.48(a) (Reference Federal Register Notice 69 FR 33536 dated June 16, 2004, ADAMS Accession No. ML041340086).

The following Tables 5-1 through 5-3 provide a cross reference of fire protection regulations associated with the post-transition HNP fire protection program and applicable industry and HNP documents that address the topic.

Table 5-1 10 CFR 50.48(a) – Applicability/Compliance References

10 CFR 50.48(a) Section(s)		Applicability/Compliance Reference
(1)	Each holder of an operating license issued under this part or a combined license issued under part 52 of this chapter must have a fire protection plan that satisfies Criterion 3 of appendix A to this part. This fire protection plan must:	See below
	(i) Describe the overall fire protection program for the facility	NFPA 805 Section 3.2 HNP NEI 04-02 Table B-1
	(ii) Identify the various positions within the licensee's organization that are responsible for the program;	NFPA 805 Section 3.2.2 HNP NEI 04-02 Table B-1
	(iii) State the authorities that are delegated to each of these positions to implement those responsibilities; and	NFPA 805 Section 3.2.2 HNP NEI 04-02 Table B-1
	(iv) Outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage	NFPA 805 Section 2.7 and Chapters 3 and 4 HNP NEI 04-02 Table B-1 and Table B-3s
(2)	The plan must also describe specific features necessary to implement the program described in paragraph (a)(1) of this section such as	See below
	(i) Administrative controls and personnel requirements for fire prevention and manual fire suppression activities;	NFPA 805 Sections 3.3 and 3.4 HNP NEI 04-02 Table B-1
	(ii) Automatic and manually operated fire detection and suppression systems; and	NFPA 805 Sections 3.5 through 3.10 and Chapter 4 HNP NEI 04-02 Table B-1 and Table B-3s
	(iii) The means to limit fire damage to structures, systems, or components important to safety so that the capability to shut down the plant safely is ensured.	NFPA 805 Section 3.3 and Chapter 4 HNP NEI 04-02 Table B-3

Table 5-1 10 CFR 50.48(a) – Applicability/Compliance References

10 CFR 50.48(a) Section(s)	Applicability/Compliance Reference
(3) The licensee shall retain the fire protection plan and each change to the plan as a record until the Commission terminates the reactor license. The licensee shall retain each superseded revision of the procedures for 3 years from the date it was superseded.	NFPA 805 Section 2.7.1.1 requires that documentation be maintained for the life of the plant. RDC-NGGC-0001
(4) Each applicant for a design approval, design certification, or manufacturing license under part 52 of this chapter must have a description and analysis of the fire protection design features for the standard plant necessary to demonstrate compliance with Criterion 3 of appendix A to this part.	Not applicable to HNP. HNP is licensed under 10 CFR 50.

Table 5-2 GDC 3 – Applicability/Compliance References

GDC 3, Fire Protection, Statement	Applicability/Compliance Reference
Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.	NFPA 805 Chapters 3 and 4 HNP NEI 04-02 Table B-1 and Table B-3
Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room.	NFPA 805 Sections 3.3 and 3.11.4 HNP NEI 04-02 Table B-1
Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety.	NFPA 805 Chapters 3 and 4 HNP NEI 04-02 Table B-1 and Table B-3
Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components	NFPA 805 Sections 3.4 through 3.10 and 4.2.1 HNP NEI 04-02 Table B-3

Table 5-3 10 CFR 50.48(c) – Applicability/Compliance References

	10 CFR 50.48(c) Section(s)	Applicability/Compliance Reference
(1)	*Approval of incorporation by reference.* National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (NFPA 805), which is referenced in this section, was approved for incorporation by reference by the Director of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. Copies of NFPA 805 may be purchased from the NFPA Customer Service Department, 1 Batterymarch Park, P.O. Box 9101, Quincy, MA 02269-9101 and in PDF format through the NFPA Online Catalog (www.nfpa.org) or by calling 1-800- 344-3555 or 617-770-3000. Copies are also available for inspection at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852-2738, and at the NRC Public Document Room, Building One White Flint North, Room O1-F15, 11555 Rockville Pike, Rockville, Maryland 20852-2738. Copies are also available at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202-741-6030, or go to: http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html.	General Information. NFPA 805 (2001 edition) is the edition adopted by Progress Energy for HNP.
(2)	Exceptions, modifications, and supplementation of NFPA 805. As used in this section, references to NFPA 805 are to the 2001 Edition, with the following exceptions, modifications, and supplementation:	General Information. NFPA 805 (2001 edition) is the edition adopted by Progress Energy for HNP.
	(i) *Life Safety Goal, Objectives, and Criteria.* The Life Safety Goal, Objectives, and Criteria of Chapter 1 are not endorsed.	The Life Safety Goal, Objectives, and Criteria of Chapter 1 of NFPA 805 are not part of the HNP LAR.
	(ii) *Plant Damage/Business Interruption Goal, Objectives, and Criteria.* The Plant Damage/Business Interruption Goal, Objectives, and Criteria of Chapter 1 are not endorsed.	The Plant Damage/Business Interruption Goal, Objectives, and Criteria of Chapter 1 of NFPA 805 are not part of the HNP LAR.
	(iii) *Use of feed-and-bleed.* In demonstrating compliance with the performance criteria of Sections 1.5.1(b) and (c), a high-pressure charging/injection pump coupled with the pressurizer power-operated relief valves (PORVs) as the sole fire-protected safe shutdown path for maintaining reactor coolant inventory, pressure control, and decay heat removal capability (i.e., feed-and-bleed) for pressurized-water reactors (PWRs) is not permitted.	Feed and bleed is not utilized as the sole fire-protected safe shutdown path at HNP.
	(iv) Uncertainty analysis. An uncertainty analysis performed in accordance with Section 2.7.3.5 is not required to support deterministic approach calculations.	Uncertainty analysis was not performed for deterministic calculations at HNP.
	(v) Existing cables. In lieu of installing cables meeting flame propagation tests as required by Section 3.3.5.3, a flame-retardant coating may be applied to the electric cables, or an automatic fixed fire suppression system may be installed to provide an equivalent level of protection. In addition, the italicized exception to Section 3.3.5.3 is not endorsed.	Electrical cable construction at HNP complies with a flame propagation test that was found acceptable to the AHJ as documented in NEI 04-02 Table B-1.
	(vi) Water supply and distribution. The italicized exception to Section 3.5.4 is not endorsed. Licensees who wish to use the exception to Section 3.5.4 must submit a request for a license amendment in accordance with paragraph (c)(2)(vii) of this section.	HNP "complies via previous NRC approval" as documented in the NEI 04-02 Table B-1.

Table 5-3 10 CFR 50.48(c) – Applicability/Compliance References

	10 CFR 50.48(c) Section(s)	Applicability/Compliance Reference
	(vii) Performance-based methods. Notwithstanding the prohibition in Section 3.1 against the use of performance-based methods, the fire protection program elements and minimum design requirements of Chapter 3 may be subject to the performance-based methods permitted elsewhere in the standard. Licensees who wish to use performance-based methods for these fire protection program elements and minimum design requirements shall submit a request in the form of an application for license amendment under § 50.90. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the performance-based approach; (A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (B) Maintains safety margins; and (C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).	The HNP LAR requests the use of performance-based methods for NFPA 805 Chapter 3 requirements based upon FAQ 06-0008. This request is in accordance with 10 CFR 50.48(c)(2)(vii).
(3)	*Compliance with NFPA 805.*	See below
	(i) A licensee may maintain a fire protection program that complies with NFPA 805 as an alternative to complying with paragraph (b) of this section for plants licensed to operate before January 1, 1979, or the fire protection license conditions for plants licensed to operate after January 1, 1979. The licensee shall submit a request to comply with NFPA 805 in the form of an application for license amendment under § 50.90. The application must identify any orders and license conditions that must be revised or superseded, and contain any necessary revisions to the plant's technical specifications and the bases thereof. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the licensee has identified orders, license conditions, and the technical specifications that must be revised or superseded, and that any necessary revisions are adequate. Any approval by the Director or the designee must be in the form of a license amendment approving the use of NFPA 805 together with any necessary revisions to the technical specifications.	The HNP LAR was submitted in accordance with 10 CFR 50.90. The LAR included applicable license conditions, orders, technical specifications/bases that needed to be revised and/or superseded.
	(ii) The licensee shall complete its implementation of the methodology in Chapter 2 of NFPA 805 (including all required evaluations and analyses) and, upon completion, modify the fire protection plan required by paragraph (a) of this section to reflect the licensee's decision to comply with NFPA 805, before changing its fire protection program or nuclear power plant as permitted by NFPA 805.	The HNP LAR and transition report summarize the evaluations and analyses performed in accordance with Chapter 2 of NFPA 805.
(4)	Risk-informed or performance-based alternatives to compliance with NFPA 805. A licensee may submit a request to use risk-informed or performance-based alternatives to compliance with NFPA 805. The request must be in the form of an application for license amendment under § 50.90 of this chapter. The Director of the Office of Nuclear Reactor Regulation, or designee of the Director, may approve the application if the Director or designee determines that the proposed alternatives:	No risk-informed or performance-based alternatives to compliance with NFPA 805 (per 10 CFR 50.48(c)(4)) were utilized by HNP.

Table 5-3 10 CFR 50.48(c) – Applicability/Compliance References

10 CFR 50.48(c) Section(s)	Applicability/Compliance Reference
(i) Satisfy the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;	Not applicable to HNP.
(ii) Maintain safety margins; and	Not applicable to HNP.
(iii) Maintain fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).	Not applicable to HNP.

5.2 Regulatory Topics

5.2.1 License Condition Changes

The current HNP fire protection license condition 2.F is being replaced with the standard license condition in Regulatory Position C.3.1 of Regulatory Guide 1.205, Revision 0, as modified by FAQ 06-0008, as shown in Attachment M. In support of this change, HNP has developed a fire Probabilistic Risk Assessment (PRA) which has been reviewed and been found acceptable by the NRC during the course of its observation of HNP's transition to NFPA 805 as a Pilot Plant.

5.2.2 Technical Specifications

HNP conducted a review of the HNP Technical Specifications to determine which Technical Specifications are required to be either revised or superseded. HNP determined that the changes to the Technical Specifications and applicable justification listed in Attachment N are adequate for HNP's adoption of the new fire protection licensing basis.

5.2.3 Orders and Exemptions

A review was conducted of the HNP docketed correspondence to determine if there were any orders or exemptions that needed to be superseded or revised. This review was performed to ensure that compliance with the physical protection requirements, security orders, and adherence to commitments applicable to their plant are maintained. A discussion of affected orders and exemptions is included in Attachment O.

5.3 Regulatory Evaluations

5.3.1 Significant Hazards Consideration

A written evaluation of the significant hazards consideration of a proposed license amendment is required by 10 CFR 50.92. Harris Nuclear Plant (HNP) has evaluated the proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92, a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

This evaluation is contained in Attachment U.

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.3.2 Environmental Consideration

Pursuant to 10 CFR 51.22(b), an evaluation of the license amendment request (LAR) has been performed to determine whether it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c). That evaluation is discussed in Attachment V. The evaluation confirms that this LAR meets the criteria set forth in 10 CFR 51.22(c)(9) for categorical exclusion from the need for an environmental impact assessment or statement.

5.4 Transition Implementation Schedule

The following schedule for transitioning HNP to the new fire protection licensing basis requires NRC approval of the license amendment in accordance with NRC's current schedule.

- Implementation of 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.
- Completion of NFPA 805 transition modifications: 4th Quarter 2010.

  Appropriate compensatory measures for any outstanding NFPA 805 related modifications will be maintained at the time of NFPA 805 program implementation.

6.0 REFERENCES

1. ANSI/ANS-58.23-2007, American National Standard - Fire PRA Methodology, November 20, 2007
2. ASME/ANS-RA-S-2007, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Application, Combined PRA Standard
3. ASME RA-S-2002, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (and 2007 addenda ASME RA-Sc-2007, Appendix A)
4. Generic Letter 2006-03, Potentially Nonconforming Hemyc and MT Fire Barrier Configurations, April 10, 2006
5. Letter, NRC to NEI, Process for Frequently Asked Questions For Title 10 of The Code Of Federal Regulations, Part 50.48(c) Transitions, July 12, 2006 (ML061660105)
6. Letter, NRC to Progress Energy, NRC Response To Progress Energy's Letter Of Intent To Adopt 10 CFR 50.48(c) (NFPA 805 Rule), September 19, 2005 (ML052140391)
7. Letter, NRC to Progress Energy, Period of Enforcement Discretion during Implementation of National Fire Protection Association Standard 805, Shearon Harris Unit 1, H. B. Robinson Unit 2, Brunswick Units 1 And 2, and Crystal River Unit 3, April 29, 2007 (ML070590625)
8. Letter, NRC to Progress Energy, Shearon Harris Nuclear Power Plant, Unit 1 – Preliminary Results of the NRC Staff Review of The Fire Probabilistic Risk Assessment Model to Support Implementation of National Fire Protection Association Standard NFPA-805, "Performance-Based Standard For Fire Protection for Light Water Reactor Electric Generating Plants" (TAC No. MC5630), March 10, 2008 (ML080650403)
9. Letter, Progress Energy to NRC, Letter Of Intent to Adopt NFPA 805, Performance-Based Standard For Fire Protection For Light Water Reactor Electric Generating Plants, 2001 Edition, June 10, 2005 (ML051720404)
10. Letter, Progress Energy to NRC, 60-Day Response to NRC Generic Letter 2006-03, "Potentially Nonconforming Hemyc And Mt Fire Barrier Configurations June 9, 2006 (ML061710062)
11. NEI 00-01, Guidance for Post-Fire Safe Shutdown Circuit Analysis, Revision 1, January 2005
12. NEI-00-02, Industry PRA Peer Review Process
13. NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c), September 2005
14. NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition
15. NRC Enforcement Policy, Policy Statement: Revision, Federal Register, Vol. 69, No. 115, June 16, 2004, pp. 33684–33685
16. NRC Enforcement Policy: Extension of Discretion Period of Interim Enforcement Policy, Federal Register, Vol. 71, No. 74, April 18, 2006, pp. 19905-19907
17. NRC Enforcement Policy; Extension of Enforcement Discretion of Interim Policy," Policy Statement: Revision, Federal Register, Vol. 70, No. 10, January 14, 2005, pp. 2662–2664
18. NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management

19. NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 3, July 2000
20. NUREG/CR-6850, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, April 2005
21. Fire Protection Program—Post-Fire Operator Manual Actions, Federal Register, Vol. 71, No. 43, March 6, 2006, pp. 11169-11172.
22. Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 1, November 2002.
23. Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision. 1 - January 2007)
24. Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, May 2006
25. Regulatory Information Summary 2006-10, Regulatory Expectations with Appendix R Paragraph III.G.2 Operator Manual Actions, June 30, 2006
26. Regulatory Information Summary 2007-19, Process For Communicating Clarifications Of Staff Positions Provided In Regulatory Guide 1.205 Concerning Issues Identified During The Pilot Application of NFPA 805, August 20, 2007
27. SECY 03-0100, Rulemaking Plan on Post-Fire Operator Manual Actions, June 17, 2003
28. SECY 06-0010, Withdraw Proposed Rulemaking - Fire Protection Program Post-Fire Operator Manual Actions, January 12, 2006
29. Voluntary Fire Protection Requirement for Light-Water Reactors; Adoption of NFPA 805 as a Risk-Informed, Performance-Based Alternative, Final Rule, Federal Register, Vol. 69, No. 115, June 16, 2004, pp. 33536–33551
30. Voluntary Fire Protection Requirements for Light-Water Reactors; Adoption of NFPA 805 as a Risk-Informed, Performance-Based Alternative, Proposed Rule, Federal Register, Vol. 67, No. 212, November 1, 2002, pp. 66578–66588.

SECURITY RELATED INFORMATION – WITHHOLD UNDER 10 CFR 2.390

Attachment A – NEI 04-02 Table B-1 - Transition of Fundamental FP Program and Design Elements (NFPA 805 Chapter 3)

46 Pages

Attachment B – NEI 04-02 Table B-2 – Nuclear Safety Capability Assessment - Methodology Review

106 Pages

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

A comprehensive list of systems and equipment and their interrelationships to be analyzed for a fire event shall be developed. The equipment list shall contain an inventory of those critical components required to achieve the nuclear safety performance criteria of Section 1.5. Components required to achieve and maintain the nuclear safety functions and components whose fire-induced failure could prevent the operation or result in the maloperation of those components needed to meet the nuclear safety criteria shall be included. Availability and reliability of equipment selected shall be evaluated.

NEI 00-01 Ref	NEI 00-01 Guidance
3 Deterministic Methodology	This section discusses a generic deterministic methodology and criteria that licensees can use to perform a post-fire safe shutdown analysis to address regulatory requirements. The plant-specific analysis approved by NRC is reflected in the plant's licensing basis. The methodology described in this section is also an acceptable method of performing a post-fire safe shutdown analysis. This methodology is indicated in Figure 3-1. Other methods acceptable to NRC may also be used. *Regardless of the method selected by an individual licensee, the criteria and assumptions provided in this guidance document may apply. The methodology* described in Section 3 is based on a computer database oriented approach, which is utilized by several licensees to model Appendix R data relationships. This guidance document, however, does not require the use of a computer database oriented approach. The requirements of Appendix R Sections III.G.1, III.G.2 and III.G.3 apply to equipment and cables required for achieving and maintaining safe shutdown in any fire area. Although equipment and cables for fire detection and suppression systems, communications systems and 8-hour emergency lighting systems are important features, this guidance document does not address them. Additional information is provided in Appendix B to this document.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	Shearon Harris' safe shutdown methodology was reviewed against the guidelines of NUREG-0800; the corresponding sections of NUREG-0800 are C.5.b and C.5.c. Except as noted in the SER and supplements, the plant's methodology met the guidelines of NUREG-0800.			NUREG-1038 Supplement 2, Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Unit 1 - Docket No. STN-50-400, Rev. SSER 2, 6/1/1985	Section 9.5.1
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B

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NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1 [A, Intro] Safe Shutdown Systems and Path Development	This section discusses the identification of systems available and necessary to perform the required safe shutdown functions. It also provides information on the process for combining these systems into safe shutdown paths. Appendix R Section III.G.1.a requires that the capability to achieve and maintain hot shutdown be free of fire damage. It is expected that the term "free of fire damage" will be further clarified in a forthcoming Regulatory Issue Summary. Appendix R Section III.G.1.b requires that repairs to systems and equipment necessary to achieve and maintain cold shutdown be completed within 72 hours. It is the intent of the NRC that requirements related to the use of manual operator actions will be addressed in a forthcoming rulemaking. [Refer to hard copy of NEI 00-01 for Figure 3-1]

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	The corresponding guidelines for Harris are found in NUREG-0800, BTP CMEB 9.5-1 Sections C.5.b(1) and (2). Section B.3 of HNP-E/ELEC-0001 define the safe shutdown divisions for the plant, and Section B.5.1 discusses component selection. As pointed out in Section 4.2.1.2, given a fire, NFPA 805 does not require a plant to transition to cold shutdown. The fire area-by-fire area assessment documents the method of accomplishment of the NFPA 805 performance goals (including the transition to cold shutdown). During transition, Shearon Harris did not attempt to change the safe shutdown analysis to remove equipment/cables (and compliance strategies) that were only required to achieve and maintain cold shutdown. However, as allowed by the NFPA 805 change process and the revised license condition, Shearon Harris may revise these strategies post-transition.			NUREG-1038 Supplement 3, Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Unit 1 - Docket No. STN-50-400, Rev. SSER 3, 5/1/1986	Section 9.5.1.4
				NUREG-1038 Supplement 2, Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Unit 1 - Docket No. STN-50-400, Rev. SSER 2, 6/1/1985	Section 9.5.1

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

Aligns with intent	The corresponding guidelines for Harris are found in NUREG-0800, BTP CMEB 9.5-1 Sections C.5.b(1) and (2). Section B.3 of HNP-E/ELEC-0001 define the safe shutdown divisions for the plant, and Section B.5.1 discusses component selection. As pointed out in Section 4.2.1.2, given a fire, NFPA 805 does not require a plant to transition to cold shutdown. The fire area-by-fire area assessment documents the method of accomplishment of the NFPA 805 performance goals (including the transition to cold shutdown). During transition, Shearon Harris did not attempt to change the safe shutdown analysis to remove equipment/cables (and compliance strategies) that were only required to achieve and maintain cold shutdown. However, as allowed by the NFPA 805 change process and the revised license condition, Shearon Harris may revise these strategies post-transition.		HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.3 and B.5.1

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1 [B, Goals] Safe Shutdown Systems and Path Development	The goal of post-fire safe shutdown is to assure that one train of shutdown systems, structures, and components remains free of fire damage for a single fire in any single plant fire area. This goal is accomplished by determining those functions important to achieve and maintain hot shutdown. Safe shutdown systems are selected so that the capability to perform these required functions is a part of each safe shutdown path. The functions important to post-fire safe shutdown generally include, but are not limited to the following: Reactivity control Pressure control systems Inventory control systems Decay heat removal systems Process monitoring Support systems - Electrical systems - Cooling systems These functions are of importance because they have a direct bearing on the safe shutdown goal of being able to achieve and maintain hot shutdown which ensures the integrity of the fuel, the reactor pressure vessel, and the primary containment. If these functions are preserved, then the plant will be safe because the fuel, the reactor and the primary containment will not be damaged. By assuring that this equipment is not damaged and remains functional, the protection of the health and safety of the public is assured.

Applicability	Comments
Applicable	This is generic guidance and information that applies to all existing safe shutdown analyses.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	HNP-E-ELEC-0001 defines the safe shutdown goals and functions for Shearon Harris.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008 HNP SER initial and Supplement 4	Section B.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

***NFPA 805 Section:* 2.4.2.1 Nuclear Safety Capability System and Equipment Selection**

NEI 00-01 Ref	NEI 00-01 Guidance
3.1 [C, Spurious Operations] Safe Shutdown Systems and Path Development	In addition to the above listed functions, Generic Letter 81-12 specifies consideration of associated circuits with the potential for spurious equipment operation and/or loss of power source, and the common enclosure failures. Spurious operations/actuations can affect the accomplishment of the post-fire safe shutdown functions listed above. Typical examples of the effects of the spurious operations of concern are the following: - A loss of reactor pressure vessel/reactor coolant inventory in excess of the safe shutdown makeup capability - A flow loss or blockage in the inventory makeup or decay heat removal systems being used for the required safe shutdown path. Spurious operations are of concern because they have the potential to directly affect the ability to achieve and maintain hot shutdown, which could affect the fuel and cause damage to the reactor pressure vessel or the primary containment. Common power source and common enclosure concerns could also affect these and must be addressed.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The Harris Safe Shutdown Analysis has considered the three types of associated circuits discussed in NRC Generic Letter 81-12.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections A.2.1, B.7.1, B.7.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1 Criteria / Assumptions	The following criteria and assumptions may be considered when identifying systems available and necessary to perform the required safe shutdown functions and combining these systems into safe shutdown paths.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	This is generic introductory information and contains no specific guidance.				

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.1 [GE BWR Paths]	[BWR] GE Report GE-NE-T43-00002-00-01-R01 entitled "Original Safe Shutdown Paths For The BWR" addresses the systems and equipment originally designed into the GE boiling water reactors (BWRs) in the 1960s and 1970s, that can be used to achieve and maintain safe shutdown per Section III.G.1 of 10CFR 50, Appendix R. Any of the shutdown paths (methods) described in this report are considered to be acceptable methods for achieving redundant safe shutdown.

Applicability	Comments
Not Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
N/A	Shearon Harris is a PWR, and this guidance is specific to BWRs.				

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.2 [SRVs / LP Systems]	[BWR] GE Report GE-NE-T43-00002-00-03-R01 provides a discussion on the BWR Owners' Group (BWROG) position regarding the use of Safety Relief Valves (SRVs) and low pressure systems (LPCI/CS) for safe shutdown. The BWROG position is that the use of SRVs and low pressure systems is an acceptable methodology for achieving redundant safe shutdown in accordance with the requirements of 10CFR50 Appendix R Sections III.G.1 and III.G.2. The NRC has accepted the BWROG position and issued an SER dated Dec. 12, 2000.

Applicability	Comments
Not Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
N/A	Shearon Harris is a PWR, and this guidance is specific to BWRs.				

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.3 [Pressurizer Heaters]	[PWR] Generic Letter 86-10, Enclosure 2, Section 5.3.5 specifies that hot shutdown can be maintained without the use of pressurizer heaters (i.e., pressure control is provided by controlling the makeup/charging pumps). Hot shutdown conditions can be maintained via natural circulation of the RCS through the steam generators. The cooldown rate must be controlled to prevent the formation of a bubble in the reactor head. Therefore, feedwater (either auxiliary or emergency) flow rates as well as steam release must be controlled.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	*Though not specifically required based on the* guidance of NRC Generic Letter 86-10, the use of pressurizer heaters is not discouraged nor does their use conflict with the guidance of NEI 00-01. The Harris SSA does credit pressurizer heaters whenever they are available. The availability of the heaters enhances the operator's ability to ensure that cooldown rate is controlled, and that the cooldown process adheres to the required pressure and temperature limits.			AOP-004, Remote Shutdown, Rev. 41	
				AOP-036, Safe Shutdown Following a Fire , Rev. 40	

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.4 [Alternative Shutdown Capability]	The classification of shutdown capability as alternative shutdown is made independent of the selection of systems used for shutdown. Alternative shutdown capability is determined based on an inability to assure the availability of a redundant safe shutdown path. Compliance to the separation requirements of Sections III.G.1 and III.G.2 may be supplemented by the use of manual actions to the extent allowed by the regulations and the licensing basis of the plant, repairs (cold shutdown only), exemptions, deviations, GL 86-10 fire hazards analyses or fire protection design change evaluations, as appropriate. These may also be used in conjunction with alternative shutdown capability.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Guidelines for alternative shutdown for Shearon Harris are found in NUREG-0800, BTP CMEB Section 9.5-1, Sections C.5.b(3) and C.5.c. Supplement 3 to the SER reviewed the plant's alternate shutdown capability.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	
				NUREG-1038 Supplement 3, Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Unit 1 - Docket No. STN-50-400, Rev. SSER 3, 5/1/1986	Page 9-6

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.5 [Initial Conditions]	At the onset of the postulated fire, all safe shutdown systems (including applicable redundant trains) are assumed operable and available for post-fire safe shutdown. Systems are assumed to be operational with no repairs, maintenance, testing, Limiting Conditions for Operation, etc. in progress. The units are assumed to be operating at full power under normal conditions and normal lineups.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	These are basic assumptions for all safe shutdown analyses and also apply to the Harris SSA.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections A.3.2;A.3.5

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.6 [Other Events in Conjunction with Fire]	No Final Safety Analysis Report accidents or other design basis events (e.g. loss of coolant accident, earthquake), single failures or non-fire induced transients need be considered in conjunction with the fire.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	NUREG 0800 Section C.1.b states that "Worst case" fires need not be postulated to be simultaneous with non-fire-related failures in safety systems, plant accidents, or the most severe natural phenomena. This assumption is stated in the Harris SSA as noted.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section A.3.6

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.7 [Offsite Power]	For the case of redundant shutdown, offsite power may be credited if demonstrated to be free of fire damage. Offsite power should be assumed to remain available for those cases where its availability may adversely impact safety (i.e., reliance cannot be placed on fire causing a loss of offsite power if the consequences of offsite power availability are more severe than its presumed loss). No credit should be taken for a fire causing a loss of offsite power. For areas where train separation cannot be achieved and alternative shutdown capability is necessary, shutdown must be demonstrated both where offsite power is available and where offsite power is not available for 72 hours.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The SSA credits offsite power where analysis has demonstrated that it will be available. A loss of offsite power was not assumed in areas where offsite power was not credited, so if the availability of offsite power could adversely affect safe shutdown, it was assumed to remain available. This assumption regarding the availability of non-credited offsite power was also applied to alternate shutdown areas.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections A.3.4, B.7.1.1.2 Items 6 & 20
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.1, Items 4 & 5

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.8 [Safety-Related Equipment]	Post-fire safe shutdown systems and components are not required to be safety-related.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	NUREG-0800, C.5.c(6). The referenced SSA section clearly states that post-fire safe shutdown trains may include non-safety related equipment.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.3

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.9 [72 Hour Coping]	The post-fire safe shutdown analysis assumes a 72-hour coping period starting with a reactor scram/trip. Fire-induced impacts that provide no adverse consequences to hot shutdown within this 72-hour period need not be included in the post-fire safe shutdown analysis. At least one train can be repaired or made operable within 72 hours using onsite capability to achieve cold shutdown.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	This is a base safe shutdown analysis assumption.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections A.1.1, B10.1.1 Item 4

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.10 [Manual / Automatic Initiation of Systems]	Manual initiation from the main control room or emergency control stations of systems required to achieve and maintain safe shutdown is acceptable where permitted by current regulations or approved by NRC; automatic initiation of systems selected for safe shutdown is not required but may be included as an option.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The manual operation of systems and components from the main control room (or ACP for alternate shutdown areas) is credited in the safe shutdown analysis, and these actions are considered allowable under the current licensing basis. The Harris SSA does not currently credit the manual initiation of engineered safeguards (ESFAS) systems.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.4, B.7.1.1.2.

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.11 [Multiple Affected Units]	Where a single fire can impact more than one unit of a multi-unit plant, the ability to achieve and maintain safe shutdown for each affected unit must be demonstrated.

Applicability	Comments
Not Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
N/A	Shearon Harris is a single unit site.				

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	**NEI 00-01 Guidance**
3.1.2 Shutdown Functions	*The following discussion on each of these shutdown functions provides guidance for selecting the systems and equipment required for safe shutdown. For* additional information on BWR system selection, refer to GE Report GE-NE-T43-00002-00-01-R01 entitled "Original Safe Shutdown Paths for the BWR."

Applicability	**Comments**
Applicable	

Alignment Statement	**Alignment Basis**	**Comments**	**Unit**	**Reference Document**	**Doc. Details**
Aligns	This is an introductory section with no specific requirements. The GE information does not apply to Shearon Harris.				

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.1 Reactivity Control	[BWR] Control Rod Drive System The safe shutdown performance and design requirements for the reactivity control function can be met without automatic scram/trip capability. Manual scram/reactor trip is credited. The post-fire safe shutdown analysis must only provide the capability to manually scram/trip the reactor. [PWR] Makeup/Charging There must be a method for ensuring that adequate shutdown margin is maintained by ensuring borated water is utilized for RCS makeup/charging.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	If an automatic reactor trip has not already occurred, the analysis credits a manual trip to establish the initial shutdown condition. The two credited sources of makeup water for post-fire safe shutdown are the boric acid tank and the RWST. The boric acid concentrations in each tank ensure that adequate shutdown margin will be maintained throughout the cooldown process. AOP-004 and AOP-036 (series) direct the operator to determine the amount of boron that must be added to ensure an adequate shutdown margin is maintained.			AOP-004, Remote Shutdown, Rev. 41	
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.2.1, B.4
				AOP-036, Safe Shutdown Following a Fire , Rev. 40	

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.2 Pressure Control Systems	The systems discussed in this section are examples of systems that can be used for pressure control. This does not restrict the use of other systems for this purpose. [BWR] Safety Relief Valves (SRVs) The SRVs are opened to maintain hot shutdown conditions or to depressurize the vessel to allow injection using low pressure systems. These are operated manually. Automatic initiation of the Automatic Depressurization System is not a required function. [PWR] Makeup/Charging RCS pressure is controlled by controlling the rate of charging/makeup to the RCS. Although utilization of the pressurizer heaters and/or auxiliary spray reduces operator burden, neither component is required to provide adequate pressure control. Pressure reductions are made by allowing the RCS to cool/shrink, thus reducing pressurizer level/pressure. Pressure increases are made by initiating charging/makeup to maintain pressurizer level/pressure. Manual control of the related pumps is acceptable.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Harris does credit operation of the pressurizer heaters and pressurizer PORVs to maintain or reduce RCS pressure as necessary during the cooldown process.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.2.3, B.4

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.3 Inventory Control	[BWR] Systems selected for the inventory control function should be capable of supplying sufficient reactor coolant to achieve and maintain hot shutdown. Manual initiation of these systems is acceptable. Automatic initiation functions are not required. [PWR]: Systems selected for the inventory control function should be capable of maintaining level to achieve and maintain hot shutdown. Typically, the same components providing inventory control are capable of providing pressure control. Manual initiation of these systems is acceptable. Automatic initiation functions are not required.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The same systems used for post reactor trip pressure control will also be used for inventory control. Specifically, the CVCS system using the boric acid tank(s) and the RWST as sources of makeup water are used to maintain pressurizer level. Manual operation of the CVCS from the control room is credited except where specifically precluded by potential fire damage.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.2.2, B.4

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.4 Decay Heat Removal	[BWR] Systems selected for the decay heat removal function(s) should be capable of: - Removing sufficient decay heat from primary containment, to prevent containment over-pressurization and failure. - Satisfying the net positive suction head requirements of any safe shutdown systems taking suction from the containment (suppression pool). - Removing sufficient decay heat from the reactor to achieve cold shutdown. [PWR] Systems selected for the decay heat removal function(s) should be capable of: - Removing sufficient decay heat from the reactor to reach hot shutdown conditions. Typically, this entails utilizing natural circulation in lieu of forced circulation via the reactor coolant pumps and controlling steam release via the Atmospheric Dump valves. - Removing sufficient decay heat from the reactor to reach cold shutdown conditions. This does not restrict the use of other systems.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Harris uses the Auxiliary Feedwater System and Steam Generator PORVs to remove decay heat while in hot standby. Once temperature is reduced to about 350F, the RHR *system is placed in service to complete* the cooldown to cold shutdown conditions.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.2.4, B.4

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref

3.1.2.5 Process Monitoring

NEI 00-01 Guidance

The process monitoring function is provided for all safe shutdown paths. IN 84-09, Attachment 1, Section IX "Lessons Learned from NRC Inspections of Fire Protection Safe Shutdown Systems (10CFR50 Appendix R)" provides guidance on the instrumentation acceptable to and preferred by the NRC for meeting the process monitoring function. This instrumentation is that which monitors the process variables necessary to perform and control the functions specified in Appendix R Section III.L.1. Such instrumentation must be demonstrated to remain unaffected by the fire. The IN 84-09 list of process monitoring is applied to alternative shutdown (III.G.3). IN 84-09 did not identify specific instruments for process monitoring to be applied to redundant shutdown (III.G.1 and III.G.2). In general, process monitoring instruments similar to those listed below are needed to successfully use existing operating procedures (including Abnormal Operating Procedures).

BWR
- Reactor coolant level and pressure
- Suppression pool level and temperature
- Emergency or isolation condenser level
- Diagnostic instrumentation for safe shutdown systems
- Level indication for tanks needed for safe shutdown

PWR
- Reactor coolant temperature (hot leg / cold leg)
- Pressurizer pressure and level
- Neutron flux monitoring (source range)
- Level indication for tanks needed for safe shutdown
- Steam generator level and pressure
- Diagnostic instrumentation for safe shutdown systems

The specific instruments required may be based on operator preference, safe shutdown procedural guidance strategy (symptomatic vs. prescriptive), and systems and paths selected for safe shutdown.

Applicability

Applicable

Comments

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	The process monitoring function is capable of providing direct readings of those plant process variables necessary for plant operators to perform and/or control the identified safe shutdown functions. In selected fire areas, indirect readings are used for some variables (for example, pressurizer level as indication of proper charging system operation, or steam generator pressure to determine steam generator saturation temperature in lieu of RCS cold cold leg temperature).			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.2.5, B.5.1.2

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.6'Support Systems	[Blank Heading - No specific guidance]

Applicability	Comments
Not Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
N/A	Support system requirements will be addressed under the corresponding NEI 00-01 sub-section.				

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.6.1 Electrical Systems	AC Distribution System Power for the Appendix R safe shutdown equipment is typically provided by a medium voltage system such as 4.16 KV Class 1E busses either directly from the busses or through step down transformers/load centers/distribution panels for 600, 480 or 120 VAC loads. For redundant safe shutdown performed in accordance with the requirements of Appendix R Section III.G.1 and 2, power may be supplied from either offsite power sources or the emergency diesel generator depending *on which has been demonstrated to be free of fire damage. No credit should be taken for a fire causing a loss of offsite power. Refer to Section* 3.1.1.7. DC Distribution System Typically, the 125VDC distribution system supplies DC control power to various 125VDC control panels including switchgear breaker controls. The 125VDC distribution panels may also supply power to the 120VAC distribution panels via static inverters. These distribution panels typically supply power for instrumentation necessary to complete the process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries are necessary to supply any required control power during the interim time period required for the diesel generators to become operational. Once the diesels are operational, the 125 VDC distribution system can be powered from the diesels through the battery chargers. [BWR] Certain plants are also designed with a 250VDC Distribution System that supplies power to Reactor Core Isolation Cooling and/or High Pressure Coolant Injection equipment. The DC control centers may also supply power to various small horsepower Appendix R safe shutdown system valves and pumps. If the DC system is relied upon to support safe shutdown without battery chargers being available, it must be verified that sufficient battery capacity exists to support the necessary loads for sufficient time (either until power is restored, or the loads are no longer required to operate).

Applicability	Comments
Applicable	For the DC Buses, the batteries are shown in the fault tree going into an "OR" gate with the corresponding battery charger. Thus, if only the battery is free of fire damage, success will not be achieved.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The power supply for each powered component was identified and included in the SSEL. The limited capacity of the battery to supply loads for more than a few hours was considered in the analysis, and is discussed in the CAFTA text file.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.1.2, Item 20
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.2.6, B.4, B.5.1.2
				HNP FSSPMD R21 00, Fire Safe Shutdown Program Manager Database, Rev. 021	

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.6.2 Cooling Systems [HVAC]	HVAC Systems HVAC Systems may be required to assure that safe shutdown equipment remains within its operating temperature range, as specified in manufacturer's literature or demonstrated by suitable test methods, and to assure protection for plant operations staff from the effects of fire (smoke, heat, toxic gases, and gaseous fire suppression agents). HVAC *systems may be required to support safe shutdown system operation, based on plant-specific configurations.* Typical uses include: - Main control room, cable spreading room, relay room - ECCS pump compartments - Diesel generator rooms - Switchgear rooms Plant-specific evaluations are necessary to determine which HVAC systems are essential to safe shutdown equipment operation.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	HVAC systems required for post-fire safe shutdown are included in the analysis.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.2.6, B.4

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.6.2 Cooling Systems [Main Section]	Various cooling water systems may be required to support safe shutdown system operation, based on plant-specific considerations. Typical uses include: - RHR/SDC/DH Heat Exchanger cooling water - Safe shutdown pump cooling (seal coolers, oil coolers) - Diesel generator cooling - HVAC system cooling water

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Cooling water systems required for post-fire safe shutdown are included in the analysis.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.2.6, B.4

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.3 Methodology for Shutdown System Selection	Refer to Figure 3-2 for a flowchart illustrating the various steps involved in selecting safe shutdown systems and developing the shutdown paths. The following methodology may be used to define the safe shutdown systems and paths for an Appendix R analysis: [Refer to hard copy of NEI 00-01 for Figure 3-2]

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	Systems are assigned to one of two (or both) safe shutdown divisions in lieu of paths. Possible combinations of systems are modeled in the CAFTA fault tree.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.3

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	**NEI 00-01 Guidance**
3.1.3.1 Identify safe shutdown functions	Review available documentation to obtain an understanding of the available plant systems and the functions required to achieve and maintain safe shutdown. Documents such as the following may be reviewed: - Operating Procedures (Normal, Emergency, Abnormal) - System descriptions - Fire Hazard Analysis - Single-line electrical diagrams -Piping and Instrumentation Diagrams (P&IDs) [BWR] GE Report GE-NE-T43-00002-00-01-R02 entitled "Original Shutdown Paths for the BWR"

Applicability	**Comments**
Applicable	

Alignment Statement	**Alignment Basis**	**Comments**	**Unit**	**Reference Document**	**Doc. Details**
Aligns	The general guidance provided in this section was followed in the development of the Harris SSA.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.1

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.3.2 Identify Combinations of Systems that Satisfy Each Safe Shutdown Function	Given the criteria/assumptions defined in Section 3.1.1, identify the available combinations of systems capable of achieving the safe shutdown functions of reactivity control, pressure control, inventory control, decay heat removal, process monitoring, and support systems such as electrical and cooling systems (refer to Section 3.1.2). This selection process does not restrict the use of other systems. In addition to achieving the required safe shutdown functions, consider spurious operations and power supply issues that could impact the required safe shutdown function.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The available equipment combinations are depicted in the CAFTA fault tree, and are further explained in the SSA.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.1 and 9.2
				HNP FSSPMD R21 00, Fire Safe Shutdown Program Manager Database, Rev. 021	
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.3 and B.5.1

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.3.3 Define Combinations of Systems for Each Safe Shutdown Path	Select combinations of systems with the capability of performing all of the required safe shutdown functions and designate this set of systems as a safe shutdown path. In many cases, safe shutdown paths may be defined on a divisional basis since the availability of electrical power and other support systems must be demonstrated for each path.

Applicability	Comments
Applicable	Specific safe shutdown paths need not be identified. This is an analytical tool that is more applicable to BWRs than to PWRs.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	The selected systems are not grouped together in specific safe shutdown "paths," but are depicted in an integrated fashion in the CAFTA fault tree and accompanying text file.			HNP FSSPMD R21 00, Fire Safe Shutdown Program Manager Database, Rev. 021	
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.5.1 and B.6.1
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.3.4 Assign Shutdown Paths to Each Combination of Systems	Assign a path designation to each combination of systems. The path will serve to document the combination of systems relied upon for safe shutdown in each fire area. Refer to Attachment 1 to this document (NEI 00-01) for an example of a table illustrating how to document the various combinations of systems for selected shutdown paths.

Applicability	Comments
Applicable	Safe shutdown paths are not defined at Shearon Harris. Equipment is defined as being required for Division I or Division II, and some components are required for both divisions. The component and system inter-relationships are also defined in the CAFTA fault tree.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	The safe shutdown paths are not identified individually, but are shown in an integrated fashion in the CAFTA fault tree. The use of such fault trees is discussed in NFPA-805, Appendix B, Section B.2.2.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.1 and 9.2.
				HNP FSSPMD R21 00, Fire Safe Shutdown Program Manager Database, Rev. 021	
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.3 and B.5.1

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2 Safe Shutdown Equipment Selection	The previous section described the methodology for selecting the systems and paths necessary to achieve and maintain safe shutdown for an exposure fire event (see Section 5.0 DEFINITIONS for "Exposure Fire"). This section describes the criteria/assumptions and selection methodology for identifying the specific safe shutdown equipment necessary for the systems to perform their Appendix R function. The selected equipment should be related back to the safe shutdown systems that they support and be assigned to the same safe shutdown path as that system. The list of safe shutdown equipment will then form the basis for identifying the cables necessary for the operation or that can cause the maloperation of the safe shutdown systems.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	Components are assigned to one (or both) of two safe shutdown divisions rather than specific safe shutdown paths, which is more applicable to BWRs. The possible combinations of systems to meet the safe shutdown functions are shown in the CAFTA fault tree.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.5.1.2
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.1.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1 Criteria / Assumptions	Consider the following criteria and assumptions when identifying equipment necessary to perform the required safe shutdown functions:

Applicability	Comments
Applicable	This is introductory guidance information, and contains no specific requirements.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	This section provides a general overview of the safe shutdown methodology suggested in NEI 00-01 and followed by Shearon Harris. Specific requirements and / or guidance discussed in NEI 00-01 are discussed in the sub-sections below.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.5

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1.1 [Primary Secondary Components]	3.2.1.1 Safe shutdown equipment can be divided into two categories. Equipment may be categorized as (1) primary components or (2) secondary components. Typically, the following types of equipment are considered to be primary components: - Pumps, motor operated valves, solenoid valves, fans, gas bottles, dampers, unit coolers, etc. - All necessary process indicators and recorders (i.e., flow indicator, temperature indicator, turbine speed indicator, pressure indicator, level recorder) - Power supplies or other electrical components that support operation of primary components (i.e., diesel generators, switchgear, motor control centers, load centers, power supplies, distribution panels, etc.). Secondary components are typically items found within the circuitry for a primary component. These provide a supporting role to the overall circuit function. Some secondary components may provide an isolation function or a signal to a primary component via either an interlock or input signal processor. Examples of secondary components include flow switches, pressure switches, temperature switches, level switches, temperature elements, speed elements, transmitters, converters, controllers, transducers, signal conditioners, hand switches, relays, fuses and various instrumentation devices. Determine which equipment should be included on the Safe Shutdown Equipment List (SSEL). As an option, include secondary components with a primary component(s) that would be affected by fire damage to the secondary component. By doing this, the SSEL can be kept to a manageable size and the equipment included on the SSEL can be readily related to required post-fire safe shutdown systems and functions.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	This section provides a general overview of the safe shutdown methodology suggested in NEI 00-01 and followed by Shearon Harris. Specific requirements and / or guidance outlined in NEI 00-01 are discussed below.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.5

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1.2 [Fire Damage to Mechanical Components (not electrically supervised)]	3.2.1.2 Assume that exposure fire damage to manual valves and piping does not adversely impact their ability to perform their pressure boundary or safe shutdown function (heat sensitive piping materials, including tubing with brazed or soldered joints, are not included in this assumption). Fire damage should be evaluated with respect to the ability to manually open or close the valve should this be necessary as a part of the post-fire safe shutdown scenario.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Due to the substantial nature of equipment and nature and location of combustibles, fire will not impact the pressure boundary function. A fire does not cause a valve to change position unless the fire also affects the electrical equipment or circuits capable of inducing spurious operation of the valve. Manual stroking of a valve once the fire is extinguished will be evaluated as part of the Manual Action Feasibility Evaluation.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.1
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section A.3.12

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1.3 [Manual Valve Positions]	Assume that manual valves are in their normal position as shown on P&IDs or in the plant operating procedures.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	A base assumption of the SSA is that the plant is in a "normal" operating lineup, which includes manual valves being in their normal position.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 3.44 and 9.3.1
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.5.1.3 and A.3.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1.4 [Check Valves]	Assume that a check valve closes in the direction of potential flow diversion and seats properly with sufficient leak tightness to prevent flow diversion. Therefore, check valves do not adversely affect the flow rate capability of the safe shutdown systems being used for inventory control, decay heat removal, equipment cooling or other related safe shutdown functions.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	There is no clear statement concerning check valves, other than that properly oriented check valves credited as system boundaries should be included in the SSEL, and that those in the flow path need not be included. Check valves credited as boundaries are included in the SSEL, but the assumption that they are leak tight is inherent in the analysis and not clearly stated.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.5.1.2, Item 9
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.1.2.5, 9.1.2.9

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1.5 [Instrument Failures]	Instruments (e.g., resistance temperature detectors, thermocouples, pressure transmitters, and flow transmitters) are assumed to fail upscale, midscale, or downscale as a result of fire damage, whichever is worse. An instrument performing a control function is assumed to provide an undesired signal to the control circuit.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	Per the references cited, instruments exposed to fire damage are assumed to fail. It is a generally accepted practice (that can be verified based on a review of the fire area by fire area analyses) that instruments are assumed to fail to their worst case position unless a specific position to the contrary is taken.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section A.3.13
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.1.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

***NFPA 805 Section:* 2.4.2.1 Nuclear Safety Capability System and Equipment Selection**

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1.6 [Spurious Components]	Identify equipment that could spuriously operate or mal-operate and impact the performance of equipment on a required safe shutdown path during the equipment selection phase. Consider Bin 1 of RIS 2004-03 during the equipment identification process.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Section 9.1.2.7 of FPIP-0104 directs that for boundaries formed by three normally closed valves or dampers in series, all three should be included in the SSEL. RIS 2004-03 is not specifically identified as the basis for identifying three series boundary valves/dampers.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.5.1.2
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.1.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1.7 [Instrument Tubing]	Identify instrument tubing that may cause subsequent effects on instrument readings or signals as a result of fire. Determine and consider the fire area location of the instrument tubing when evaluating the effects of fire damage to circuits and equipment in the fire area.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	*Instrument tubing and its fire area routing is* included in the FSSPMD. Instrument sensing lines exposed to fire are assumed by the SSA to result in erratic indications.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections A.3.13 and B.7.1.2 Item 8
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.1.6

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.2 Methodology for Equipment Selection	Refer to Figure 3-3 for a flowchart illustrating the various steps involved in selecting safe shutdown equipment. Use the following methodology to select the safe shutdown equipment for a post-fire safe shutdown analysis: [Refer to hard copy of NEI 00-01 for Figure 3-3]

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	This introductory section contains no specific requirement, The sub-paragraphs with specific requirements are addressed separately as required.				

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	**NEI 00-01 Guidance**
3.2.2.1 Identify the System Flow Path for Each Shutdown Path	Mark up and annotate a P&ID to highlight the specific flow paths for each system in support of each shutdown path. Refer to Attachment 2 for an example of an annotated P&ID illustrating this concept.

Applicability	**Comments**
Applicable	Harris maintains marked-up safe shutdown flow diagrams. Prior to the revalidation effort, these diagrams also served as the safe shutdown *equipment list, as a SSEL was not specifically generated.*

Alignment Statement	**Alignment Basis**	**Comments**	**Unit**	**Reference Document**	**Doc. Details**
Aligns with intent	Individual safe shutdown paths are not identified, but the available paths are shown on the Safe Shutdown Flow Diagrams and are also reflected in the CAFTA fault tree.			2165-G-1000S Series, Safe Shutdown Flow Diagrams, Rev. Latest	

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.2.2 Identify the Equipment in Each Safe Shutdown System Flow Path Including Equipment That May Spuriously Operate and Affect System Operation	Review the applicable documentation (e.g. P&IDs, electrical drawings, instrument loop diagrams) to assure that all equipment in each system's flow path has been identified. Assure that any equipment that could spuriously operate and adversely affect the desired system function(s) is also identified. If additional systems are identified which are necessary for the operation of the safe shutdown system under review, include these as systems required for safe shutdown. Designate these new systems with the same safe shutdown path as the primary safe shutdown system under review (Refer to Figure 3-1).

Applicability	Comments
Applicable	It is not necessary that systems and components be assigned to a specific safe shutdown path.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	The credited portions of the safe shutdown systems are shown on the SSD flow diagrams. The component's safe shutdown division (1 or 2) is also shown on these diagrams and is reflected in the FSSPMD. The safe shutdown divisions are defined in Section B.3 of the SSA.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.1.2
				2165-G-1000S Series, Safe Shutdown Flow Diagrams, Rev. Latest	
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.3 and B.5.1.2, and Appendix 2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.2.3 Develop a List of Safe Shutdown Equipment and Assign the Corresponding System and Safe Shutdown Path(s) Designation to Each.	Prepare a table listing the equipment identified for each system and the shutdown path that it supports. Identify any valves or other equipment that could spuriously operate and impact the operation of that safe shutdown system. Assign the safe shutdown path for the affected system to this equipment. During the cable selection phase, identify additional equipment required to support the safe shutdown function of the path (e.g., electrical distribution system equipment). Include this additional equipment in the safe shutdown equipment list. Attachment 3 to this document provides an example of a (SSEL). The SSEL identifies the list of equipment within the plant considered for safe shutdown and it documents various equipment-related attributes used in the analysis.

Applicability	Comments
Applicable	The Harris SSEL does not assign equipment to a specific safe shutdown path. The equipment and system inter-relationships required to meet the safe shutdown functions and goals are depicted in the CAFTA fault tree.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	The SSEL does not assign each component to a safe shutdown path, but it does assign components to safe shutdown divisions (SSD-1 or SSD-2) as defined in Section B.3 of the SSA (HNP-E/ELEC-0001).			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.1.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.3 and B.5.1.2, and Appendix 2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.2.4 Identify Equipment Information Required for the Safe Shutdown Analysis	Collect additional equipment-related information necessary for performing the post-fire safe shutdown analysis for the equipment. In order to facilitate the analysis, tabulate this data for each piece of equipment on the SSEL. Refer to Attachment 3 to this document for an example of a SSEL. Examples of related equipment data should include the equipment type, equipment description, safe shutdown system, safe shutdown path, drawing reference, fire area, fire zone, and room location of equipment. Other information such as the following may be useful in performing the safe shutdown analysis: normal position, hot shutdown position, cold shutdown position, failed air position, failed electrical position, high/low pressure interface concern, and spurious operation concern.

Applicability	Comments
Applicable	The contents and specific fields of the SSEL table should be modified according to each plant's needs and existing data.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Required equipment is shown on the marked up SSD flow diagrams and in the SSEL report from FSSPMD. The SSEL is included as Appendix 2 to HNP-E-ELEC-0001.			HNP FSSPMD R21 00, Fire Safe Shutdown Program Manager Database, Rev. 021	Appendix 2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.5.1.3 and Appendix 2
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.1.2.

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.1 Nuclear Safety Capability System and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.2.5 Identify Dependencies Between Equipment, Supporting Equipment, Safe Shutdown Systems and Safe Shutdown Paths.	In the process of defining equipment and cables for safe shutdown, identify additional supporting equipment such as electrical power and interlocked equipment. As an aid in assessing identified impacts to safe shutdown, consider modeling the dependency between equipment within each safe shutdown path either in a relational database or in the form of a Safe Shutdown Logic Diagram (SSLD). Attachment 4 provides an example of a SSLD that may be developed to document these relationships.

Applicability	Comments
Applicable	The equipment and system dependencies are modeled in the CAFTA fault tree and FSSPMD.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The CAFTA fault tree captures the system and equipment inter-dependencies. Power supply and associated circuit dependencies are also captured in the FSSPMD and provided in Appendix 7 of HNP-E/ELEC-0001. The text file that corresponds to the CAFTA fault tree is contained in Appendix 4 of HNP-E/ELEC-0001.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.1.2, 9.2, 9.3
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.6.1 and Appendices 4 and 7

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

2.4.2.2.1 Circuits Required in Nuclear Safety Functions. Circuits required for the nuclear safety functions shall be identified. This includes circuits that are required for operation, that could prevent the operation, or that result in the maloperation of the equipment identified in 2.4.2.1. This evaluation shall consider fire-induced failure modes such as hot shorts (external and internal), open circuits, and shorts to ground, to identify circuits that are required to support the proper operation of components required to achieve the nuclear safety performance criteria, including spurious operation and signals. This will ensure that a comprehensive population of circuitry is evaluated.
2.4.2.2.2 Other Required Circuits. Other circuits that share common power supply and/or common enclosure with circuits required to achieve nuclear safety performance criteria shall be evaluated for their impact on the ability to achieve nuclear safety performance criteria.

(a) Common Power Supply Circuits. Those circuits whose fire-induced failure could cause the loss of a power supply required to achieve the nuclear safety performance criteria shall be identified. This situation could occur if the upstream protection device (i.e., breaker or fuse) is not properly coordinated with the downstream protection device.
(b) Common Enclosure Circuits. Those circuits that share enclosures with circuits required to achieve the nuclear safety performance criteria and whose fire-induced failure could cause the loss of the required components shall be identified. The concern is that the effects of a fire can extend outside of the immediate fire area due to fire-induced electrical faults on inadequately protected cables or via inadequately sealed fire area boundaries.

NEI 00-01 Ref	NEI 00-01 Guidance
3.3 Safe Shutdown Cable Selection and Location	This section provides industry guidance on the recommended methodology and criteria for selecting safe shutdown cables and determining their potential impact on equipment required for achieving and maintaining safe shutdown of an operating nuclear power plant for the condition of an exposure fire. The Appendix R safe shutdown cable selection criteria are developed to ensure that all cables that could affect the proper operation or that could cause the maloperation of safe shutdown equipment are identified and that these cables are properly related to the safe shutdown equipment whose functionality they could affect. Through this cable-to-equipment relationship, cables become part of the safe shutdown path assigned to the equipment affected by the cable.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	This section provides a general overview of the safe shutdown methodology suggested in NEI 00-01 and followed by Shearon Harris. Specific requirements or guidance outlined in NEI 00-01 is discussed below.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.1

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1 Criteria / Assumptions	To identify an impact to safe shutdown equipment based on cable routing, the equipment must have cables that affect it identified. Carefully consider how cables are related to safe shutdown equipment so that impacts from these cables can be properly assessed in terms of their ultimate impact on safe shutdown system equipment. Consider the following criteria when selecting cables that impact safe shutdown equipment:

Applicability	Comments
Applicable	The functional requirements of the component should be considered during the cable selection process.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Generic information in this introductory section. Specific guidance is in the subsections below.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.1.1.2

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.1 [Cable Selection]	The list of cables whose failure could impact the operation of a piece of safe shutdown equipment includes more than those cables connected to the equipment. The relationship between cable and affected equipment is based on a review of the electrical or elementary wiring diagrams. To assure that all cables that could affect the operation of the safe shutdown equipment are identified, investigate the power, control, instrumentation, interlock, and equipment status indication cables related to the equipment. Consider reviewing additional schematic diagrams to identify additional cables for interlocked circuits that also need to be considered for their impact on the ability of the equipment to operate as required in support of post-fire safe shutdown. As an option, consider applying the screening criteria from Section 3.5 as a part of this section. For an example of this see Section 3.3.1.4.

Applicability	Comments
Applicable	At Harris, the FSSPMD is also used to "link" associated cables to the safe shutdown equipment they could adversely affect.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	FIR-NGGC-0101 discusses the cable selection process in significant detail.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.1.1.2

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.2 [Cables Affecting Multiple Components]	In cases where the failure (including spurious actuations) of a single cable could impact more than one piece of safe shutdown equipment, include the cable with each piece of safe shutdown equipment.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	Although the specific guidance contained in this section of NEI 00-01 is not repeated in any Harris document, the procedures do not preclude listing a given cable against more than one component. The FSSPMD links all cables that could affect the operation of a given component to that component when augmenting the CAFTA Fault Tree. In addition, each circuit analysis is performed independently from other circuit analyses, therefore a cable or circuit will not be overlooked because it may be in more than one circuit.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.1.1.2 and Appendix 6
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.3 [Isolation Devices]	Electrical devices such as relays, switches and signal resistor units are considered to be acceptable isolation devices. In the case of instrument loops, review the isolation capabilities of the devices in the loop to determine that an acceptable isolation device has been installed at each point where the loop must be isolated so that a fault would not impact the performance of the safe shutdown instrument function.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Isolation devices are defined in the SSA, HNP-E/ELEC-0001, Appendix 6, Section 2.1.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Appendix 6, Section 2.1

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.4 [Identify "Not Required" Cables]	Screen out cables for circuits that do not impact the safe shutdown function of a component (i.e., annunciator circuits, space heater circuits and computer input circuits) unless some reliance on these circuits is necessary. However, they must be isolated from the component's control scheme in such a way that a cable fault would not impact the performance of the circuit.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Cables that are not required for safe shutdown have an "A" entered in the FMEA section of the circuit information form in FSSPMD. The "A" indicates that the component "achieves" its safe shutdown function even if that cable is damaged by fire. Cables that were analyzed as part of the circuit analysis but are not electrically connected to the component being analyzed had an "N/A" entered in the FMEA columns.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Appendix 6
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Attachment 1, Item 4
				HNP FSSPMD R21 00, Fire Safe Shutdown Program Manager Database, Rev. 021	Circuit Information Form

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.5 [Identification of Power Supplies]	For each circuit requiring power to perform its safe shutdown function, identify the cable supplying power to each safe shutdown and/or required interlock component. Initially, identify only the power cables from the immediate upstream power source for these interlocked circuits and components (i.e., the closest power supply, load center or motor control center). Review further the electrical distribution system to capture the remaining equipment from the electrical power distribution system necessary to support delivery of power from either the offsite power source or the emergency diesel generators (i.e., onsite power source) to the safe shutdown equipment. Add this equipment to the safe shutdown equipment list. Evaluate the power cables for this additional equipment for associated circuits concerns.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Power supplies are linked to their components in FSSPMD in the "Power Supplies, Related, Auxiliary, and Other Important Circuits" portion of the Circuit Information Form. A standard note "A" entered for a power supply in this section indicates that the power supply is required for the component to perform its safe shutdown function. The power supply requirement is modeled in the CAFTA fault tree.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.3.1 and 9.3.4
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.7.1.1.2 and B.7.2, and Appendix 6

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.6 [ESFAS Initiation]	The automatic initiation logics for the credited post-fire safe shutdown systems are not required to support safe shutdown. Each system can be controlled manually by operator actuation in the main control room or emergency control station. If operator actions outside the MCR are necessary, those actions must conform to the regulatory requirements on manual actions. However, if not protected from the effects of fire, the fire-induced failure of automatic initiation logic circuits must not adversely affect any post-fire safe shutdown system function.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The actions required to mitigate spurious ESFAS signals have been identified and are included in HNP-E/ELEC-0001. Sections B.4 and B.7.1.1.2 reflect the consideration of spurious ESFAS signals in the safe shutdown analysis.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.4, B.7.1.1.2
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3.1, Item 23

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.7 [Circuit Coordination]	Cabling for the electrical distribution system is a concern for those breakers that feed associated circuits and are not fully coordinated with upstream breakers. With respect to electrical distribution cabling, two types of cable associations exist. For safe shutdown considerations, the direct power feed to a primary safe shutdown component is associated with the primary component. For example, the power feed to a pump is necessary to support the pump. Similarly, the power feed from the load center to an MCC supports the MCC. However, for cases where sufficient branch-circuit coordination is not provided, the same cables discussed above would also support the power supply. For example, the power feed to the pump discussed above would support the bus from which it is fed because, for the case of a common power source analysis, the concern is the loss of the upstream power source and not the connected load. Similarly, the cable feeding the MCC from the load center would also be necessary to support the load center.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	This is a discussion of the common power supply concern, which is taken into consideration in the safe shutdown analysis.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.2

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.2 Associated Circuit Cables	Appendix R, Section III.G.2, requires that separation features be provided for equipment and cables, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve hot shutdown. The three types of associated circuits were identified in Reference 6.1.5 and further clarified in a NRC memorandum dated March 22, 1982 from R. Mattson to D. Eisenhut, Reference 6.1.6. They are as follows: - Spurious actuations - Common power source - Common enclosure.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	This section provides an introduction to the requirements to analyze associated circuits. Specific requirements of NEI 00-01 are discussed below.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.7.1, B.7.2
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 3.3 and 9.3
				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	SSER 3, page 9-15

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.2 [A] Associated Circuit Cables - Cables Whose Failure May Cause Spurious Actuations	Safe shutdown system spurious actuation concerns can result from fire damage to a cable whose failure could cause the spurious actuation/mal-operation of equipment whose operation could affect safe shutdown. These cables are identified in Section 3.3.3 together with the remaining safe shutdown cables required to support control and operation of the equipment.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Cables that can cause an undesired spurious actuation are identified by an "S" in the FMEA code of the circuit information form in FSSPMD. They are evaluated in the SSA in the same manner as "required" cables.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.1 and Appendix 6
				HNP FSSPMD R21 00, Fire Safe Shutdown Program Manager Database, Rev. 021	Circuit Information Form
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3 and Attachment 1

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.2 [B] Associated Circuit Cables - Common Power Source Cables	The concern for the common power source associated circuits is the loss of a safe shutdown power source due to inadequate breaker/fuse coordination. In the case of a fire-induced cable failure on a non-safe shutdown load circuit supplied from the safe shutdown power source, a lack of coordination between the upstream supply breaker/fuse feeding the safe shutdown power source and the load breaker/fuse supplying the non-safe shutdown faulted circuit can result in loss of the safe shutdown bus. This would result in the loss of power to the safe shutdown equipment supplied from that power source preventing the safe shutdown equipment from performing its required safe shutdown function. Identify these cables together with the remaining safe shutdown cables required to support control and operation of the equipment. Refer to Section 3.5.2.4 for an acceptable methodology for analyzing the impact of these cables on post-fire safe shutdown.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The analysis has taken into account associated circuits by common power supply as defined by NRC Generic Letter 81-12 and its supplement.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.2
				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	SSER 3, page 9-15
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3.1

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.2 [C] Associated Circuit Cables - Common Enclosure Cables	The concern with common enclosure associated circuits is fire damage to a cable whose failure could propagate to other safe shutdown cables in the same enclosure either because the circuit is not properly protected by an isolation device (breaker/fuse) such that a fire-induced fault could result in ignition along its length, or by the fire propagating along the cable and into an adjacent fire area. This fire spread to an adjacent fire area could impact safe shutdown equipment in that fire area, thereby resulting in a condition that exceeds the criteria and assumptions of this methodology (i.e., multiple fires). Refer to Section 3.5.2.5 for an acceptable methodology for analyzing the impact of these cables on post-fire safe shutdown.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Analysis aligns based on a combination of design considerations and circuit coordination studies.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3.1
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.2

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.3 Methodology for Cable Selection and Location	Refer to Figure 3-4 for a flowchart illustrating the various steps involved in selecting the cables necessary for performing a post-fire safe shutdown analysis. Use the following methodology to define the cables required for safe shutdown including cables that may cause associated circuits concerns for a post-fire safe shutdown analysis: [Refer to hard copy of NEI 00-01 for Figure 3-4]

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	This is an introductory paragraph with no specific criteria. Requirements are in the subsequent subsections.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.1

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.3.1 Identify Circuits Required for the Operation of the Safe Shutdown Equipment	For each piece of safe shutdown equipment defined in section 3.2, review the appropriate electrical diagrams including the following documentation to identify the circuits (power, control, instrumentation) required for operation or whose failure may impact the operation of each piece of equipment: - Single-line electrical diagrams - Elementary wiring diagrams - Electrical connection diagrams - Instrument loop diagrams. For electrical power distribution equipment such as power supplies, identify any circuits whose failure may cause a coordination concern for the bus under evaluation. If power is required for the equipment, include the closest upstream power distribution source on the safe shutdown equipment list. Through the iterative process described in Figures 3-2 and 3-3, include the additional upstream power sources up to either the offsite or the emergency power source.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The circuit analysis procedure (contained in FIR-NGGC-0101) directs that all cables that could adversely affect the component's ability to perform its safe shutdown function be identified. It also includes the identification of all required power supplies.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.1

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.3.2 Identify Interlocked Circuits and Cables Whose Spurious Operation or Mal-operation Could Affect Shutdown	In reviewing each control circuit, investigate interlocks that may lead to additional circuit schemes, cables and equipment. Assign to the equipment any cables for interlocked circuits that can affect the equipment. While investigating the interlocked circuits, additional equipment or power sources may be discovered. Include these interlocked equipment or power sources in the safe shutdown equipment list (refer to Figure 3-3) if they can impact the operation of the equipment under consideration.

Applicability	Comments
Applicable	As an alternative to adding the interlocked equipment to the SSEL, it is acceptable to include the cables that are required for the interlocking function (or that could cause the spurious actuation) with the main component that was originally under consideration. Adding the components may ease the development of a suitable mitigating strategy in areas where the interlocked cables may be damaged by the fire.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	Interlocked circuits were either included in the analysis, or the interlocked contact or relay was assumed to be in its worst-case position. Associated circuits identified for each component are either included in the main circuit analysis with a code of "A" in the existing basis column, or are included by listing the applicable circuit in the "Power Supplies, Related, Auxiliary, and Other Important Circuits" on the Circuit Information Form.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.1.1.2

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.3.3 Assign Cables to the Safe Shutdown Equipment	Given the criteria/assumptions defined in Section 3.3.1, identify the cables required to operate or that may result in maloperation of each piece of safe shutdown equipment. Tabulate the list of cables potentially affecting each piece of equipment in a relational database including the respective drawing numbers, their revision and any interlocks that are investigated to determine their impact on the operation of the equipment. In certain cases, the same cable may support multiple pieces of equipment. Relate the cables to each piece of equipment, but not necessarily to each supporting secondary component. If adequate coordination does not exist for a particular circuit, relate the power cable to the power source. This will ensure that the power source is identified as affected equipment in the fire areas where the cable may be damaged.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Information is maintained in the FSSPMD, and is also provided in Attachment 7 of HNP-E/ELEC-0001.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.7.1.1.2 and B.7.2, Attachment 7
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.3

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5 Circuit Analysis and Evaluation	This section on circuit analysis provides information on the potential impact of fire on circuits used to monitor, control and power safe shutdown equipment. Applying the circuit analysis criteria will lead to an understanding of how fire damage to the cables may affect the ability to achieve and maintain post-fire safe shutdown in a particular fire area. This section should be used in conjunction with Section 3.4, to evaluate the potential fire-induced impacts that require mitigation. Appendix R Section III.G.2 identifies the fire-induced circuit failure types that are to be evaluated for impact from exposure fires on safe shutdown equipment. Section III.G.2 of Appendix R requires consideration of hot shorts, shorts-to-ground and open circuits.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	BTP CMEB 9.5-1 Section C.5.c.(7) requires consideration of hot shorts, shorts-to-ground and open circuits for NUREG-0800 plants. The Harris analysis did consider all of these circuit failure modes.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	**NEI 00-01 Guidance**
3.5.1 Criteria / Assumptions	Apply the following criteria/assumptions when performing fire-induced circuit failure evaluations.

Applicability	**Comments**
Applicable	

Alignment Statement	**Alignment Basis**	**Comments**	**Unit**	**Reference Document**	**Doc. Details**
Aligns	The circuit analysis performed at Shearon Harris aligns with the general criteria that follows this introductory section, which contains no specific requirements.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0 HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.1.1 [Circuit Failure Types and Impact]	Consider the following circuit failure types on each conductor of each unprotected safe shutdown cable to determine the potential impact of a fire on the safe shutdown equipment associated with that conductor. - A hot short may result from a fire-induced insulation breakdown between conductors of the same cable, a different cable or from some other external source resulting in a compatible but undesired impressed voltage or signal on a specific conductor. A hot short may cause a spurious operation of safe shutdown equipment. - An open circuit may result from a fire-induced break in a conductor resulting in the loss of circuit continuity. An open circuit may prevent the ability to control or power the affected equipment. An open circuit may also result in a change of state for normally energized equipment. (e.g. [for BWRs] loss of power to the Main Steam Isolation Valve (MSIV) solenoid valves due to an open circuit will result in the closure of the MSIVs). Note that RIS 2004-03 indicates that open circuits, as an initial mode of cable failures, are considered to be of very low likelihood. The risk-informed inspection process will focus on failures with relatively high probabilities. - A short-to-ground may result from a fire-induced breakdown of a cable insulation system, resulting in the potential on the conductor being applied to ground potential. A short-to-ground may have all of the same effects as an open circuit and, in addition, a short-to-ground may also cause an impact to the control circuit or power train of which it is a part. Consider the three types of circuit failures identified above to occur individually on each conductor of each safe shutdown cable on the required safe shutdown path in the fire area.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details

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Aligns with intent	Circuit analysis was performed and documented in the FSSPMD prior to performing the fire area assessments. Thus, only those cables that have been previously determined to adversely affect the ability of the component to perform its safe shutdown function have been identified as required cables. *The augmented CAFTA fault tree is* used to identify a success path using the minimum set of equipment that may actually be damaged by the fire. Note that the referenced section of Supplement 3 to the SER states in part: "Spurious operations due to stray voltages between cables within a common raceway (cable-to-cable faults) resulting from fire damage have been considered a noncredible event by the applicants. One reason for this is that conductor-to-conductor faults are much more likely to occur before cable-to-cable faults, and conductor-to-conductor faults would preclude cable-to-cable faults. To cause spurious operations by two-wire 125-V ac or dc control or power cable, the applicants indicated that two circuits in contiguous cables (one energized, one deenergized) would need to be damaged by the fire and reconnected in proper sequence. This could occur if, for example, the positive energized wire in the one cable were to be exposed (through cable and wire insulation) to the positive unenergized wire in the adjacent cable and were to make contact with each other. This could only occur in the unlikely event that the insulation for both cables and both wires was to be removed in the same general area to permit this contact. Much more likely is the possibility for contact between the positive and negative energized wires in one cable or for the energized positive wire to contact the metallic raceway where either contact would cause the circuit breaker to open, thus removing the possibility for spurious operation. On the basis of the above, the staff finds the applicants' response relating *to spurious operation of associated circuits as* a result of wire-to-wire or cable-to-cable faults acceptable."	FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3.2, 9.3.3 and Attachment 1.

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In performing the FMEA for the identified safe shutdown circuits, this position is taken into account in the RDM "Revised Design Methodology" column of the Circuit Information Form. The CDM FMEA does consider inter-cable hot shorts, and the fault tree augmentation uses the more conservative of the two. *Compliance strategies were* employed in the revalidation of the safe shutdown analysis that relied on no inter-cable hot shorts based on the approval granted in SSER 3.

In the fire PRA, no credit was taken for the SSER 3 position, as inter-cable hot shorts were considered with probabilities assigned in accordance with NUREG-6850. Any risk significant scenarios identified will be addressed. Post-transition analyses and modifications will consider inter-cable hot shorts to be a credible circuit failure mode, and the SSER 3 position will not be relied upon as the sole means of compliance.

HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections B.7.1.1.3, B.7.1.1.4, and Appendix 7
NUREG-1038 Supplement 3, Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Unit 1 - Docket No. STN-50-400, Rev. SSER 3, 5/1/1986	Page 9-15

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.1.2 [Circuit Contacts and Operational Modes]	Assume that circuit contacts are positioned (i.e., open or closed) consistent with the normal mode/position of the safe shutdown equipment as shown on the schematic drawings. The analyst must consider the position of the safe shutdown equipment for each specific shutdown scenario when determining the impact that fire damage to a particular circuit may have on the operation of the safe shutdown equipment.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Per the analysis, components are assumed to be in their normal operating position. Where component's have multiple "normal" positions, the most conservative position for the scenario under analysis is assumed. Similarly, components that have multiple safe shutdown functions were analyzed to perform all potential functions.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.1.1.2
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3.1

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.1.3 [Duration of Circuit Failures]	Assume that circuit failure types resulting in spurious operations exist until action has been taken to isolate the given circuit from the fire area, or other actions have been taken to negate the effects of circuit failure that is causing the spurious actuation. The fire is not assumed to eventually clear the circuit fault. Note that RIS 2004-03 indicates that fire-induced hot shorts typically self-mitigate after a limited period of time.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The analysis takes no credit for "self-mitigating" circuit failures. EGR-NGGC-0102, Attachment 4, Section 3.4, under the heading "Issues Requiring Further Research" states in part "Duration of hot shorts...Cable test data indicates that the duration of a hot short is limited; PE general methodology is to conservatively assume the hot short is maintained until action is taken to mitigate its affects."			EGR-NGGC-0102, Safe Shutdown/Fire Protection Review, Rev. 006	Attachment 4, Section 3.4
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.3.1

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.1.4 [Cable Failure Configurations]	When both trains are in the same fire area outside of primary containment, all cables that do not meet the separation requirements of Section III.G.2 are assumed to fail in their worst case configuration.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	All cables in the area under consideration are assumed to fail in their worst case configuration.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.4.1.11 and 9.4.12
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections A.3.10 and A.3.11

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NEI 00-01 Ref	NEI 00-01 Guidance
3.5.1.5 [A, Circuit Failure Risk Assessment Guidance]	The following guidance provides the NRC inspection focus from Bin 1 of RIS 2004-03 in order to identify any potential combinations of spurious operations with higher risk significance. Bin 1 failures should also be the focus of the analysis; however, NRC has indicated that other types of failures required by the regulations for analysis should not be disregarded even if in Bin 2 or 3. If Bin 1 changes in subsequent revisions of RIS 2004-03, the guidelines in the revised RIS should be followed.

Applicability	Comments
Applicable	Provides guidance on assessing the risk-significance of circuit failures based on RIS 2004-03, Rev. 1. Note that SSER 3 approved Harris' original methodology which considered inter-cable hot shorts to be non-credible (SSER 3, pages 9-15, 9-16). The approval was based on inter-cable shorts being less likely than intra-cable shorts or shorts to ground.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Shearon Harris performed a multiple spurious operations review in accordance with the guidelines of NRC RIS 2004-03. The results of the review are contained in Appendix 14 of the safe shutdown analysis.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Appendix 14

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NEI 00-01 Ref	NEI 00-01 Guidance
3.5.1.5 [B, *Cable Failure* Modes]	*For multiconductor cables testing has demonstrated that conductor-to-conductor shorting within the same cable is the most common mode of failure. This is often* referred to as "intra-cable shorting." It is reasonable to assume that given damage, more than one conductor-to-conductor short will occur in a given cable. A second primary mode of cable failure is conductor-to-conductor shorting between separate cables, commonly referred to as "inter-cable shorting." Inter-cable shorting is less likely than intra-cable shorting. Consistent with the current knowledge of fire-induced cable failures, the following configurations should be considered: A. For any individual multiconductor cable (thermoset or thermoplastic), any and all potential spurious actuations that may result from intra-cable shorting, including any possible combination of conductors within the cable, may be postulated to occur concurrently regardless of number. However, as a practical matter, the number of combinations of potential hot shorts increases rapidly with the number of conductors within a given cable. For example, a multiconductor cable with three conductors (3C) has 3 possible combinations of two (including desired combinations), while a five conductor cable (5C) has 10 possible combinations of two (including desired combinations), and a seven conductor cable (7C) has 21 possible combinations of two (including desired combinations). To facilitate an inspection that considers most of the risk presented by postulated hot shorts within a multiconductor cable, inspectors should consider only a few (three or four) of the most critical postulated combinations. B. For any thermoplastic cable, any and all potential spurious actuations that may result from intra-cable and inter-cable shorting with other thermoplastic cables, including any possible combination of conductors within or between the cables, may be postulated to occur concurrently regardless of number. (The consideration of thermoset cable inter-cable shorts is deferred pending additional research.) C. For cases involving the potential damage of more than one multiconductor cable, a maximum of two cables should be assumed to be damaged concurrently. The spurious actuations should be evaluated as previously described. The consideration of more than two cables being damaged (and subsequent spurious actuations) is deferred pending additional research. D. For cases involving direct current (DC) circuits, the potential spurious operation due to failures of the associated control cables (even if the spurious operation requires two concurrent hot shorts of the proper polarity, e.g., plus-to-plus and minus-to-minus) should be considered when the required source and target conductors are each located within the same multiconductor cable. E. Instrumentation Circuits. Required instrumentation circuits are beyond the scope of this associated circuit approach and must meet the same requirements as required power and control circuits. There is one case where an instrument circuit could potentially be considered an associated circuit. If fire-induced damage of an instrument circuit could prevent operation (e.g., lockout permissive signal) or cause maloperation (e.g., unwanted start/stop/reposition signal) of systems necessary to achieve and maintain hot shutdown, then the instrument circuit may be considered an associated circuit and handled accordingly.

Applicability	Comments
Applicable	Provides guidance on assessing the risk-significance of circuit failures based on RIS 2004-03, Rev. 1.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Shearon Harris performed a multiple spurious operations review in accordance with the guidelines of NRC RIS 2004-03. The results of the review are contained in Appendix 14 of the safe shutdown analysis.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Appendix 14

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NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2 Types of Circuit Failures	Appendix R requires that nuclear power plants must be designed to prevent exposure fires from defeating the ability to achieve and maintain post-fire safe shutdown. Fire damage to circuits that provide control and power to equipment on the required safe shutdown path and any other equipment whose spurious operation/mal-operation could affect shutdown in each fire area must be evaluated for the effects of a fire in that fire area. Only one fire at a time is assumed to occur. The extent of fire damage is assumed to be limited by the boundaries of the fire area. Given this set of conditions, it must be assured that one redundant train of equipment capable of achieving hot shutdown is free of fire damage for fires in every plant location. To provide this assurance, Appendix R requires that equipment and circuits required for safe shutdown be free of fire damage and that these circuits be designed for the fire-induced effects of a hot short, short-to-ground, and open circuit. With respect to the electrical distribution system, the issue of breaker coordination must also be addressed. This section will discuss specific examples of each of the following types of circuit failures: - Open circuit - Short-to-ground - Hot short.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	NUREG-0800 contains similar guidelines. This section provides a brief synopsis of safe shutdown requirements as an introduction to a detailed discussion of three specific types of circuit failures that are required to be postulated. These circuit failure types (open circuits, shorts to ground, and hot shorts) were all considered in the Harris SSA.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.1 Circuit Failures Due to an Open Circuit	This section provides guidance for addressing the effects of an open circuit for safe shutdown equipment. An open circuit is a fire-induced break in a conductor resulting in the loss of circuit continuity. An open circuit will typically prevent the ability to control or power the affected equipment. An open circuit can also result in a change of state for normally energized equipment. For example, a loss of power to the main steam isolation valve (MSIV) solenoid valves [for BWRs] due to an open circuit will result in the closure of the MSIV. NOTE: The EPRI circuit failure testing indicated that open circuits are not likely to be the initial fire-induced circuit failure mode. Consideration of this may be helpful within the safe shutdown analysis. Consider the following consequences in the safe shutdown circuit analysis when determining the effects of open circuits: Loss of electrical continuity may occur within a conductor resulting in de-energizing the circuit and causing a loss of power to, or control of, the required safe shutdown equipment. In selected cases, a loss of electrical continuity may result in loss of power to an interlocked relay or other device. This loss of power may change the state of the equipment. Evaluate this to determine if equipment fails safe. Open circuit on a high voltage (e.g., 4.16 kV) ammeter current transformer (CT) circuit may result in secondary damage. Figure 3.5.2-1 shows an open circuit on a grounded control circuit. [Refer to hard copy of NEI 00-01 for Figure 3.5.2-1] Open circuit No. 1: An open circuit at location No. 1 will prevent operation of the subject equipment. Open circuit No. 2: An open circuit at location No. 2 will prevent opening/starting of the subject equipment, but will not impact the ability to close/stop the equipment.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The Harris SSA does consider open circuits. This section provides information related to the effects of an open circuit on different types of typical circuits.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.1
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 3.9, 3.27, 3.47, and 9.3.2

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.2 Circuit Failures Due to a Short-to-Ground [A, General]	This section provides guidance for addressing the effects of a short-to-ground on circuits for safe shutdown equipment. A short-to-ground is a fire-induced breakdown of a cable insulation system resulting in the potential on the conductor being applied to ground potential. A short-to-ground can cause a loss of power to or control of required safe shutdown equipment. In addition, a short-to-ground may affect other equipment in the electrical power distribution system in the cases where proper coordination does not exist. Consider the following consequences in the post-fire safe shutdown analysis when determining the effects of circuit failures related to shorts-to-ground: - A short to ground in a power or a control circuit may result in tripping one or more isolation devices (i.e. breaker/fuse) and causing a loss of power to or control of required safe shutdown equipment. - In the case of certain energized equipment such as HVAC dampers, a loss of control power may result in loss of power to an interlocked relay or other device that may cause one or more spurious operations.

Applicability	Comments
Applicable	*This section provides specific examples of shorts to ground on a representative sample of typical control and power circuits*

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The Harris SSA does consider shorts to ground. This section provides information related to the effects of a short to ground on different types of typical circuits.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 3.9, 3.61, and 9.3.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections A.1.1, B.7.1

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.2 Circuit Failures Due to a Short-to-Ground [B, Grounded Circuits]	This section provides guidance for addressing the effects of a short-to-ground on circuits for safe shutdown equipment. A short-to-ground is a fire-induced breakdown of a cable insulation system resulting in the potential on the conductor being applied to ground potential. A short-to-ground can cause a loss of power to or control of required safe shutdown equipment. In addition, a short-to-ground may affect other equipment in the electrical power distribution system in the cases where proper coordination does not exist. Short-to-Ground on Grounded Circuits Typically, in the case of a grounded circuit, a short-to-ground on any part of the circuit would present a concern for tripping the circuit isolation device thereby causing a loss of control power. Figure 3.5.2-2 illustrates how a short-to-ground fault may impact a grounded circuit. [Refer to hard copy of NEI 00-01 Rev. 1 for Figure 3.5.2-2] Short-to-ground No. 1: A short-to-ground at location No. 1 will result in the control power fuse blowing and a loss of power to the control circuit. This will result an inability to operate the equipment using the control switch. Depending on the coordination characteristics between the protective device on this circuit and upstream circuits, the power supply to other circuits could be affected. Short-to-ground No. 2: A short-to-ground at location No. 2 will have no effect on the circuit until the close/stop control switch is closed. Should this occur, the effect would be identical to that for the short-to-ground at location No. 1 described above. Should the open/start control switch be closed prior to closing the close/stop control switch, the equipment will still be able to be opened/started.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The Harris SSA does consider shorts to ground. This section provides information related to the effects of a short to ground on typical grounded circuits.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 3.9, 3.61, and 9.3.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section A.1.1, B.7.1

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.2 Circuit Failures Due to a Short-to-Ground [C, Ungrounded Circuits]	Short-to-Ground on Ungrounded Circuits In the case of an ungrounded circuit, postulating only a single short-to-ground on any part of the circuit may not result in tripping the circuit isolation device. Another short-to-ground on the circuit or another circuit from the same source would need to exist to cause a loss of control power to the circuit. Figure 3.5.2-3 illustrates how a short to ground fault may impact an ungrounded circuit. [Refer to hard copy of NEI 00-01 Rev. 1 for Figure 3.5.2-3] Short-to-ground No. 1: A short-to-ground at location No. 1 will result in the control power fuse blowing and a loss of power to the control circuit if short-to-ground No. 3 also exists either within the same circuit or on any other circuit fed from the same power source. This will result in an inability to operate the equipment using the control switch. Depending on the coordination characteristics between the protective device on this circuit and upstream circuits, the power supply to other circuits could be affected. Short-to-ground No. 2: A short-to-ground at location No. 2 will have no effect on the circuit until the close/stop control switch is closed. Should this occur, the effect would be identical to that for the short-to-ground at location No. 1 described above. Should the open/start control switch be closed prior to closing the close/stop control switch, the equipment will still be able to be opened/started.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The Harris SSA does consider shorts to ground. This section provides information related to the effects of a short to ground on typical ungrounded circuits.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 3.9, 3.61, and 9.3.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section A.1.1, B.7.1

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.3 Circuit Failures Due to a Hot Short [A, General]	This section provides guidance for analyzing the effects of a hot short on circuits for required safe shutdown equipment. A hot short is defined as a fire-induced insulation breakdown between conductors of the same cable, a different cable or some other external source resulting in an undesired impressed voltage on a specific conductor. The potential effect of the undesired impressed voltage would be to cause equipment to operate or fail to operate in an undesired manner. Consider the following specific circuit failures related to hot shorts as part of the post-fire safe shutdown analysis: - A hot short between an energized conductor and a de-energized conductor within the same cable may cause a spurious actuation of equipment. The spuriously actuated device (e.g., relay) may be interlocked with another circuit that causes the spurious actuation of other equipment. This type of hot short is called a conductor-to-conductor hot short or an internal hot short. - A hot short between any external energized source such as an energized conductor from another cable (thermoplastic cables only) and a de-energized conductor may also cause a spurious actuation of equipment. This is called a cable-to-cable hot short or an external hot short. Cable-to-cable hot shorts between thermoset cables are not postulated to occur pending additional research.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The Harris SSA does consider hot shorts. This section provides information related to the effects of a hot short on typical circuits.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 3.9, 3.33, 3.34, 3.35, and 9.3.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections A.1.1, B.7.1

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.3 Circuit Failures Due to a Hot Short [B, Grounded Circuits]	A Hot Short on Grounded Circuits A short-to-ground is another failure mode for a grounded control circuit. A short-to-ground as described above would result in de-energizing the circuit. This would further reduce the likelihood for the circuit to change the state of the equipment either from a control switch or due to a hot short. Nevertheless, a hot short still needs to be considered. Figure 3.5.2-4 shows a typical grounded control circuit that might be used for a motor-operated valve. However, the protective devices and position indication lights that would normally be included in the control circuit for a motor-operated valve have been omitted, since these devices are not required to understand the concepts being explained in this section. In the discussion provided below, it is assumed that a single fire in a given fire area could cause any one of the hot shorts depicted. The following discussion describes how to address the impact of these individual cable faults on the operation of the equipment controlled by this circuit. [Refer to hard copy of NEI 00-01 Rev. 1 for Figure 3.5.2-4] Hot short No. 1: A hot short at this location would energize the close relay and result in the undesired closure of a motor-operated valve. Hot short No. 2: A hot short at this location would energize the open relay and result in the undesired opening of a motor-operated valve.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The Harris SSA does consider hot shorts. This section provides information related to the effects of a hot short on typical grounded circuits.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 3.9, 3.33, 3.34, 3.35, and 9.3.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections A.1.1, B.7.1

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.3 Circuit Failures Due to a Hot Short [C, Ungrounded Circuits]	A Hot Short on Ungrounded Circuits In the case of an ungrounded circuit, a single hot short may be sufficient to cause a spurious operation. A single hot short can cause a spurious operation if the hot short comes from a circuit from the positive leg of the same ungrounded source as the affected circuit. In reviewing each of these cases, the common denominator is that in every case, the conductor in the circuit between the control switch and the start/stop coil must be involved. Figure 3.5.2-5 depicted below shows a typical ungrounded control circuit that might be used for a motor-operated valve. However, the protective devices and position indication lights that would normally be included in the control circuit for a motor-operated valve have been omitted, since these devices are not required to understand the concepts being explained in this section. In the discussion provided below, it is assumed that a single fire in a given fire area could cause any one of the hot shorts depicted. The discussion provided below describes how to address the impact of these cable faults on the operation of the equipment controlled by this circuit. [Refer to hard copy of NEI 00-01 Rev. 1 for Figure 3.5.2-5] *Hot short* No. 1: A hot short at this location from the same control power source would energize the close relay and result in the undesired closure of a motor operated valve. Hot short No. 2: A hot short at this location from the same control power source would energize the open relay and result in the undesired opening of a motor operated valve.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The Harris SSA does consider hot shorts. This section provides information related to the effects of a hot short on typical ungrounded circuits.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 3.9, 3.33, 3.34, 3.35, and 9.3.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Sections A.1.1, B.7.1

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location.

Nuclear Safety Equipment and Cable Location. Physical location of equipment and cables shall be identified.

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.3.4 Identify Routing of Cables	Identify the routing for each cable including all raceway and cable endpoints. Typically, this information is obtained from joining the list of safe shutdown cables with an existing cable and raceway database

Applicability	Comments
Applicable	As a minimum, the cable to fire area information must be obtained.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Cable to raceway information is contained in the Cable Information Form of the FSSPMD. It is presented in Appendix 9 of HNP-E/ELEC-0001.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.3.1 and Appendix 9

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location.

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.3.5 Identify Location of Raceway and Cables by Fire Area	Identify the fire area location of each raceway and cable endpoint identified in the previous step and join this information with the cable routing data. In addition, identify the location of field-routed cable by fire area. This produces a database containing all of the cables requiring fire area analysis, their locations by fire area, and their raceway.

Applicability	Comments
Applicable	The particular raceway a cable is routed in within the fire area under consideration is important in a risk-informed, performance-based approach. Such information helps the analyst determine the extent to which the cable may be damaged in a credible fire scenario.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The fire area routing of each cable was identified and entered in the FSSPMD. Raceway and cable tray route point to fire area information is also contained.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.3 and Appendix 9

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location.

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.4 Circuit Failures Due to Inadequate Circuit Coordination	The evaluation of associated circuits of a common power source consists of verifying proper coordination between the supply breaker/fuse and the load breakers/fuses for power sources that are required for safe shutdown. The concern is that, for fire damage to a single power cable, lack of coordination between the supply breaker/fuse and the load breakers/fuses can result in the loss of power to a safe shutdown power source that is required to provide power to safe shutdown equipment. For the example shown in Figure 3.5.2-6, the circuit powered from load breaker 4 supplies power to a non-safe shutdown pump. This circuit is damaged by fire in the same fire area as the circuit providing power to from the Train B bus to the Train B pump, which is redundant to the Train A pump. To assure safe shutdown for a fire in this fire area, the damage to the non-safe shutdown pump powered from load breaker 4 of the Train A bus cannot impact the availability of the Train A pump, which is redundant to the Train B pump. To assure that there is no impact to this Train A pump due to the associated circuits' common power source breaker coordination issue, load breaker 4 must be fully coordinated with the feeder breaker to the Train A bus. [Refer to hard copy of NEI 00-01 Rev. 1 for Figure 3.5.2-6] A coordination study should demonstrate the coordination status for each required common power source. For coordination to exist, the time-current curves for the breakers, fuses and/or protective relaying must demonstrate that a fault on the load circuits is isolated before tripping the upstream breaker that supplies the bus. Furthermore, the available short circuit current on the load circuit must be considered to ensure that coordination is demonstrated at the maximum fault level. The methodology for identifying potential associated circuits of a common power source and evaluating circuit coordination cases of associated circuits on a single circuit fault basis is as follows: - Identify the power sources required to supply power to safe shutdown equipment. - For each power source, identify the breaker/fuse ratings, types, trip settings and coordination characteristics for the incoming source breaker supplying the bus and the breakers/fuses feeding the loads supplied by the bus. - For each power source, demonstrate proper circuit coordination using acceptable industry methods. - For power sources not properly coordinated, tabulate by fire area the routing of cables whose breaker/fuse is not properly coordinated with the supply breaker/fuse. Evaluate the potential for disabling power to the bus in each of the fire areas in which the associated circuit cables of concern are routed and the power source is required for safe shutdown. Prepare a list of the following information for each fire area: - Cables of concern. - Affected common power source and its path. - Raceway in which the cable is enclosed. - Sequence of the raceway in the cable route. - Fire zone/area in which the raceway is located. For fire zones/areas in which the power source is disabled, the effects are mitigated by appropriate methods. Develop analyzed safe shutdown circuit dispositions for the associated circuit of concern cables routed in an area of the same path as required by the power source. Evaluate adequate separation based upon the criteria in Appendix R, NRC staff guidance, and plant licensing bases.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location.

Aligns	Associated circuits by common power supply were identified and dispositioned during the cable selection and circuit analysis process. Where a lack of coordination created a compliance issue, the cables were dispositioned in a manner similar to other cables in the area under analysis that could adversely affect safe shutdown.	E-5505, Worst Case 120VAC/125VDC Panel Appendix 'R'/Non Appendix 'R' Circuits Short Circtuit Levels, Rev. 004	
		HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.2.1
		HNP FSSPMD R21 00, Fire Safe Shutdown Program Manager Database, Rev. 021	Component, Cable, and Fault Tree Logic
		FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.3.1 and 9.3.4.
		E-5506, Appendix 'R' Coordination Study, Rev. 007	

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location.

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.5 *Circuit Failures Due to* Common Enclosure Concerns	*The common enclosure associated circuit concern deals with the possibility of causing secondary failures due to fire damage to a circuit either whose isolation* device fails to isolate the cable fault or protect the faulted cable from reaching its ignition temperature, or the fire somehow propagates along the cable into adjoining fire areas. The electrical circuit design for most plants provides proper circuit protection in the form of circuit breakers, fuses and other devices that are designed to isolate cable faults before ignition temperature is reached. Adequate electrical circuit protection and cable sizing are included as part of the original plant electrical design maintained as part of the design change process. Proper protection can be verified by review of as-built drawings and change documentation. Review the fire rated barrier and penetration designs that preclude the propagation of fire from one fire area to the next to demonstrate that adequate measures are in place to alleviate fire propagation concerns.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Common enclosure concerns are addressed in two fundamental ways. First, Calculation E-5506 evaluated all safe shutdown power supplies to ensure proper protective device coordination. Thus, any electrical fault should clear prior to starting a secondary fire. Should a fire occur, the design and installation criteria for cable and electrical penetrations ensures that the fire will not propagate beyond the fire area of concern. Thus, any fire would be limited to the "common enclosure" that is the fire area itself. The SSER 3 approval of the plant's common enclosure response is contained within the "Alternate Shutdown" review, but clearly applies to all plant fire areas.			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	SSER 3, page 9-15
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.3.1 and 9.3.4
				E-5506, Appendix 'R' Coordination Study, Rev. 007	
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.7.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location.

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

Fire Area Assessment. An engineering analysis shall be performed in accordance with the requirements of Section 2.3 for each fire area to determine the effects of fire or fire suppression activities on the ability to achieve the nuclear safety performance criteria of Section 1.5. [See Chapter 4 for methods of achieving these performance criteria (performance-based or deterministic).

NEI 00-01 Ref	NEI 00-01 Guidance
3.4 Fire Area Assessment and Compliance Assessment	By determining the location of each component and cable by fire area and using the cable to equipment relationships described above, the affected safe shutdown equipment in each fire area can be determined. Using the list of affected equipment in each fire area, the impacts to safe shutdown systems, paths and functions can be determined. Based on an assessment of the number and types of these impacts, the required safe shutdown path for each fire area can be determined. The specific impacts to the selected safe shutdown path can be evaluated using the circuit analysis and evaluation criteria contained in Section 3.5 of this document. Having identified all impacts to the required safe shutdown path in a particular fire area, this section provides guidance on the techniques available for individually mitigating the effects of each of the potential impacts.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	This introductory paragraph provides an overview of the compliance assessment process that was generally followed by Shearon Harris.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10.1

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1 Criteria / Assumptions	The following criteria and assumptions apply when performing fire area compliance assessment to mitigate the consequences of the circuit failures identified in the previous sections for the required safe shutdown path in each fire area.

Applicability	Comments
Applicable	Introductory information directing use of the suggested methodology.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Specific criteria are addressed in the sub-paragraph sections.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.1 [Number of Postulated Fires]	Assume only one fire in any single fire area at a time.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	A separate fire is not postulated to occur before, during, or following the fire in accordance with NUREG-0800.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section A.3.6
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.1

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.2 [Damage to Unprotected Equipment and Cables]	Assume that the fire may affect all unprotected cables and equipment within the fire area. This assumes that neither the fire size nor the fire intensity is known. This is conservative and bounds the exposure fire that is required by the regulation.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The analysis considers all potential failures in each area analyzed.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section A.3.11
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Sections 9.4.1

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.3 [Assess Impacts to Required Components]	Address all cable and equipment impacts affecting the required safe shutdown path in the fire area. All potential impacts within the fire area must be addressed. The focus of this section is to determine and assess the potential impacts to the required safe shutdown path selected for achieving post-fire safe shutdown and to assure that the required safe shutdown path for a given fire area is properly protected.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The use of the CAFTA fault tree tool does not require that all affected components be addressed. Components must be addressed until the fault tree shows success.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.4 [Manual Actions]	Use manual actions where appropriate to achieve and maintain post-fire safe shutdown conditions in accordance with NRC requirements.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	The specific criteria regarding what constitutes a feasible manual action, a previously approved manual action, and an acceptable *manual action are all under review within the* FAQ process and other industry and NRC initiatives. Manual action feasibility studies have been performed, and modifications have been implemented when necessary (to ensure adequate lighting, reflective equipment identification tags, etc.). The actions credited in the safe shutdown analysis were binned in accordance with FAQ 06-12, which was approved by the NRC in ML072340368, dated 1/24/08.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10.1.1, Appendices 16 and 16.1

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.5 [Repairs]	Where appropriate to achieve and maintain cold shutdown within 72 hours, use repairs to equipment required in support of post fire shutdown.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Repairs are considered recovery actions under NFPA 805. Currently, the analysis does not credit any cold shutdown repairs.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10.1.1, Item 4

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.6 [Assess Compliance with Deterministic Criteria]	Appendix R compliance requires that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage (III.G.1.a). When cables or equipment, including associated circuits, are within the same fire area outside primary containment and separation does not already exist, provide one of the following means of separation for the required safe shutdown path(s): - Separation of cables and equipment and associated non-safety circuits of redundant trains within the same fire area by a fire barrier having a 3-hour rating (III.G.2.a) - Separation of cables and equipment and associated non-safety circuits of redundant trains within the same fire area by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area (III.G.2.b). - Enclosure of cable and equipment and associated non-safety circuits of one redundant train within a fire area in a fire barrier having a one-hour rating. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area (III.G.2.c). For fire areas inside noninerted containments, the following additional options are also available: - Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards (III.G.2.d); - Installation of fire detectors and an automatic fire suppression system in the fire area (III.G.2.e); or - Separation of cables and equipment and associated non-safety circuits of redundant trains by a noncombustible radiant energy shield (III.G.2.f). Use exemptions, deviations and licensing change processes to satisfy the requirements mentioned above and to demonstrate equivalency depending upon the plant's license requirements.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The sections of Appendix R referenced in NEI 00-01 are mirrored in Sections C.5.b and C.7.a.(1)(b). The similar deterministic criteria of NFPA-805 are part of the acceptable compliance strategies used in the revalidation.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10.1.1

Table B-2 Nuclear Safety Capability Assessment

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.7 [Consider Additional Equipment]	Consider selecting other equipment that can perform the same safe shutdown function as the impacted equipment. In addressing this situation, each equipment impact, including spurious operations, is to be addressed in accordance with regulatory requirements and the NPP's current licensing basis.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	This consideration is not clearly stated but is inherent in performing safe shutdown analyses. Proof that this was considered is the inclusion of the Normal Service Water System as a credited system in the analysis during the revalidation.			EGR-NGGC-0102, Safe Shutdown/Fire Protection Review, Rev. 006	
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire *and Fire Hazards Analysis*, Rev. 2, 3/14/2008	

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.8 [Consider Instrument Tubing Effects]	Consider the effects of the fire on the density of the fluid in instrument tubing and any subsequent effects on instrument readings or signals associated with the protected safe shutdown path in evaluating post-fire safe shutdown capability. This can be done systematically or via procedures such as Emergency Operating Procedures.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Instrument tubing and its fire area routing is included in the FSSPMD. When necessary, it is treated in a manner similar to that in which cable damage is assessed.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.1.6
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.9 and Appendix 11

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.2 Methodology for Fire Area Assessment	Refer to Figure 3-5 for a flowchart illustrating the various steps involved in performing a fire area assessment. Use the following methodology to assess the impact to safe shutdown and demonstrate Appendix R compliance: [Refer to hard copy of NEI 00-01 for Figure 3-5]

Applicability	Comments
Applicable	Introductory Information.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Specific requirements are detailed in the sub-paragraphs.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.2.1 Identify the Affected Equipment by Fire Area	Identify the safe shutdown cables, equipment and systems located in each fire area that may be potentially damaged by the fire. Provide this information in a report format. The report may be sorted by fire area and by system in order to understand the impact to each safe shutdown path within each fire area (see Attachment 5 for an example of an Affected Equipment Report).

Applicability	Comments
Applicable	The FSSPMD provides the affected equipment report in a Division I / Division II format.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Affected equipment is sorted alpha-numerically by safe shutdown division.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10 and Appendix 18
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.2.2 Determine the Shutdown Paths Least Impacted By a Fire in Each Fire Area	Based on a review of the systems, equipment and cables within each fire area, determine which shutdown paths are either unaffected or least impacted by a postulated fire within the fire area. Typically, the safe shutdown path with the least number of cables and equipment in the fire area would be selected as the required safe shutdown path. Consider the circuit failure criteria and the possible mitigating strategies, however, in selecting the required safe shutdown path in a particular fire area. Review support systems as a part of this assessment since their availability will be important to the ability to achieve and maintain safe shutdown. For example, impacts to the electric power distribution system for a particular safe shutdown path could present a major impediment to using a particular path for safe shutdown. By identifying this early in the assessment process, an unnecessary amount of time is not spent assessing impacts to the frontline systems that will require this power to support their operation. Based on an assessment as described above, designate the required safe shutdown path(s) for the fire area. Identify all equipment not in the safe shutdown path whose spurious operation or mal-operation could affect the shutdown function. Include these cables in the shutdown function list. For each of the safe shutdown *cables (located in the fire area) that are part of the required safe shutdown path in the fire area, perform an evaluation to determine the impact of a fire-induced* cable failure on the corresponding safe shutdown equipment and, ultimately, on the required safe shutdown path. When evaluating the safe shutdown mode for a particular piece of equipment, it is important to consider the equipment's position for the specific safe shutdown scenario for the full duration of the shutdown scenario. It is possible for a piece of equipment to be in two different states depending on the shutdown scenario or the stage of shutdown within a particular shutdown scenario. Document information related to the normal and shutdown positions of equipment on the safe shutdown equipment list.

Applicability	Comments
Applicable	At Harris, the least affected "division" may be selected as a starting point since specific safe shutdown paths are not identified.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns with intent	Specific safe shutdown paths are not designated or identified. The least affected safe shutdown division is selected and the CAFTA fault tree and other information in the FSSPMD is used to develop the best overall safe shutdown strategy.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10, Appendix 18
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.2.3 Determine Safe Shutdown Equipment Impacts	Using the circuit analysis and evaluation criteria contained in Section 3.5 of this document, determine the equipment that can impact safe shutdown and that can potentially be impacted by a fire in the fire area, and what those possible impacts are.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	The FSSPMD Fault Tree Logic SSD Report provides *a list of equipment potentially affected* by the fire. The augmented CAFTA fault tree further displays the potential consequences of that potential damage. The Circuit Information Form from FSSPMD provides the FMEA for all cables assigned to the component, so the effects of the postulated fire damage can be readily determined.			HNP-E/ELEC-0001, Safe *Shutdown in Case of Fire* and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.2.4 Develop a Compliance Strategy or Disposition to Mitigate the Effects Due to Fire Damage to Each Required Component or Cable	The available deterministic methods for mitigating the effects of circuit failures are summarized as follows (see Figure 1-2): - Provide a qualified 3-fire rated barrier. - Provide a 1-hour fire rated barrier with automatic suppression and detection. - Provide separation of 20 feet or greater with automatic suppression and detection and demonstrate that there are no intervening combustibles within the 20 foot separation distance. - Reroute or relocate the circuit/equipment, or perform other modifications to resolve vulnerability. - Provide a procedural action in accordance with regulatory requirements. - Perform a cold shutdown repair in accordance with regulatory requirements. - Identify other equipment not affected by the fire capable of performing the same safe shutdown function. - Develop exemptions, deviations, Generic Letter 86-10 evaluation or fire protection design change evaluations with a licensing change process. Additional options are available for non-inerted containments as described in 10 CFR 50 Appendix R section III.G.2.d, e and f.

Applicability	Comments
Applicable	

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Compliance strategies are entered into the database as described in FIR-NGGC-0101.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10.1.1 and Appendix 18
				FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.2

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.2.5 Document the Compliance Strategy or Disposition Determined to Mitigate the Effects Due to Fire Damage to Each Required Component or Cable	Assign compliance strategy statements or codes to components or cables to identify the justification or mitigating actions proposed for achieving safe shutdown. The justification should address the cumulative effect of the actions relied upon by the licensee to mitigate a fire in the area. Provide each piece of safe shutdown equipment, equipment not in the path whose spurious operation or mal-operation could affect safe shutdown, and/or cable for the required safe shutdown path with a specific compliance strategy or disposition. Refer to Attachment 6 for an example of a Fire Area Assessment Report documenting each cable disposition.

Applicability	Comments
Applicable	In the CAFTA fault tree, basic events and gates are recovered until "success" is achieved. All affected equipment is not required to be addressed.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Resolution strategies are added to the augmented CAFTA fault tree until the fault tree indicates success and that it has been demonstrated that safe shutdown can be achieved.			FIR-NGGC-0101, Fire Protection Nuclear Safety Capability Assessment (NSCA), Rev. 0	Section 9.4.2
				HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Section B.10.1.1 and Appendix 18

Table B-2 Nuclear Safety Capability Assessment

Methodology Review

NFPA 805 Section: 2.4.2.4 Fire Area Assessment.

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.1.5 [C, Likelihood of Undesired Consequences]	Determination of the potential consequence of the damaged associated circuits is based on the examination of specific NPP piping and instrumentation diagrams (P&IDs) and review of components that could prevent operation or cause maloperation such as flow diversions, loss of coolant, or other scenarios that could significantly impair the NPP's ability to achieve and maintain hot shutdown. When considering the potential consequence of such failures, the [analyst] should also consider the time at which the prevented operation or maloperation occurs. Failures that impede hot shutdown within the first hour of the fire tend to be most risk significant in a first-order evaluation. Consideration of cold-shutdown circuits is deferred pending additional research.

Applicability	Comments
Applicable	This is additional guidance from RIS 2004-03 concerning the disposition and risk-significance of multiple spurious operations.

Alignment Statement	Alignment Basis	Comments	Unit	Reference Document	Doc. Details
Aligns	Shearon Harris performed a multiple spurious operations review in accordance with the guidelines of NRC RIS 2004-03. The results of the review are contained in Appendix 14 of the safe shutdown analysis.			HNP-E/ELEC-0001, Safe Shutdown in Case of Fire and Fire Hazards Analysis, Rev. 2, 3/14/2008	Appendix 14

SECURITY RELATED INFORMATION – WITHHOLD UNDER 10 CFR 2.390

Attachment C – NEI 04-02 Table B-3 – Fire Area Transition

163 Pages

SECURITY RELATED INFORMATION – WITHHOLD UNDER 10 CFR 2.390

Attachment D – NEI 04-02 Table F-1 Non-Power Operational Modes Transition

10 Pages

Attachment E – NEI 04-02 Table G-1 – Radioactive Release Transition

4 Pages

Table G-1 - Radioactive Release Transition Report

NFPA 805 Section 1.5.2 Radioactive Release Performance Criteria

Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, Limits.

Implementing Guidance Appendix G Step 1

Review pre-fire plans.
Ensure for locations that have the potential for contamination that specific steps are included for containment and monitoring of potentially contaminated fire suppression water. Update pre-fire plans as necessary.

Review

All HNP Fire Pre-plans were screened for applicability for this review. Those Fire Pre-plans that address areas where there is no possibility of radiological hazards were screened out from further review. The full list of Fire Pre-plans is provided below.

Unit Applicability 1

Comments

FPP-12-01-CNMT Containment - Review
FPP-12-02-RAB190-216 - Review
FPP-12-02-RAB236 - Review
FPP-12-02-RAB261 - Review
FPP-12-02-RAB286 - Review
FPP-12-02-RAB305-324 - Review
FPP-12-03-FHB - Review
FPP-12-04-DGB - Screen Out
FPP-12-05-DFOS - Screen Out
FPP-12-06-WPB - Review
FPP-12-07-TB - Review
FPP-12-08-SEC -Screen Out
FPP-12-09-LAF -Screen Out

FPP-012-03-FHB - Fuel Handling Building
General Building has floor drains which route to the floor drain tanks which are monitored. The normal FHB ventilation is monitored but not filtered. FHB Emergency Exhaust is monitored and filtered.

FPP-012-01-CNMT - Containment Building
General Building has floor drains which route to the floor drain tanks which are monitored. At power operations with containment integrity no release paths for water drainage or smoke can be established.

During non-power operations with containment open, fire fighting activities will not create any radiological release due to direct effects of the fire fighting suppression activities. The containment opening is bounded by existing controls to establish containment integrity.

FPP-012-02-RAB 190-216 - Reactor Auxiliary Building Elevations 190 and 216
General Building has floor drains which route to the floor drain tanks which are monitored. The normal RAB ventilation is monitored and filtered. RAB Emergency Exhaust is monitored and filtered.

FPP-012-02-RAB 236 - Reactor Auxiliary Building Elevation 236
General Building has floor drains which route to the floor drain tanks which are monitored. The normal RAB ventilation is monitored and filtered. RAB Emergency Exhaust is monitored and filtered.

FPP-012-02-RAB 261 - Reactor Auxiliary Building Elevation 261
General Building has floor drains which route to the floor drain tanks which are monitored. The normal RAB ventilation is monitored and filtered. RAB Emergency Exhaust is monitored and filtered.

FPP-012-02-RAB 286 - Reactor Auxiliary Building Elevation 286
General Building has floor drains which route to the floor drain tanks which are monitored. The normal RAB ventilation is monitored and filtered. RAB Emergency Exhaust is monitored and filtered.

FPP-012-02-RAB 305-324 - Reactor Auxiliary Building Elevations 305 and 324
General Building has floor drains which route to the floor drain tanks which are monitored. The normal RAB ventilation is monitored and filtered. RAB Emergency Exhaust is monitored and filtered.

FPP-012-06-WPB - Waste Processing Building
General Building has floor drains which route to the floor drain tanks which are monitored. The normal WPB ventilation is monitored and filtered. WPB does not have Emergency Exhaust.

FPP-012-07-TB - Turbine Building
General Building is open to the outdoors. Elevation 240 floor drains route to the monitored tanks in the WPB.

Table G-1 - Radioactive Release Transition Report

NFPA 805 Section 1.5.2 Radioactive Release Performance Criteria

Reference Document	Document Detail
FPP-012-01-CNMT, Containment Building Fire Pre-Plan, Rev. 3, 7/12/2005	
FPP-012-02-RAB190-216, Reactor Auxiliary Building Elevations 190 and 216 Fire Pre-Plan, Rev. 1, 6/15/2006	
FPP-012-02-RAB236, Reactor Auxiliary Building Elevation 236 Fire Pre-Plan, Rev. 1, 5/9/2006	
FPP-012-02-RAB261, Reactor Auxiliary Building Elevation 261 Fire Pre-Plan, Rev. 1, 5/7/2006	
FPP-012-02-RAB286, Reactor Auxiliary Building Elevation 286 Fire Pre-Plan, Rev. 0, 11/1/2004	
FPP-012-02-RAB305-324, Reactor Auxiliary Building Elevation 305 and 324 Fire Pre-Plan, Rev. 2, 5/7/2006	
FPP-012-03-FHB, Fuel Handling Building Fire Pre-Plan, Rev. 2, 7/12/2005	
FPP-012-04-DGB, Diesel Generator Building Fire Pre-Plan, Rev. 2, 4/11/2005	
FPP-012-05-DFOSB, Diesel Fuel Oil Storage Building Fire Pre-Plan, Rev. 2, 4/11/2005	
FPP-012-06-WPB, Waste Processing Building Fire Pre-Plan, Rev. 2, 5/5/2005	
FPP-012-07-TB, Turbine Building Fire Pre-Plan, Rev. 4, 6/8/2006	
FPP-012-08-SEC, Out Building Fire Pre-Plans, Rev. 4, 11/22/2005	
FPP-012-09-LAF, Large Area Fire Pre-Plan, Rev. 2, 6/8/2006	

Attachment E

Open Items Detail Report
(Table G-1 - Include in LAR/TR - All Items - Open Items)

Source	Reference	Open Item ID	Description	Date Entered	Disposition	Open/Closed
Table G-1	1.5.2 Radioactive Release Performance Criteria	9	Outside Areas " Need to develop a Fire Pre-Plan for outside Yard areas to address Radioactive Materials Areas (RMAs) and Sea-Land type container storage.	5/20/2008	The Outside Areas Fire Pre-Plan is revised to add cautions for the SIC regarding the potential for radioactive airborne and water surcharge in the event of certain fires associated with HP storage containers that may be located outdoors.	Open

Corrective Action Reference	
Include in LAR/TR	No
Change Eval / Modification Reference	
Supporting Detail	
Due Date	
Responsibility	Jimmy Nobles

Attachment F – Fire-Induced Multiple Spurious Operations – Resolution Methodology

As part of the NFPA 805 transition project, a comprehensive review and evaluation of HNP susceptibility to fire-induced multiple spurious operations (MSOs) was performed. The process was conducted in accordance with NEI 04-02 Revision 1 and Regulatory Guide 1.205 Revision 0, as supplemented by FAQ 07-0038 Revision 1 (draft May 2008).

Background

NEI 04-02 suggests that a licensee submit a summary of its approach for addressing potential fire-induced multiple spurious operations (MSOs) for NRC review and approval. As a minimum, NEI 04-02 suggest that the summary contain sufficient information relevant to methods, tools, and acceptance criteria used to enable the staff to determine the acceptability of the licensee's methodology. Progress Energy requests NRC approval of the process described below.

Methodology

The NRC staff has reviewed Revision 1 of NEI 00-01 and concluded that Chapter 3 provides an acceptable deterministic approach for analysis of post-fire safe shutdown circuits when applied in accordance with the regulatory expectations described in RIS 2005-30 and when used in conjunction with NFPA 805 and Regulatory Guide 1.205 for a plant that has transitioned to a 10 CFR 50.48(c) licensing basis (Reference: RIS 2005-30 and Regulatory Guide 1.205 Revision 0) In addition, an acceptable Fire PRA as defined in Regulatory Guide 1.205 Regulatory Position C.4.3 includes methods for the selection of cables and detailed circuit failure modes analysis, as well as the integration of these circuit failures into the overall Fire PRA (e.g., NUREG/CR-6850 Tasks 3, 9, 10, and 14).

The approach outlined in Figure F-1 below is one acceptable method to address fire-induced MSOs. This method uses insights from a Fire PRA that meets the requirements of Regulatory Guide 1.205, Revision 0.

This process is intended to be in support of transition to a new licensing basis. Post-transition changes would use the risk-informed, performance-based change process. The post-transition change process for the assessment of a specific MSO would be a simplified version of this process, and may not need the level of detail shown in the following section (e.g., An expert panel may not be necessary to identify and assess a new potential MSO. Identification of new potential MSOs may be part of the plant change review process and/or inspection process).

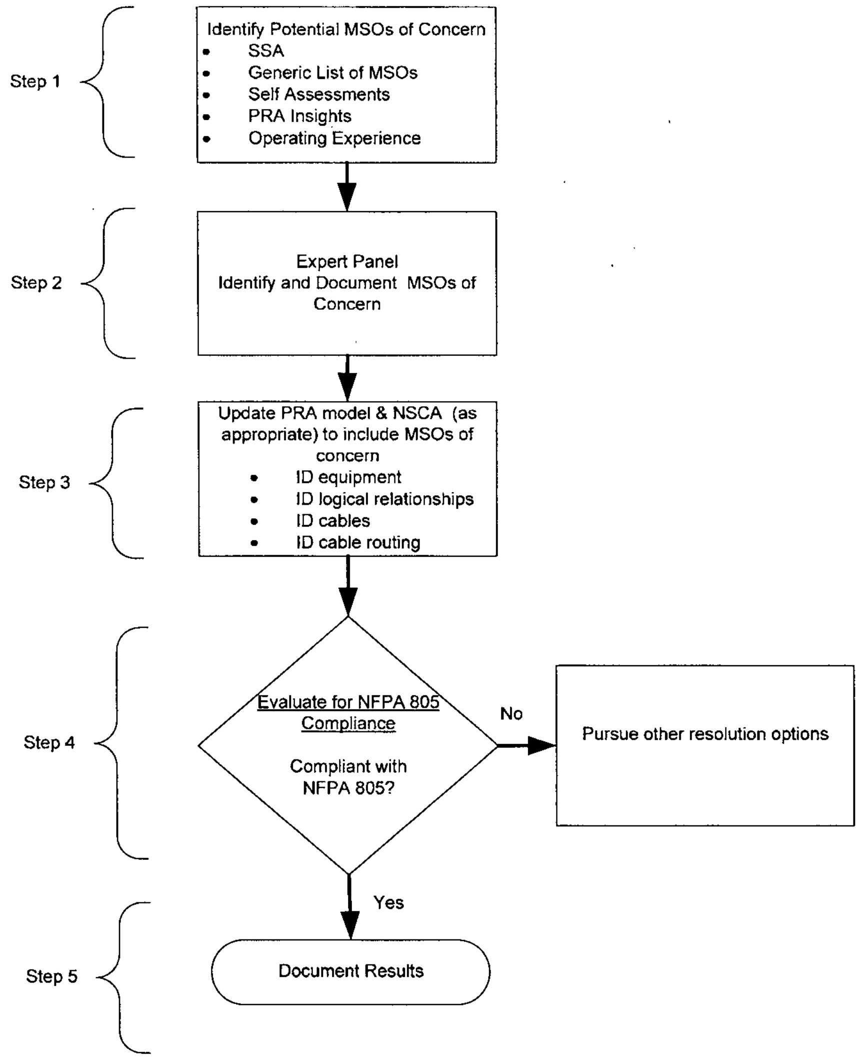

Figure F-1 – Multiple Spurious Operations – Transition Resolution Process

Table F-1 – FAQ 07-0038 Rev. 1 Summary Table

Guidance (NEI 04-02 FAQ 07-0038)	HNP Process/Results
Step 1 Identify potential MSOs of concern Information sources that may be used as input include: • Post-fire safe shutdown analysis (NEI 00-01, Revision 1, Chapter 3) • Generic lists of MSOs (e.g., from Owners Groups, if available.) • Self assessment results (e.g., NEI 04-06 assessments performed to addressed RIS 2004-03) • PRA insights (e.g., NEI 00-01 Revision 1, Appendix F) • Operating Experience (e.g., licensee event reports, NRC Inspection Findings, etc.)	The following was used as input to the overall assessment of MSOs at HNP: • Post-fire safe shutdown analysis (HNP-E/ELEC-0001) • PWROG generic list of MSOs • Miscellaneous operating experience • Fire PRA Cut-sets
Step 2 Conduct an expert panel to assess plant specific vulnerabilities (e.g., per NEI 00-01, Rev. 1 Section F.4.2). The expert panel should focus on system and component interactions that could impact nuclear safety. This information will be used in later tasks to identify cables and potential locations where vulnerabilities could exist. [Note: The physical location of the cables of concern (e.g., fire zone/area routing of the identified MSO cables), if known, may be used at this step in the process to focus the scope of the detailed review in further steps.]	An initial expert panel was conducted for HNP in 2005, prior to the development of Progress Energy Project Instruction FPIP-0122, Expert Panel Review of Multiple Spurious Operations. The results of this panel were integrated into the post-fire safe shutdown analysis (HNP-E/ELEC-0001, Appendix 14) and provided as input into the Fire PRA development (HNP-F/PSA-0077). A second expert panel was conducted in March 2008 using Revision 1 to FPIP-0122 (which accounted for industry changes since 2005 and to incorporate progress with PWROG MSO lists.).
Step 3 – Update the fire PRA model and NSCA to include the MSOs of concern. This includes the: • Identification of equipment (NUREG/CR-6850 Task 2) • Identification of cables that, if damaged by fire, could result in the spurious operation (NUREG/CR-6850 Task 3, Task 9) • Identify routing of the cables identified above. Include the equipment/cables of concern in the Nuclear Safety Capability Assessment (NSCA). Including the equipment and cable information in the NSCA does not necessarily imply that the interaction is possible since separation/protection may exist throughout the plant fire areas such that the interaction is not possible). Note: Instances may exist where update of the MSOs may not warrant update of the Fire PRA and NSCA analysis. For example, Fire PRA analysis in NUREG/CR-6850 Task 2, Component Selection, may determine that the particular interaction may not lead to core damage, or pre-existing equipment and cable routing information may determine that the particular MSO interaction is not physically possible. The rationale for exclusion of identified MSOs from the Fire PRA and NSCA should be documented and the configuration control mechanisms should be reviewed to provide reasonable confidence that the exclusion basis will remain valid.	Task 7.2 (NUREG/CR-6850 Task 2) of the HNP FPRA addressed spurious operations, including multiple spurious operations, identified in the post-fire safe shutdown analysis. These include those that resulted from the expert panel review. The results of the Fire PRA model update are included in Calculation HNP-F/PSA-0077, which includes: • Correlation of safe shutdown components and PRA basic events, • Correlation of PRA basic events and safe shutdown components • A listing of MSOs considered with documentation of their disposition The MSO combination components of concern were also evaluated as part of the NSCA (i.e., FSSPMD database and calculation HNP-E/ELEC-0001). For cases where the pre-transition MSO combination components did not meet the deterministic compliance, the MSO combination components were added to the scope of the risk-informed, performance-based change evaluations.

Table F-1 – FAQ 07-0038 Rev. 1 Summary Table

Guidance (NEI 04-02 FAQ 07-0038)	HNP Process/Results
Step 4 – Evaluate for NFPA 805 Compliance MSOs of concern should be included in the compliance assessment in the NSCA, consistent with the process for all NSCA components. The compliance assessment may use both deterministic and performance-based approaches. The performance-based approach may include the use of feasible and reliable recovery actions. During transition, if the recovery actions are deemed unallowed per the pre-transition licensing basis (Bin H for FAQ 06-0012), a risk-informed performance-based change evaluation may be used as potential means of demonstrating NFPA 805 compliance. Note that during the NFPA 805 transition, deterministic separation/protection is per the current licensing basis (10 CFR 50, Appendix R/NUREG-0800) with consideration of approved exemptions, etc. MSOs that meet the separation/protection requirements of the pre-transition licensing basis should be documented and the appropriate transition documentation updated as necessary. MSOs that are not in compliance with NFPA 805 will be reviewed for other resolution options, such as plant modifications.	The MSO combination components of concern were evaluated as part of the NSCA (i.e., FSSPMD database and calculation HNP-E/ELEC-0001). For cases where the pre-transition MSO combination components did not meet the deterministic compliance, the MSO combination components were added to the scope of the risk-informed, performance-based change evaluations. The process and results for change evaluations are summarized in Section 4.5 of the Transition Report.
Step 5 - Document Results The results of the process should be documented. High level methodology utilized as part of the transition process should be included in the 10 CFR 50.48(c) License Amendment Request/Transition Report. This table provides the guidance from FAQ 07-0038 and a summary/roadmap of the results and documents that contain the detailed assessments.	The results are documented in: • Change Package HNP-0033 (HNP Expert Panel) • HNP-E/ELEC-0001, Appendix 14 • Fire PRA development calculation HNP-F/PSA-0077 • Change Evaluation Calculations (contained in Fire Safety Analysis Calculations)

Attachment G – Operator Manual Actions – Transition to Recovery Actions

Background

NEI 04-02 FAQ 06-0012 was developed to provide clarification on allowable operator manual actions (OMAs) and to define the scope of the associated risk-informed, performance-based change evaluations. FAQ 06-0012 includes a binning process to determine if post-fire OMAs are allowed under the pre-transition licensing basis. FAQ 06-0012, Revision 5 was accepted by the NRC via closure memo dated January 24, 2008 (ML072340368). In addition, Section 4.2.4 of NFPA 805 requires that the additional risk presented by the use of recovery actions be evaluated.

OMAs and repairs have been transitioned as "recovery actions" in the new NFPA 805 licensing basis. Additional considerations from FAQ 06-0012 (ML072340368), and FAQ 06-0011 (ML080300121) were also included in the treatment of OMAs. FAQ 07-0030, under development at the time of the HNP LAR submittal, was discussed in concept with the NRC during pilot meetings held on December 12, 2007 and April 15-16, 2008.

NEI 04-02 suggests that a licensee submit a summary of its approach for addressing the transition of OMAs to recovery actions in the license amendment request (Regulatory Position C.2.2 and NEI-04-02, Rev. 1, Section 4.6). As a minimum, NEI 04-02 suggests that the assumptions, criteria, methodology, and overall results be included for the staff to determine the acceptability of the licensee's methodology. Progress Energy requests NRC approval of the process described below.

Operator Manual Action Transition Process

OMAs credited in the current safe shutdown analysis (SSA) were 'binned' in accordance with FAQ 06-0012.

The following process was utilized to make this determination:

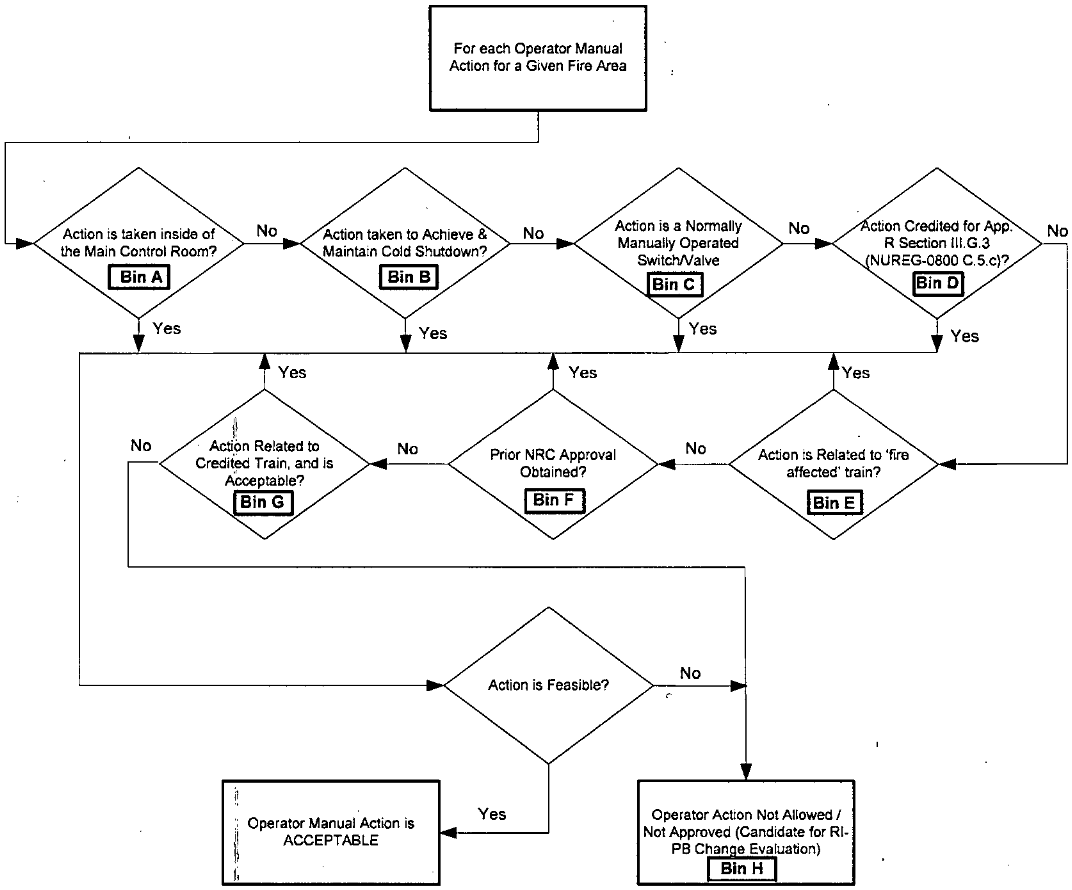

Figure G-1 – FAQ 06-0012 Operator Manual Action Binning Process (FAQ 06-0012 Rev. 5 – Figure B-4)

Bin H OMAs were identified as candidates for the risk-informed performance-based change evaluation process per NFPA 805 as part of the Nuclear Safety Performance Criteria Transition Review (refer to Transition Report Section 4.2 and the NEI 04-02 Table B-3 in Attachment C). Bin H OMAs were addressed in the NFPA 805 change evaluation process by evaluating the fire induced cables/component failures that caused the demand for the action. If the change evaluation determined that the action was not required to meet the acceptance criteria, the OMA can be characterized as a "defense-in-depth" action and, therefore, no longer credited for the nuclear safety capability assessment (NSCA). Defense in depths OMAs are evaluated to ensure there are no adverse impacts on NSCA performance.

OMAs to support implementation of alternative shutdown capability per C.5.c of NUREG-0800 are allowed (Bin D per FAQ 06-0012). Due to the unique circumstances associated with alternative shutdown capability, the OMAs will be retained as required by the NSCA.

A review was performed during the development of the Fire PRA for the remaining (Bins A, B, C, E, F, or G per FAQ 06-0012) OMAs and Control Room Actions (CRAs) to determine those actions that could have a negative impact on plant risk. Those that had a negative risk impact are being resolved during NFPA 805 implementation via an alternate strategy that eliminates the need for crediting the OMA for success in the NSCA. An allowed OMA not eliminated per this process is considered a "defense-in-depth" action using the insights from the PRA review. Any manual actions still required for NSCA success criteria are credited (e.g. not Defense in Depth), but, since they are allowed per NRC guidance, they do not require NRC approval.

As part of the Fire PRA development, OMAs were typically not included in the Fire PRA model unless needed for reduction in risk. Therefore, a detailed assessment of risk of performing these actions would not be necessary and the risk of the individual scenarios without credit for the action would bound the risk of crediting the action in the Fire PRA. OMAs and CRAs that were included in the internal events PRA model or were added to the internal events PRA as part of the Fire PRA development were addressed in accordance with the simplified Human Reliability Analysis (HRA) methodology developed by the Pilot Plants (see Flow Charts and methods presented to NRC in November and December 2007 and planned for inclusion in FAQ 07-0030, which was under development during the HNP LAR submittal (ADAMS Accession Nos. ML080220229 and ML073371166).

Those "defense-in-depth" OMAs that will continue to be used in the fire safe shutdown plant procedures will be demonstrated to meet the Progress Energy OMA feasibility criteria. The feasibility process ensures that operator manual actions (recovery actions) credited for success of performance goals in the NSCA are given priority over defense-in-depth actions.

Figure G-2 graphically depicts this process, which was used to categorize each pre-transition OMA and CRA as either:

- Credited for meeting the NSCA. NRC approval not required.
- Not Credited for meeting the NSCA, but credited as a defense-in-depth action. NRC approval not required.
- Credited for meeting the NSCA. NRC approval required.

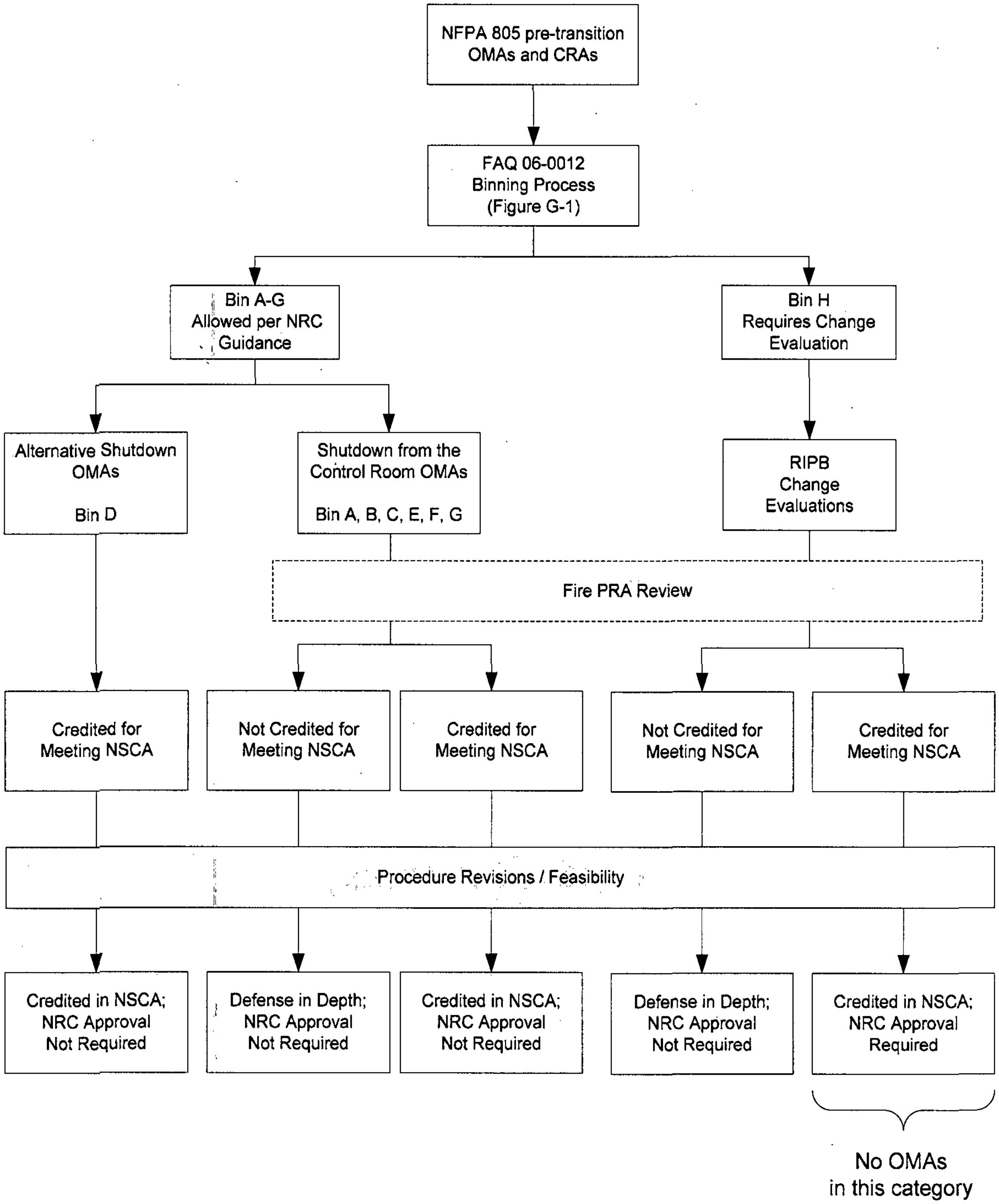
NFPA 805 pre-transition OMAs and CRAs
FAQ 06-0012 Binning Process (Figure G-1)
Bin A-G Allowed per NRC Guidance
Bin H Requires Change Evaluation
Alternative Shutdown OMAs
Bin D
Shutdown from the Control Room OMAs
Bin A, B, C, E, F, G
RIPB Change Evaluations
Fire PRA Review
Credited for Meeting NSCA
Not Credited for Meeting NSCA
Credited for Meeting NSCA
Not Credited for Meeting NSCA
Credited for Meeting NSCA
Procedure Revisions / Feasibility
Credited in NSCA; NRC Approval Not Required
Defense in Depth; NRC Approval Not Required
Credited in NSCA; NRC Approval Not Required
Defense in Depth; NRC Approval Not Required
Credited in NSCA; NRC Approval Required
No OMAs in this category

Figure G-2

Conclusions

HNP reviewed and documented pre-transition operator manual actions using the process from FAQ 06-0012. OMAs categorized as "Bin H" per FAQ 06-0012 were evaluated for acceptability using the risk-informed, performance-based change evaluation process. In addition, the additional risk associated with recovery actions per Section 4.2.4 of NFPA 805 was evaluated using the process summarized above. Final OMA procedures will be developed as part of the NFPA 805 implementation process.

Based on this process utilizing NFPA 805, no operator manual actions require approval by the NRC as part of this LAR/Transition Report.

Attachment H – NEI 04-02 Frequently Asked Question – Summary Table

The NRC staff has worked with NEI and two Pilot Plants (Oconee Nuclear Station and Harris Nuclear Plant) to define the licensing process for transitioning to a new licensing basis under 10 CFR 50.48(c) and NFPA 805. Both the NRC and the industry recognized the need for additional clarifications to the guidance provided in Regulatory Guide 1.205 and NEI 04-02, Revision 1. The NFPA 805 Frequently Asked Question (FAQ) Process was jointly developed by NEI and NRC to facilitate timely clarifications of NRC positions. This process is described in a letter from the NRC dated July 12, 2006, to NEI (ML061660105) and in RIS 2007-19 dated August 20, 2007.

Under the FAQ Process, transition issues were submitted to the NEI NFPA 805 Task Force for review, and subsequently presented to the NRC during public FAQ meetings. Once an acceptable FAQ was submitted to the NRC, the NRC staff issued a memorandum to file to indicate that the revised FAQ was acceptable. NEI 04-02 will be revised to incorporate the approved FAQs. Final closure will occur when Regulatory Guide 1.205, which endorses the new revision of NEI 04-02, is approved by the NRC.

Attachment H contains the FAQs that were used to clarify the guidance in Regulatory Guide 1.205 and NEI 04-02 and in the preparation of this License Amendment Request.

Table H-1 - NEI 04-02 FAQs – Status and Reference Table

No.	Rev.	Title	FAQ Ref.	FAQ NRC Comment Ref.	Technical Agreement	Closure Memo	Open FAQ Cross Ref.
06-0001		Alternate method for Engineering Evaluations	ML061440419	ML062060303	**WITHDRAWN** 12/14/06 ML063480169	**WITHDRAWN** 12/14/06 ML063480169	N/A
06-0002	2	NEI 04-02 Section 5.3.3 and App. I, Order of Questions for Change Analysis Screening	ML061440420 ML063170357 ML063350515	ML062060303	01/04/07 ML070030276	01/04/07 ML070030276	Note 1
06-0003	1b	Change Analysis Screening	ML061440422 ML063170355	ML062060303	01/04/07 ML070030242	01/04/07 ML070030242	Note 1
06-0004	0	Clarify NFPA 805 Chapter 4 and 3 relationship for 'required' FP systems/features	ML061440430	ML062060303 ML063350442			4.1, 4.2, 4.8.1
06-0005	2	Guidance on FPP-related changes	ML062350095 ML063180544 ML072820015	ML072400021 ML073060462			4.5
06-0006	2	High-low pressure interface definition and NEI 00-01/NFPA 805 discrepancies	ML062350109 ML063170360 ML063540308	ML062890268 ML070660071	03/12/07 ML070030117	03/12/07 ML070030117	Note 1
06-0007	3	NFPA 805 Chapter 3 Requirements for Fire Brigades	ML062350121 ML070030325 ML070510442 ML071550408	ML063170365 ML071380338	6/21/07 ML071940375	11/13/07 ML072560733	4.2
06-0008	8	Alternate method for Engineering Evaluations	ML062860250 ML070510499 ML070800007 ML071020160 ML071020169 ML071080099 ML071340180 ML072820016 ML073370025	ML063350442 ML070640544 ML071380177 ML071380182 ML072050214 ML072740231 ML073370775	1/24/08 ML080430163		5.1, Att. M, Att. P

Table H-1 - NEI 04-02 FAQs – Status and Reference Table

No.	Rev.	Title	FAQ Ref.	FAQ NRC Comment Ref.	Technical Agreement	Closure Memo	Open FAQ Cross Ref.
06-0011	2	Clarify III.G.3 Compliance Transition	ML062890271 ML070510505 ML072740248	ML063350442 ML072400023	10/18/07 ML073200763	3/04/08 ML080300121	Note 1
06-0012	5	Clarify Manual Action Transition in Appendix B	ML062860255 ML063170362 ML070850610 ML071380229 ML071570260 ML073320028	ML063350442 ML071380186 ML072820170 ML072820168	6/21/07 ML071940375 11/29/07 ML073400502	1/24/08 ML072340368	Att. B
06-0016	1	Ignition Source counting guidance for Electrical Cabinets	ML070030348 ML071020174	ML070640555	5/17/07 ML071510425	10/05/07 ML072700475	Note 1
06-0017	2	Ignition Source counting guidance for High Energy Arcing Faults (HEAF)	ML070030383 ML071350432 ML071570255	ML071730038	6/21/07 ML071940375	9/26/07 ML072500300	Note 1
06-0018	1	Ignition Source counting guidance for Main Control Board (MCB)	ML070030427 ML071020181	ML070640562	5/17/07 ML071510425	9/7/07 ML072500273	Note 1
06-0019	4	Define "power block" and "plant"	ML070030437 ML071340184 ML072550063 ML072740255 ML073060545	ML070510365 ML073060471	11/15/07 ML073200936	3/05/08 ML080510224	Note 1
06-0020	1	Definition of "applicable"	ML070030443 ML071340188	ML070510369	5/17/07 ML071510425	11/28/07 ML072420286	Note 1
06-0021	1a	Clarify that air drops are acceptable.	ML070030457 ML071340192	ML070510417	5/17/07 ML071510425	11/13/07 ML072420306	Note 1
06-0022	2	Identify a list of typical flame propagation tests which are considered acceptable.	ML070030459 ML072340055	ML072050222 ML072740236			Att. A
06-0023		Grant exception for Diesel Generator Day Tanks located within Diesel Generator Buildings.	ML070030470		**WITHDRAWN** 5/17/07 ML071510425	10/3/07 ML072700552	N/A

Table H-1 - NEI 04-02 FAQs – Status and Reference Table

No.	Rev.	Title	FAQ Ref.	FAQ NRC Comment Ref.	Technical Agreement	Closure Memo	Open FAQ Cross Ref.
06-0024	1	Define what "adequate clearance" is.	ML070030472 ML072340062	ML071380189	8/23/07 ML072550213	10/16/07 ML072740225	Note 1
06-0025	5	Define minimum acceptable pre-plan scope.	ML070030476 ML071340194 ML073400147 ML073510082 ML073550021	ML070300588 ML073510074	7/19/07 ML072080246		Att. A
06-0026		Clarify NFPA code requirements for gear maintenance	ML070030480	ML071380194	**WITHDRAWN** 5/17/07 ML071510425	**WITHDRAWN** 10/15/07 ML072560564	N/A
06-0027	0	Clarify the "where provided" statement.	ML071380236		10/18/07 ML073200763		Att. A
06-0028	2	Clarify intent of "familiarization with plant fire prevention procedures, fire reporting, and plant emergency alarms" regarding scope of or depth of the training.	ML070030489 ML071340195 ML071550415	ML070510427 ML071380349	6/21/07 ML071940375	10/17/07 ML072740233	Note 1
07-0031	0	Misc Binning Issues	ML071380238	ML072880327 ML073060480	11/29/07 ML073400502	12/17/07 ML072840658	Note 1
07-0032	2	10CFR 50.48(a) and GDC 3 clarification	ML071930378 ML080700411 ML081300697	ML073060492 ML081300689			5.1
07-0033	1	Review of Existing Engineering Equivalency Evaluations	ML071930379 ML073550023	ML072700037	2/21/08 ML080730007		4.2.2.2, Att. J
07-0035	0	Bus Duct counting guidance for High Energy Arcing Faults	ML071650151	ML073540262			
07-0036	1	Define compliance categories for NEI 04-02 Table B-1	ML072320155 ML073550025	ML072700038	2/21/08 ML080730007		4.1, Att. A
07-0038	0	Lessons learned for MSOs	ML072740262	ML073060506			4.8.2.1, Att. F

Table H-1 - NEI 04-02 FAQs – Status and Reference Table

No.	Rev.	Title	FAQ Ref.	FAQ NRC Comment Ref.	Technical Agreement	Closure Memo	Open FAQ Cross Ref.
07-0039	1	Provide update of NEI 04-02 B-2 and B-3 Processes	ML072740268 ML080910136	ML073330556			4.2, Att. B, Att. C
07-0040	1	Clarification on Non-Power Operations	ML073060550 ML080720027	ML073170227 ML081150739			4.3, Att. D
07-0041	0	Chapter 3 Codes and Standards	**WITHDRAWN** ML073310447				4.1, Att. A
07-0042	0	Vented Cabinets		1/9/08 ML080230438			
08-0047	0	Spurious Operation Probability	ML081200126				

Note 1 – *These FAQs are closed by the issuance of an NRC closure memo.*

Attachment I – Definition of Power Block

For the purposes of establishing the structures included in the HNP fire protection program in accordance with 10 CFR 50.48(c) and NFPA 805, the following plant structures are considered to be part of the 'power block'. The following table provides a listing of power block structures as described in FAQ 06-0019, Define Power Block (ML080510224).

Power block equipment includes all the SSCs required for the safe and reliable operation of the station. It includes all safety-related and balance-of-plant systems and components required for the operation of the station, including radioactive waste processing and storage and switchyard equipment maintained by the station. Systems, structures, or components required to maintain federal or state regulatory compliance are included in this grouping. This equipment does not include buildings or structures that support station staff, such as offices or storage structures, or the HVAC and support systems focused only on habitability of those structures.

The power block is the group of buildings composed of:

Table I-1 HNP Power Block Definition

Building	Comments
Turbine Building	
Reactor Auxiliary Building	
Waste Processing Building	
Fuel Handling Building	Excluding Areas "K" and "M"
Diesel Generating Building	
Diesel Fuel Oil Storage Building	
Intake Structures	Emergency Service Water Intake Structure
	Emergency Service Water Intake Screening Structure
Reactor Containment Building	

SECURITY RELATED INFORMATION – WITHHOLD UNDER 10 CFR 2.390

Attachment J – Existing Engineering Equivalency Evaluation Transition

38 Pages

SECURITY RELATED INFORMATION – WITHHOLD UNDER 10 CFR 2.390

Attachment K – Existing Licensing Action Transition

69 Pages

Attachment L – NFPA 805 Chapter 3 Requirements for Approval

NFPA 805 Section 3.5.16

NFPA 805 Section 3.5.16 states:

> *"The fire protection water supply system shall be dedicated for fire protection use only.*
>
> *Exception No. 1: Fire protection water supply systems shall be permitted to be used to provide backup to nuclear safety systems, provided the fire protection water supply systems are designed and maintained to deliver the combined fire and nuclear safety flow demands for the duration specified by the applicable analysis.*
>
> *Exception No. 2: Fire protection water storage can be provided by plant systems serving other functions, provided the storage has a dedicated capacity capable of providing the maximum fire protection demand for the specified duration as determined in this section."*

The fire protection system water supply is not permanently connected to any non-fire-related purposes, except to provide makeup to the Non-Essential Services Chilled Water Expansion Tank. However, under normal conditions there would be no water going into this system as makeup. Maximum demand is approximately 10 gpm, which is within the jockey fire pump capacity. Additionally, the Unit SCO may approve use of fire protection water for plant evolutions under the following conditions: Unit SCO approval is obtained and documented. Controls/communications are in place to ensure the fire protection water demand can be secured immediately if a fire occurs. The fire protection demand must be less than 250 gpm (flow through one 2-1/2 inch hose) which is below the allowance for hose streams thereby having no adverse impact on automatic suppression functions in any plant location.

NRC approval is requested for approval of the permanent connection to makeup to the Non-Essential Services Chilled Water Expansion Tank for 10 gpm and for the temporary use of the Fire Protection water supply with the following restrictions: Unit SCO approval is obtained and documented. Controls/communications are in place to ensure the fire protection water demand can be secured immediately if a fire occurs. The fire protection demand must be less than 250 gpm (flow through one 2-1/2 inch hose) which is below the allowance for hose streams thereby having no adverse impact on automatic suppression functions in any plant location.

NFPA 805 Section 3.6.5

NFPA 805 Section 3.6.5 states:

> *"Where the seismic required hose stations are cross-connected to essential seismic non-fire protection water supply systems, the fire flow shall not degrade the essential water system requirement."*

FSAR Section 9.5.1 states "The system cross connections are capable of supplying two 75 gpm hose stations. Piping between these valves and the emergency service water system are designed as ASME Safety Class 3."

Based on calculation SW-0087, cross tie ESW to FP will result in ESW being inoperable. Procedure OP-139 required one train of the ESW system to be declared inoperable if cross tied to the FP seismic hose stand pipe. In post SSE conditions, if a fire occurs, the cross tie would be performed, ESW would be declared inoperable, and the fire extinguished as soon as possible. The cross tie would then be removed to restore ESW.

NRC approval is requested for one train of ESW being declared inoperable during a fire due to cross tie ESW to FP until the fire is extinguished as specified in plant procedures. This is to maintain the current plant operational flexibility.

Attachment M - License Condition Changes

Replace the current HNP fire protection license condition 2.F with the standard license condition in Regulatory Position C.3.1 of Regulatory Guide 1.205, Revision 0, as modified by FAQ 06-0008, as shown below. In support of this change, HNP has developed a fire Probabilistic Risk Assessment (PRA) which has been reviewed and been found acceptable by the NRC during the course of its observation of HNP's transition to NFPA 805 as a Pilot Plant. Outstanding high level findings from the NRC's pilot observations of the Fire PRA are included in Attachment Q.

==

Carolina Power & Light shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c) as specified in the license amendment request dated May 31, 2008 and as approved in the safety evaluation report dated __________ (and supplements dated ___________). The licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a), 10 CFR 50.48(c), and the following:

(a) Prior NRC review and approval is not required for a change that results in a net decrease in risk for both CDF and LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the change evaluation.

(b) Prior NRC review and approval is not required if the change results in a net calculated risk increase less than 1E-7/yr for CDF and less than 1E-8/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the change evaluation. Change reports need not be submitted to the NRC for these changes.

(c) Where the calculated plant change risk increase is < 1E-6/yr, but ≥1E-7/yr for CDF or < 1E-7/yr, but ≥1E-8/yr for LERF, the licensee must submit a summary description of the change to the NRC following completion of the change evaluation. The proposed change also must be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. If the NRC does not object to the change within 90 days, the licensee may proceed with implementation of the proposed change.

Carolina Power & Light may perform change evaluations for deviations from the codes, standards, and listings referenced in NFPA 805, without a 10 CFR 50.90 submittal, as long as the specific requirement for the feature is not included in NFPA 805 Chapter 3, and the NFPA 805 change process is used.

Supersede the following license condition 2.F:

"F. Fire Protection Program

Carolina Power & Light Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility as amended and as approved in the Safety Evaluation Report (SER) dated November 1983 (and supplements 1 through 4), and the Safety Evaluation dated January 12, 1987, subject to the following provision below. The licensees may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire."

The Nuclear Regulatory Commission issued Amendment No. 123 – Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment on use of fire-resistive cable (TAC No. MC8134) that authorized the use of fire-resistive cables in lieu of the alternatives specified in Section C5.b.2 of Branch Technical Position Chemical Engineering Branch 9.5-1 (NUREG-0800). The NRC staff approved the deviation from fire protection program requirements in License Condition 2.F because HNP adequately demonstrated that the protection provided by the ire-resistive cable, in the specific application of Meggit Safety Systems cables for control circuits, associated with the volume control tank outlet valves is equivalent to the protection provided by a 3-hour rated fire barrier system.

It is HNP's understanding that implicit in the superseding of this license condition, all prior fire protection program SERs and commitments have been superseded in their entirety by the revised license condition.

No other license conditions need to be superseded or revised.

HNP implemented the following process for determining that these are the only license conditions required to be either revised or superseded to implement the new fire protection program which meets the requirements in 10 CFR 50.48(a) and 50.48(c):

- A review was conducted of the HNP Facility Operating License NPF-63, Amendment 126, by HNP licensing staff and Progress Energy fire protection staff. The review was performed by reading the Operating License and performing electronic searches. Outstanding License Amendment Requests that have been submitted to the NRC were also reviewed for potential impact on the license conditions.

Attachment N – Technical Specification Changes

Supersede the following Technical Specifications:

- Section 6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

  h. Fire protection program implementation.

The Bases for Technical Specification 3/4.3.3.5 states:

> "This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR Part 50".

The bases for Technical Specification 3/4.3.3.5 should be revised to state:

> "This capability is consistent with General Design Criterion 3, 10 CFR 50.48(a) and 10 CFR 50.48(c)."

No other Technical Specifications or Bases need to be revised.

HNP implemented the following process for determining that these are the only Technical Specifications required to be either revised or superseded to implement the new fire protection program which meets the requirements in 10 CFR 50.48(a) and 10 CFR 50.48(c).

- A review was conducted of the HNP Technical Specifications by HNP licensing staff and Progress Energy fire protection staff. The review was performed by reading the Technical Specifications and performing electronic searches. Outstanding Technical Specification changes that have been submitted to the NRC were also reviewed for potential impact on the license conditions.

HNP determined that these changes to the Technical Specifications are adequate for HNP's adoption of the new fire protection licensing basis, for the following reasons.

- The requirement for establishing, implementing, and maintaining fire protection procedures is now contained in the regulation (10 CFR 50.48(a) and 10 CFR 50.48(c) NFPA 805 Chapter 3).
- 10 CFR 50, Appendix R is no longer an appropriate basis for the HNP fire protection program. 10 CFR 50.48(a) and 10 CFR 50.48(c) are appropriate references.

Attachment N-1

Proposed Technical Specification Change (Mark-Up)

3 Pages

PROCEDURES AND PROGRAMS (Continued)

g. Quality Assurance Program for effluent and environmental monitoring; and

h. ~~Fire protection program implementation.~~ Deleted. |

i. Technical Specification Equipment List Program.

6.8.2 DELETED

6.8.3 DELETED

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage, to as low as practical levels, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The systems include:

1. Residual Heat Removal System and Containment Spray System, except spray additive subsystem and RWST,

2. Safety Injection System, except boron injection recirculation subsystem and accumulator,

3. Portions of the Chemical and Volume Control System:

   a. Letdown subsystem, including demineralizers,
   b. Boron re-cycle holdup tanks, and
   c. Charging/safety injection pumps,

4. Post-Accident Sample System (until such time as a modification eliminates the Post-Accident Sample System as a potential leakage path),

3/4.3.3.2 MOVABLE INCORE DETECTORS - DELETED

3/4.3.3.3 DELETED

3/4.3.3.4 DELETED

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, control, and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor.

REMOTE SHUTDOWN SYSTEM (Continued)

This capability is consistent with General Design Criterion 3 ~~and Appendix R to 10 CFR Part 50~~ 10 CFR 50.48(a) and 10CFR50.48(c).

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. The RVLIS and In Core Thermocouple design meets the intent of Regulatory Guide 1.97. The HNP design (and Regulatory Guide 1.97) stipulates redundancy for RVLIS and In Core Thermocouples. A fully 100% functional channel would be available should a channel fail.

The RVLIS and In Core Thermocouple systems do not automatically actuate any component. These monitoring systems are used for indication only. Diverse monitoring is available for core cooling indication requirements such as Reactor Coolant Hot and Cold Leg temperature indications as well as Reactor Coolant System pressure.

The thirty-day completion time for one inoperable channel of RVLIS or In Core Thermocouples is based on operating experience and takes into account the remaining OPERABLE channel, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring an instrument during this interval. If the thirty-day completion time was not met, then a written report to the NRC would be required to outline the preplanned alternate method of monitoring (in this case the other redundant channel would be available), the cause of the inoperability, and plans and a schedule for restoring the instrumentation channels of the Function to operable status.

If both channels of RVLIS or In Core Thermocouples are inoperable, then restore an inoperable channel within 7 days. The completion time of 7 days is based on the relatively low probability of an event requiring RVLIS and In Core Thermocouple instrumentation operation and the availability of alternate means to obtain the required information. Diverse monitoring is available for core cooling indication requirements such as Reactor Coolant Hot and Cold Leg temperature indications as well as Reactor Coolant System pressure. These parameters can be used to manually determine subcooling margin, which normally uses core exit temperatures.

3/4.3.3.7 DELETED

3/4.3.3.8 DELETED

3/4.3.3.9 DELETED

3/4.3.3.10 DELETED

Attachment N-2

Retyped Technical Specification Page

3 Pages

PROCEDURES AND PROGRAMS (Continued)

g. Quality Assurance Program for effluent and environmental monitoring; and

h. Deleted.

i. Technical Specification Equipment List Program.

6.8.2 DELETED

6.8.3 DELETED

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage, to as low as practical levels, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The systems include:

1. Residual Heat Removal System and Containment Spray System, except spray additive subsystem and RWST,

2. Safety Injection System, except boron injection recirculation subsystem and accumulator,

3. Portions of the Chemical and Volume Control System:

   a. Letdown subsystem, including demineralizers,
   b. Boron re-cycle holdup tanks, and
   c. Charging/safety injection pumps,

4. Post-Accident Sample System (until such time as a modification eliminates the Post-Accident Sample System as a potential leakage path),

3/4.3.3.2 MOVABLE INCORE DETECTORS - DELETED

3/4.3.3.3 DELETED

3/4.3.3.4 DELETED

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, control, and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor.

REMOTE SHUTDOWN SYSTEM (Continued)

This capability is consistent with General Design Criterion 3, 10 CFR 50.48(a) and 10 CFR 50.48(c).

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. The RVLIS and In Core Thermocouple design meets the intent of Regulatory Guide 1.97. The HNP design (and Regulatory Guide 1.97) stipulates redundancy for RVLIS and In Core Thermocouples. A fully 100% functional channel would be available should a channel fail.

The RVLIS and In Core Thermocouple systems do not automatically actuate any component. These monitoring systems are used for indication only. Diverse monitoring is available for core cooling indication requirements such as Reactor Coolant Hot and Cold Leg temperature indications as well as Reactor Coolant System pressure.

The thirty-day completion time for one inoperable channel of RVLIS or In Core Thermocouples is based on operating experience and takes into account the remaining OPERABLE channel, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring an instrument during this interval. If the thirty-day completion time was not met, then a written report to the NRC would be required to outline the preplanned alternate method of monitoring (in this case the other redundant channel would be available), the cause of the inoperability, and plans and a schedule for restoring the instrumentation channels of the Function to operable status.

If both channels of RVLIS or In Core Thermocouples are inoperable, then restore an inoperable channel within 7 days. The completion time of 7 days is based on the relatively low probability of an event requiring RVLIS and In Core Thermocouple instrumentation operation and the availability of alternate means to obtain the required information. Diverse monitoring is available for core cooling indication requirements such as Reactor Coolant Hot and Cold Leg temperature indications as well as Reactor Coolant System pressure. These parameters can be used to manually determine subcooling margin, which normally uses core exit temperatures.

3/4.3.3.7 DELETED

3/4.3.3.8 DELETED

3/4.3.3.9 DELETED

3/4.3.3.10 DELETED

Attachment O – Orders and Exemptions

No Orders or Exemptions need to be superseded or revised. HNP implemented the following process for making this determination:

- A review was conducted of the HNP docketed correspondence by HNP licensing staff and Progress Energy fire protection staff. The review was performed by reviewing the correspondence files and performing electronic searches of internal HNP records and the NRC's ADAMS document system.

A specific review was performed of the license amendment that incorporated the mitigation strategies required by Section B.5.b of Commission Order EA-02-026 (TAC No. MD4537) to ensure that any changes being made to ensure compliance with 10 CFR 50.48(c) do not invalidate existing commitments applicable to the plant. The review of this order demonstrated that changes to the fire protection program will not affect measures required by B.5.b.

Attachment P – Performance-Based Methods - NFPA 805 Chapter 3 – 10 CFR 50.48.(c)(2)(vii)

In accordance with 10 CFR 50.48(c)(2)(vii), Carolina Power & Light requests approval of the following performance-based method for the use of performance-based methods for specific sections of NFPA 805 Chapter 3. This method is based upon the process developed under FAQ 06-0008, Revision 5 (ML071340180).

Method:

All fire protection systems and features required by NFPA 805 Chapter 3 will continue to be required (unless specifically addressed separately from this process in an LAR). Secondary features (See Table below) may be changed based on an evaluation, using the required methods in a similar manner that was previously allowed under the Generic Letter 86-10 license condition, without prior NRC approval.

Specifically, the method applies to sections of NFPA 805 Chapter 3 containing referenced codes, standards, and listings. Note the 'method' applies to the secondary features of the referenced codes, standards, and listings contained within these sections, and the process cannot be used to change the NFPA 805 Chapter 3 specific requirements.

Each individual change will be evaluated using the NFPA 805 change process (NFPA 805 performance goals/objectives/criteria, defense-in-depth and safety margins evaluation).

Certain fire protection systems and features have performance requirements that are conditional upon NFPA 805 Chapter 4 requirements. These systems and features are:

- Fire Alarm and Detection Systems [NFPA 805 Section 3.8]
- Automatic and Manual Water-Based Fire Suppression Systems [NFPA 805 Section 3.9]
- Gaseous Fire Suppression Systems [NFPA 805 Section 3.10]
- Passive Fire Protection Features [NFPA 805 Section 3.11]

For these systems and features, the performance requirements are established by the deterministic and/or performance-based analyses used in demonstrating how the NFPA 805 Chapter 4 performance criteria are met. Fire Protection Engineering Analyses may be used to demonstrate how these systems and features meet the NFPA 805 Chapter 4 criteria (e.g., coverage/performance of a detection/suppression system, ability of fire barriers to withstand expected fire hazards, etc.). These Fire Protection Engineering Analyses, however, are allowed under 10 CFR 50.48(c) and do not require specific permission under 10 CFR 50.48(c)(2)(vii), "Performance-Based Methods".

This method does not apply to NFPA 805 Chapter 3 changes that do not relate to referenced codes, standards, or listings. These types of changes continue to require individual 10 CFR 50.90 license amendment requests addressing the specific deviation.

The following provides the sections of NFPA 805 that will utilize this method.

Column Heading Definition:

Fire Protection Engineering Analysis Process Applicable: Sections of NFPA 805 Chapter 3 containing referenced codes and listings. Note the "Applicability" would only apply to the referenced codes, standards, and listings contained within these sections, and the process could not be used to change the NFPA 805 Chapter 3 specific requirements.

Fire Protection Engineering Analysis Process Not Applicable: These NFPA 805 Chapter 3 sections do not have referenced codes, standards, or listings. Therefore, the method associated with this FAQ is not applicable and would be outside the scope of the associated LAR.

Table P-1 Performance-Based Methods - NFPA 805 Chapter 3

Section	Title	FP Eng. Analysis Process Applicable	FP Eng. Analysis Process Not Applicable	Referenced Code/Standard/Listing
3.1	General		X	
3.2	Fire Protection Plan		X	
3.2.1	Intent		X	
3.2.2	Management Policy Direction and Responsibility		X	
3.2.3	Procedures		X	
3.3	Prevention		X	
3.3.1	Fire Prevention for Operational Activities	X		3.3.1.2 (2) NFPA 701 (5) NFPA 30 (6) "applicable NFPA codes and standards" 3.3.1.2.1 NFPA 51B NFPA 241
3.3.2	Structural	X		3.3.2 NFPA 220
3.3.3	Interior Finishes	X		3.3.3 NFPA 101
3.3.4	Insulation Materials		X	
3.3.5	Electrical	X		3.3.5.1 …electrical wiring shall be listed for plenum use.." (Note 1)
3.3.6	Roofs	X		NFPA 256
3.3.7	Bulk Flammable Gas Storage	X		3.3.7.1 NFPA 50A

Table P-1 *Performance-Based Methods* - NFPA 805 Chapter 3

Section	Title	FP Eng. Analysis Process Applicable	FP Eng. Analysis Process Not Applicable	Referenced Code/Standard/Listing
3.3.8	Bulk Storage of Flammable and Combustible Liquids	X		NFPA 30
3.3.9	Transformers		X	
3.3.10	Hot Pipes and Surfaces		X	
3.3.11	Electrical Equipment		X	
3.3.12	Reactor Coolant Pumps		X	
3.4	Industrial Fire Brigade	See sub-sections		
3.4.1	On-Site Fire Fighting Capability	X		(a)(1), (2), and (3) NFPA 600 NFPA 1500 NFPA 1582
3.4.2	Fire Pre-Plans		X	
3.4.3	Training and Drills	X		(a)(1) NFPA 600 NFPA 1500
3.4.4	Fire Fighting Equipment	X		"....with the applicable NFPA standards."
3.4.5	Off-Site Fire Department Interface		X	
3.4.6	Communications		X	

Table P-1 Performance-Based Methods - NFPA 805 Chapter 3

Section	Title	FP Eng. Analysis Process Applicable	FP Eng. Analysis Process Not Applicable	Referenced Code/Standard/Listing
3.5	Water Supply	X		3.5.1(b) NFPA 13 NFPA 15 3.5.2 NFPA 22 3.5.3 NFPA 20 3.5.10 NFPA 24 3.5.13 ANSI B31.1 3.5.15 NFPA 24
3.6	Standpipe and Hose Stations	X		3.5.1 NFPA 14 3.5.3 "Listed electrically safe fixed fog nozzles…"
3.7	Fire Extinguishers	X		NFPA 10
3.8	Fire Alarm and Detection Systems	See sub-sections		
3.8.1	Fire Alarm	X		NFPA 72
3.8.2	Detection	X		NFPA 72
3.9	Automatic and Manual Water-Based Fire Suppression Systems	X		3.9.1 NFPA 13 NFPA 15 NFPA 750 NFPA 16
3.10.	Gaseous Fire Suppression Systems	X		3.10.1 NFPA 12 NFPA 12A NFPA 2001
3.11	Passive Fire Protection Features	See sub-sections		

Table P-1 Performance-Based Methods - NFPA 805 Chapter 3

Section	Title	FP Eng. Analysis Process Applicable	FP Eng. Analysis Process Not Applicable	Referenced Code/Standard/Listing
3.11.1	Building Separation (Note 2)	X		NFPA 80A
3.11.2	Fire Barriers	X		NFPA 251 ASTM E 119
3.11.3	Fire Barrier Penetrations	X		"…listed fire-rated door assemblies or *listed fire rated fire dampers*…" (1) NFPA 80 (2) NFPA 90A (3) NFPA 101
3.11.4	Through Penetration Fire Stops	X		"….with a fire test protocol acceptable to the AHJ or be protected by a listed fire-rated device…."
3.11.5	Electrical Raceway Fire Barrier Systems (ERFBS)		X	(Note 3)

Note 1 – Flame propagation tests/standards for electrical cable construction are addressed by FAQ 06-0022.

Note 2 – Section 3.11.1 of NFPA 805 also contains an exception for performance-based analysis.

Note 3 – Generic Letter 86-10, Supplement 1 is not considered a referenced code, standard, or listing referenced in NFPA 805 for the purposes of this method. However, Section 3.11.5 of NFPA 805 is conditional based on NFPA 805 Chapter 4 and performance-based methods are allowed for this section.

Conclusion:

The use of the described method will ensure that the following requirements of 10 CFR 50.48(c)(2)(vii) are met:

Table P-2 10 CFR 50.48(c)(2)(Vii) Method of Accomplishment

10 CFR 50.48(c)(2)(vii) Requirement	Method of Accomplishment
(a) The required NFPA 805 performance goals, performance objectives, and performance criteria are satisfied.	The fire protection engineering analysis process includes the assessment of impact on NFPA 805 performance goals, performance objectives, and performance criteria are satisfied. Impact will be assessed per risk-informed, performance-based change process in NEI 04-02 Chapter 5 and Appendices I and J and supplemented by RG 1.205, Revision 0, Regulatory Position 3.2.
(b) Safety margins are maintained.	Maintaining safety margins will be ensured using the risk-informed, performance-based change process in NEI 04-02 Chapter 5 and Appendices I and J and supplemented by RG 1.205, Revision 0, Regulatory Position C.3.2.
(c) Fire protection defense-in-depth is maintained.	Maintaining fire protection defense-in-depth will be ensured using the risk-informed, performance-based change process in NEI 04-02 Chapter 5 and Appendices I and J and supplemented by RG 1.205 Regulatory Position C.3.2.

Attachment Q – Risk-Informed, Performance-Based Alternatives to NFPA 805 – 10 CFR 50.48(c)(4)

No risk–informed, performance-based alternatives (10 CFR 50.48(c)(4)) have been applied that deviate from compliance with NFPA 805 requirements.

Attachment R – FSAR Changes

A discussion of the changes to the Final Safety Analysis Report (FSAR) necessitated by the license amendment is provided in this attachment. These changes will be made in accordance with 10 CFR 50.71(e) by applying HNP's FSAR update procedures. (Regulatory Guide 1.205, Regulatory Position C.2.2 and NEI-04-02, Rev. 1, Section 4.5.1).

9.5.1 Fire Protection

The following information provides a general discussion of the fire protection program and systems at the Shearon Harris Nuclear Plant (HNP).

Progress Energy has implemented the process for transitioning from the former deterministic fire protection program and licensing basis to compliance with a risk-informed, performance-based fire protection program as described in 10 CFR 50.48(c). Adoption of NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition in accordance with 10 CFR 50.48(c) serves as the HNP method of satisfying 10 CFR 50.48(a) and General Design Criterion 3. Prior to adoption of NFPA 805, General Design Criterion 3, "Fire Protection" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," was followed in the design of safety and non-safety related structures, systems, and components, as required by 10 CFR 50.48(a). NFPA 805 does not supersede the requirements of GDC 3, 10 CFR 50.48(a), or 10 CFR 50.48(f). Those regulatory requirements continue to apply to licensees that adopt NFPA 805. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements may be met is different than under 10 CFR 50.48(b). Specifically, whereas GDC 3 refers to SSCs important to safety, NFPA 805 identifies fire protection systems and features required to meet the Chapter 1 performance criteria through the methodology in Chapter 4 of NFPA 805. Also, under NFPA 805, the 10 CFR 50.48(a)(2)(iii) requirement to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is ensured is satisfied by meeting the performance criteria in Section 1.5.1 of NFPA 805.

The HNP fire protection program is based on the Nuclear Regulatory Commission (NRC) guidelines, Nuclear Electric Insurance Limited (NEIL) Property Loss Prevention Standards and related industry standards. With regard to NRC criteria, the HNP fire protection program follows the guidance of 10 CFR 50.48(c), which endorses, with exceptions, the National Fire Protection Association's 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition. HNP has further used the guidance of NEI 04-02, Revision 1, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c) as endorsed by Regulatory Guide 1.205, Risk-Informed, Performance Fire Protection for Existing Light-Water Nuclear Power Plants.

9.5.1.1 Design Basis Summary

9.5.1.1.1 Application of NFPA 805

The design basis for Fire Protection at HNP is based on fire protection, nuclear safety objectives, and radiological release objectives put in effect under 10 CFR 50.48(c), which endorses, with exceptions, the National Fire Protection Association's 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition. HNP has further used the guidance of NEI 04-02, Revision 1, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c) as endorsed by Regulatory Guide 1.205, Risk-Informed, Performance Fire Protection for Existing Light-Water Nuclear Power Plants. To this end fire protection features are capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. This is demonstrated through meeting the following performance criteria:

a. *Reactivity Control.* Reactivity control is capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting will occur rapidly enough such that fuel design limits are not exceeded.

b. *Inventory and Pressure Control.* With fuel in the reactor vessel, head on and tensioned, inventory and pressure control is capable of controlling coolant level such that subcooling is maintained such that fuel clad damage as a result of a fire is prevented.

c. *Decay Heat Removal.* Decay heat removal is capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.

d. *Vital Auxiliaries.* Vital auxiliaries are capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.

e. *Process Monitoring.* Process monitoring is capable of providing the necessary indication to assure the criteria addressed in (a) through (d) has been achieved and are maintained.

9.5.1.1.2 Defense-in-Depth

The HNP fire protection program is focused on protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations is paramount. The fire protection program is based on the concept of defense-in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:

(1) Preventing fires from starting,

(2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage,

(3) Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed of fire safety.

9.5.1.1.3 Performance Objectives

10 CFR 50.48(c) through NFPA 805 provides performance objectives for the HNP as follows;

- Nuclear Safety Objectives for the plant, in the event of a fire during any operational mode and plant configuration, are as follows:
    - Reactivity Control shall ensure the capability of rapidly achieving and maintaining subcritical conditions.
        - Fuel Cooling shall ensure the capability of achieving and maintaining decay heat removal and inventory control functions.
        - Fission Product Boundary shall ensure the capability of preventing fuel clad damage so that the primary containment boundary is not challenged.
    - Radioactive Release Objectives shall ensure either of the following objectives shall be met during all operational modes and plant configurations.
        - Containment integrity is capable of being maintained.
        - The source term is capable of being limited.

9.5.1.1.4 Codes of Record

The codes, standards and guidelines used for the design and installation of plant fire protection systems are as follows: (for specific applications and interpretations of codes refer to design documents such as specifications and drawings)

a) American National Standards Institute (ANSI)

B 31.1 1973 - Power Piping

N45.2.9 1974 - Quality Assurance Records, protection from fire hazards

b) American Society for Testing Materials (ASTM)

D-92 -1978 Test for Flash and Fire Points by Cleveland open cup
E-84 -1980 Test for Surface Burning Characteristics of Building Materials
E-119 -1980 Standard Test Method for Fire Test of Building Construction and Materials
E-136 -1979 Standard Test Method for Behavior of Materials in a Vertical Tube Furnace at 750°C

c) Factory Mutual Research (FM) Fire Protection Equipment Approval Guide

d) Institute of Electrical and Electronic Engineers (IEEE) Std. 383-1974 Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations, Std. 384-1974 Criteria for Separation of Class 1E Equipment and Criteria, Std. 634-1978 Standard Cable Penetration Fire Stop Qualification Test

e) National Fire Protection Association (NFPA)

Std. No. 10-1978 – Standard for Portable Fire Extinguishers
Std. No. 13-1978 - Installation of Sprinkler System
Std. No. 14-1976 - Standpipe and Hose Systems
Std. No. 15-1977 - Water Spray Fixed Systems
Std. No. 20-1972 – Standard for Centrifugal Fire Pumps
Std. No. 24-1977 - Outside Protection
Std. No. 30-1977 - Flammable and Combustible Liquids Code
Std. No. 51B-1971 - Cutting and Welding Process
Std. No. 72A-1975 - Local Protective Signaling Systems
Std. No. 72D-1975 - Proprietary Protective Signaling Systems
Std. No. 72E-1978 - Automatic Fire Detectors
Std. No. 80 - 1979- Standard for Fire Doors and Fire Windows
Std. No. 90A-1981 - Air Conditioning and Ventilation Systems
Std. No. 101-1976 - Life Safety Code
Std. No. 251-1979 - Fire Tests, Building Construction and Materials
Std. No. 252-1976 - Fire Tests of Door Assemblies
Std. No. 600-2000 – Standard on Industrial Fire Brigades
Std. No. 701-1999- Standard Methods of Fire Tests for Flame Propagation of Textiles and Films
Std. No.805-2001-Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants

NOTE: HNP Code Compliance Evaluations, as listed below, have been conducted to determine the level of compliance to NFPA Standards for plant areas required to meet the performance criteria of NFPA 805, Section 1.5;

HNP-M/BMRK-0001, Code Compliance Evaluation NFPA 72E, Automatic Fire Detectors
HNP-M/BMRK-0002, Code Compliance Evaluation NFPA 72D, Proprietary Protective Signaling Systems
HNP-M/BMRK-0003, Code Compliance Evaluation NFPA 80, Standard for Fire Doors and Windows
HNP-M/BMRK-0004, Code Compliance Evaluation NFPA 90A, Air Conditioning and Ventilating Systems
HNP-M/BMRK-0005, Code Compliance Evaluation NFPA 10, Portable Fire Extinguishers
HNP-M/BMRK-0006, Code Compliance Evaluation NFPA 14, Standpipe and Hose Systems
HNP-M/BMRK-0007, Code Compliance Evaluation NFPA 20, Centrifugal Fire Pumps
HNP-M/BMRK-0008, Code Compliance Evaluation NFPA 24, Standard for Outside Protection
HNP-M/BMRK-0009, Code Compliance Evaluation NFPA 13, Sprinkler Systems
HNP-M/BMRK-0011, Code Compliance Evaluation NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants
HNP-M/BMRK-0012, Code Compliance Evaluation NFPA 600, Standard on Industrial Fire Brigades
HNP-M/BMRK-0013, Code Compliance Evaluation NFPA 51B, Standard for Cutting and Welding Processes
HNP-M/BMRK-0014, Code Compliance Evaluation NFPA 30, Standard on Flammable and Combustible Liquids

9.5.1.2 Systems Description

9.5.1.2.1 Applicable Systems

Applicable active fire protection systems are those fire suppression and detection systems installed where required to meet the performance or deterministic requirements of NFPA 805, Chapter 4, and are further described in the Progress Energy Shearon Harris Nuclear Power Plant, NFPA 805, Transition to 10 CFR 50.48(c) – NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition, Transition Report, Attachment A – NEI 04-02 Table B-1 – Transition of Fundamental FP Program and Design Elements (NFPA 805 Chapter 3). Credited/required active fire suppression sprinkler systems are identified in the fire area FSAs and the NFPA 805 Code Compliance Calculation.

Likewise, passive fire protection system components are those systems and features installed where required to meet the performance or deterministic requirements of NFPA 805, Chapter 4, and are further described in the fire area FSAs and the NFPA 805 Code Compliance Calculation.

NFPA 805, Chapter 4 establishes the methodology to determine the fire protection systems and features required to achieve the performance criteria outlined in NFPA 805, Section 1.5. The methodology shall be permitted to be either deterministic or performance-based. Deterministic requirements shall be "deemed to satisfy" the performance criteria and require no further engineering analysis. Once a determination has been made that a fire protection system or feature is required to achieve the performance criteria of Section 1.5, its design and qualification shall meet the applicable requirement of Chapter 3.

9.5.1.2.2 Definition of "Power Block" Structures

For the purposes of establishing the structures included in the HNP fire protection program in accordance with 10 CFR 50.48(c) and NFPA 805, the following plant structures are considered to be part of the 'power block'. The following table provides clarification as described in FAQ 06-0019, Define Power Block (ML0850510224) to NEI 04-02.

Power block equipment includes all Structures, Systems and Components (SSCs) required for the safe and reliable operation of the station. It includes all safety-related and balance-of-plant systems and components required for the operation of the station, including radioactive waste processing and storage and switchyard equipment maintained by the station. SSCs required to maintain federal or state regulatory compliance are included in this grouping. This equipment does not include buildings or structures that support station staff, such as offices or storage structures, or the HVAC and support systems focused only on habitability of those structures.

"Power Block" buildings are listed in Table 9.5.1-1.

9.5.1.3 Safety Evaluation (Fire Hazard Safety Analyses)

As part of the NFPA 805 fire protection program a Fire Safety Analysis (FSA) calculation document is provided for each plant fire area. The purpose of the FSA is to demonstrate the achievement of the nuclear safety and radioactive release performance criteria of NFPA 805 as required by 10 CFR 50.48(c), and as such are incorporated by reference to this section. The FSA is a key part of compliance with Section 2.7.1.2 "Fire Protection Program Design Basis Document" of NFPA 805. This analysis also documents results of risk-informed, performance-based evaluations and serves as Progress Energy's design basis document (DBD) as described in NFPA 805, Section 2.7.1.2. FSAs provide evaluation associated with non-compliances from the pre-NFPA 805 licensing basis that have been analyzed and approved during NFPA 805 transition (transition change evaluations). Changes to the post-transition fire protection program that have been analyzed and approved per the requirements of the plant fire protection license condition (post-transition change evaluations).

The following information is documented in each FSA:

- Review and documentation of existing classical fire protection strategy and features in the area. The information is typically what would have been in a plant's Fire Hazards Analysis prior to transition to NFPA 805. This may include suppression system assumptions, and inadvertent actuation or mal-operation evaluations.
- Identify significant fire hazards in the fire area. This is based on NFPA 805 approach to analyze the plant from an ignition source and fuel package perspective.
- Summarize Nuclear Safety Capability Assessment (NSCA) compliance strategies. This is the result of the NEI 04-02 Table B-3 review of the Safe Shutdown Analysis.
- Summarize Non-Power Operations Modes compliance strategies.
- Summarize Radioactive Release compliance strategies. The transition review process required per NEI 04-02 is used to develop these results.
- Provide Fire Probabilistic Risk Assessment (PRA) summary of results. This is based on the results from the plant Fire PRA.
- Perform risk informed, performance based evaluations if needed for the performance based approach.
- Summarize Defense-in-Depth strategy for each fire area.
- Determine key analysis assumptions that are to be included in the NFPA 805 monitoring program.
- Provide conclusions relative to NFPA 805 compliance.

Individual change evaluations, as needed, are documented in the Fire Safety Analysis (FSA) performed for each fire area. The format for the FSA is organized to follow the requirements for preparation and control of design analyses and calculations.

9.5.1.4 Inspection and Testing

9.5.1.4.1 Surveillance Requirements

Guidance for the operability, action, and surveillance requirements for fire protection systems required to meet the performance or deterministic requirements of NFPA 805, Chapter 4 are provided in the Plant Operating Manual procedure(s), and supersedes the original plant technical specifications. Required systems are identified by fire area and required application in the Fire Safety Analysis (FSA) for each fire area.

Applicable systems are further described in HNP-M/BMRK-0011, Code Compliance Evaluation NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants. Required system(s) requirements for operability, action, and surveillance are contained in the plant Fire Protection Program Manual.

9.5.1.4.2 Monitoring

A monitoring program is established to assess the performance of the fire protection program in meeting the performance criteria established in the standard. The monitoring program will be established to ensure that the availability and reliability of the fire protection systems and features are maintained and to assess the performance of the fire protection program in meeting the performance criteria. Monitoring will ensure that the assumptions in the engineering analysis remain valid and acceptable levels of availability, reliability and performance are maintained. Assumptions that are not subject to change do not need to be monitored (e.g., ceiling height input maintained by configuration control process). Deterministic monitoring may carry forward as part of current surveillance processes. The level of monitoring of assumptions and performance are commensurate with associated risk significance.

9.5.1.5 Personnel Qualification & Training

9.5.1.5.1 Program Management

A fire protection program has been established as described in the HNP Fire Protection Program Manual. This manual along with other site POM documents provide the management policy and program direction and defines the responsibilities of those individuals responsible for the program implementation. This program manual establishes the criteria for an integrated combination of components, procedures, and personnel to implement all fire protection program activities. The program manual defines management authority and responsibilities and establishes the general policy for the site fire protection program. The manual designates the senior management position with immediate authority and responsibility for the fire protection program, along with designation of the position responsible for the daily administration and coordination of the fire protection program and its implementation. Qualifications for individuals responsible for administration of a fire protection program are discussed in Section 3.2 and Appendix A, of NFPA 805 (ref., HNP-M/BMRK-0011, Code Compliance Evaluation NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants). This includes recommendations that individuals responsible for day-to-day administration of the fire protection programs be experienced in nuclear power plant fire protection, preferably with a qualified fire protection engineer meeting Society of Fire Protection Engineers (SFPE) member grade qualifications. The manual defines the fire protection interfaces with other organizations and assign responsibilities for the coordination of activities. In addition, the manual identifies the various plant positions having the authority for implementing the various areas of the fire protection program. The program manual also identifies the appropriate Authority Having Jurisdiction (AHJ) for various site areas and portions of the fire protection program. Procedures have been established for implementation of the various facets of the fire protection program in accordance with applicable regulatory and industry requirements and guidance.

9.5.1.5.2 Fire Brigade

On-site firefighting capability as described in NFPA 805 is ensured by the HNP Fire Brigade. Implementation and administrative controls for the NFPA 805 Fire Brigade Program (i.e. training, qualifications, drills) for Progress Energy at the HNP are contained in procedure FIR-NGGC-0007, NFPA 805 Fire Brigade Program, and detailed in HNP-M/BMRK-0011, Code Compliance Evaluation NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants and HNP-M/BMRK-0012, Code Compliance Evaluation NFPA 600, Standard on Industrial Fire Brigades.

Fire Pre-Plans are available for use by the fire brigade and other plant staff responders. Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) will be as low as reasonably achievable and shall not exceed applicable 10 CFR 50, Part 20, Limits. HNP meets the Radioactive Release Performance Criteria of NFPA 805. The detailed review of which is further described in the HNP-M/BMRK-0011, Code Compliance Evaluation NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants.

TABLE 9.5.1-1

Building	Comments
Turbine Building	
Reactor Auxiliary Building	
Waste Processing Building	
Fuel Handling Building	Excluding Areas "K" and "M"
Diesel Generating Building	
Diesel Fuel Oil Storage Building	
Intake Structures	-Essential Service Water Intake Screening Structure -Essential Service Water Intake Structure
Reactor Containment Building	

HNP FSAR

Post-Transitional

17.3.4.1.5 Outside Agency Inspection and Audit Program

a) Fire protection and loss prevention inspection and audit activities under 10 CFR 50.48(c), NFPA 805 program are described in FSAR Section 9.5.1.4.2, Monitoring.

b) As part of the monitoring program a fire protection assessment shall be performed at least once per 36 months using an outside (external to Progress Energy) qualified, (meeting Member grade qualifications of the SFPE) fire protection engineer.

Attachment S – Plant Modifications

The modifications completed to date necessary to support the new licensing basis are identified in this attachment (Regulatory Guide 1.205, Revision 0, Regulatory Position C.2.2. and NEI-04-02, Rev. 1, Section 4.5.1).

Separation of Safe Shutdown Plant Equipment and Cables:

- Remove thermo-lag wall and replace with Interam wrap in ACP
- Re-power component cooling water system valves from an alternate MCC
- Provide alternate power for a water chiller and a control valve in the AFW system
- Eliminate manual action for the RAB electrical equipment protection room return dampers
- Install RWST level indicator at the ACP
- Install manual transfer switch for C CSIP
- Establish VCT valve gallery as fire area and cable protection in 1-A-EPA, 1-A-EPB, and 1-A-BAL-B.
- Re-power CCW valve from alternate MCC and provide cable protection for the associated cables
- Re-analyze Fire Area 1-A-BAL-B1 as 3 new areas
- Provide cable protection for cables in 1-A-CSRB
- Rack out breaker for charging pump suction valves during operations
- Qualification of a steam supply valve for fire in fire Area 1-A-BAL-B
- Provide smoke damper to prevent smoke interaction in rod drive MG set room and ACP
- Provide alternate access entry way in fan area
- Add emergency lighting at 3 normal charging valves
- Modify transfer circuit and power supply for a chiller condenser outlet valve

Attachment T – Clarification of Prior NRC Approvals

The elements of HNP's current fire protection licensing basis for which specific NRC previous approval is uncertain are identified below. Also provided below is sufficient detail to demonstrate how those elements of the current fire protection licensing basis meet the requirements in 10 CFR 50.48(c). (Regulatory Guide 1.205, Revision 0, Regulatory Position C.2.2).

There are no sections of NFPA 805 Chapter 3 that require clarification of prior NRC approval.

Attachment U – No Significant Hazards Consideration

A written evaluation of the significant hazards consideration of a proposed license amendment is required by 10 CFR 50.92. Harris Nuclear Plant (HNP) has evaluated the proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92, a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or
2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or
3. Involve a significant reduction in a margin of safety

This amendment does not involve a significant hazards consideration for the following reasons:

To the extent that these conclusions apply to compliance with the requirements in NFPA 805, these conclusions are based on the following NRC statements in the Statements of Consideration accompanying the adoption of alternative fire protection requirements based on NFPA 805.

1. Does the transition involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Operation of HNP in accordance with the proposed amendment does not increase the probability or consequences of accidents previously evaluated. Engineering analyses, which may include engineering evaluations, probabilistic safety assessments, and fire modeling calculations, have been performed to demonstrate that the performance-based requirements per NFPA 805 have been met. The Final Safety Analysis Report (FSAR) documents the analyses of design basis accidents (DBA) at HNP. The proposed amendment does not adversely affect accident initiators nor alter design assumptions, conditions, or configurations of the facility and does not adversely affect the ability of structures, systems, or components (SSCs) to perform their design function. SSCs required to safely shut down the reactor and to maintain it in a safe shutdown condition will remain capable of performing their design functions.

   The purpose of this amendment is to permit HNP to adopt a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and (c) and the guidance in Revision 0 of Regulatory Guide 1.205. The NRC considers that NFPA 805 provides an acceptable methodology and performance criteria for licensees to identify fire protection requirements that are an acceptable alternative to the Appendix R fire protection features (69 Fed. Reg. 33536, June 16, 2004). Engineering analyses, which may include engineering evaluations, probabilistic safety assessments, and fire modeling calculations, have been performed to demonstrate that the performance-based requirements per NFPA 805 have been met.

NFPA 805, taken as a whole, provides an acceptable alternative for satisfying General Design Criterion 3 (GDC 3) of Appendix A to 10 CFR Part 50, meets the underlying intent of the NRC's existing fire protection regulations and guidance, and achieves defense-in-depth and the goals, performance objectives, and performance criteria specified in Chapter 1 of the standard and, if there are any increases in core damage frequency (CDF) or risk, the increase will be small and consistent with the intent of the Commission's Safety Goal Policy.

Based on this, the implementation of this amendment does not increase the probability of any accident previously evaluated. Equipment required to mitigate an accident remains capable of performing the assumed function. Therefore, the consequences of any accident previously evaluated are not increased with the implementation of this amendment.

2. Does the transition create the possibility of a new or different kind of accident from any kind of accident previously evaluated?

Response: No.

Operation of HNP in accordance with the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. Any scenario or previously analyzed accident with offsite dose was included in the evaluation of design basis accidents (DBA) documented in the FSAR. The proposed change does not alter the requirements or function for systems required during accident conditions. Implementation of the new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and (c) and the guidance in Revision 0 of Regulatory Guide 1.205 will not result in new or different accidents.

The proposed amendment does not adversely affect accident initiators nor alter design assumptions, conditions, or configurations of the facility. The proposed amendment does not adversely affect the ability of SSCs to perform their design function. SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition remain capable of performing their design functions.

The purpose of this amendment is to permit HNP to adopt a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and (c) and the guidance in Revision 0 of Regulatory Guide 1.205. The NRC considers that NFPA 805 provides an acceptable methodology and performance criteria for licensees to identify fire protection requirements that are an acceptable alternative to the Appendix R fire protection features (69 Fed. Reg. 33536, June 16, 2004).

The requirements in NFPA 805 address only fire protection and the impacts of fire on the plant have already been evaluated. Based on this, the implementation of this amendment does not create the possibility of a new of different kind of accident from any kind of accident previously evaluated. The proposed changes do not involve new failure mechanisms or malfunctions that can initiate a new accident. Therefore, the possibility of a new or different kind of accident from any kind of accident previously evaluated is not created with the implementation of this amendment.

3. Does the transition involve a significant reduction in the margin of safety?

Response: No.

Operation of HNP in accordance with the proposed amendment does not involve a significant reduction in the margin of safety. The risk evaluation of plant changes, as appropriate, were measured quantitatively for acceptability using the ΔCDF and ΔLERF criteria from Section 5.3.5 of NEI 04-02 and Regulatory Guide 1.205. The proposed amendment does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by this change. The proposed amendment does not adversely affect existing plant safety margins or the reliability of equipment assumed to mitigate accidents in the SAR. The proposed amendment does not adversely affect the ability of SSCs to perform their design function. SSCs required to safely shut down the reactor and to maintain it in a safe shutdown condition remain capable of performing their design functions.

The purpose of this amendment is to permit HNP to adopt a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and (c) and the guidance in Revision 0 of Regulatory Guide 1.205. The NRC considers that NFPA 805 provides an acceptable methodology and performance criteria for licensees to identify fire protection requirements that are an acceptable alternative to the Appendix R fire protection features (69 Fed. Reg. 33536, June 16, 2004). Engineering analyses, which may include engineering evaluations, probabilistic safety assessments, and fire modeling calculations, have been performed to demonstrate that the performance-based methods do not result in a significant reduction in the margin of safety.

Based on this, the implementation of this amendment does not significantly reduce the margin of safety. The proposed changes are evaluated to ensure that risk and safety margins are kept within acceptable limits. Therefore, the transition does not involve a significant reduction in the margin of safety.

NFPA 805 continues to protect public health and safety and the common defense and security because the overall approach of NFPA 805 is consistent with the key principles for evaluating license basis changes, as described in Regulatory Guide 1.174, is consistent with the defense-in-depth philosophy, and maintains sufficient safety margins.

Based on the above, PEC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and, accordingly, a finding of "no significant hazards consideration" is justified.

Attachment V – Environmental Consideration

Pursuant to 10 CFR 51.22(b), an evaluation of the license amendment request (LAR) has been performed to determine whether it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c). That evaluation shows that the criteria for a categorical exclusion are satisfied for the following reasons. The LAR does not involve:

A significant hazards consideration.
This conclusion is supported by the determination of no significant hazards consideration.

A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.
Compliance with NFPA 805 explicitly requires the attainment of performance criteria, objectives, and goals for radioactive releases to the environment. This radioactive release goal is to provide reasonable assurance that a fire will not result in a radiological release that affects the public, plant personnel, or the environment. The NFPA 805 transition based on fire suppression activities, but not involving fuel damage, has been evaluated and does not create any new source terms. Therefore, this LAR will not change the types or amounts of any effluents that may be released offsite.

A significant increase in the individual or cumulative occupational radiation exposure.
Compliance with NFPA 805 explicitly requires the attainment of performance criteria, objectives, and goals for occupational exposures. Therefore, this LAR will not change the types or amounts of occupational exposures based on the results of the analysis performed under in Table G-1 of NEI 04-02, Attachment E to this document based on fire fighting activities.

In summary, this LAR meets the criteria set forth in 10 CFR 51.22(c)(9) for categorical exclusion from the need for an environmental impact assessment or statement.

Therefore, pursuant to 10 CFR 51.22, Paragraph (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.