Edwin I. Hatch Nuclear Plant Unit 1 and 2
License Amendment Request to Adopt NFPA-805 Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)

Enclosure 2

NFPA 805 LAR Transition Report (Redacted)

Southern Nuclear Operating Company, Inc.

Edwin I. Hatch Nuclear Plant
Docket 50-321 and 50-366

Transition to 10 CFR 50.48(c) - NFPA 805
Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition

Southern Nuclear

Transition Report

April 2018

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Executive Summary

Southern Nuclear Operating Company (SNC) will transition the Hatch Nuclear Plant Units 1 and 2 (HNP) fire protection program to a new Risk-Informed, Performance-Based (RI-PB) alternative per 10 CFR 50.48(c) which incorporates by reference NFPA 805. The licensing basis per 10 CFR 50.48(b) and 10 CFR 50 Appendix R will be superseded.

SNC submitted a letter of intent to the Nuclear Regulatory Commission (NRC) on October 4, 2013 for HNP to adopt National Fire Protection Association (NFPA) standard 805 in accordance with 10 CFR 50.48(c). By letter dated December 2, 2013, the Nuclear Regulatory Commission granted a three-year enforcement discretion period. By letter dated July 6, 2016, SNC requested that the period of enforcement discretion be extended to April 4, 2018. By letter dated October 3, 2016, the NRC approved the enforcement discretion extension request.

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. 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 2 and appropriate Frequently Asked Questions (FAQs).
- Recommended by guidance document Regulatory Guide 1.205 Revision 1.

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

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

Section 5 of the Transition Report provides regulatory evaluations and associated attachments, including:

- Changes to License Condition
- Changes to Technical Specifications, Orders, and Exemptions,
- Determination of No Significant Hazards and evaluation of Environmental Considerations.

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

Attachment H contains the approved FAQs not yet incorporated into the endorsed revision of NEI 04-02. These FAQs have been used to clarify the guidance in RG 1.205, NEI 04-02, and the requirements of NFPA 805 and in the preparation of this License Amendment Request.

Acronym List

AC	Alternating Current
ADS	Automatic Depressurization System
AHJ	Authority Having Jurisdiction
AHU	Air Handling Unit
AISI	American Iron and Steel Institute
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOV	Air Operated Valve
APCSB	Auxiliary Power Conversion Systems Branch
ASD	Alternative Shutdown
ASDC	Alternate Shutdown Cooling
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATTS	Analog Trip Transmitter System
BTP	Branch Technical Position
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners' Group
CAFTA	Computer Aided Fault Tree Analysis
CB	Control Building
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CFAST	Consolidated Fire and Smoke Transport
CFMLA	Circuit Failure Mode Likelihood Analysis
CFR	Code of Federal Regulations
CO	Confirmatory Order
CO_2	Carbon Dioxide
CRD	Control Rod Drive
CS	Core Spray
CST	Condensate Storage Tank
CT	Current Transformer
DBD	Design Basis Document
DBE	Design Basis Event
DC	Direct Current
DH	Decay Heat
DHR	Decay Heat Removal
DID	Defense in Depth

ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EDG, DG	Emergency Diesel Generator
EEEE	Existing Engineering Equivalency Evaluation
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ERFBS	Electrical Raceway Fire Barrier System
ERO	Emergency Response Organization
ESFAS	Engineered Safety Feature Actuation System
F&Os	Facts and Observations
FAQ	Frequently Asked Question
FDM	Fire Data Manager
FDT^{s}	Fire Dynamics Tools
FHA	Fire Hazards Analysis
FIF	Fire Ignition Frequency
FIVE	Fire-Induced Vulnerability Evaluation
FMWB	Fire Modeling Workbook
FPP	Fire Protection Program
FPPR	Fire Protection Program Reevaluation
FPRA	Fire Probabilistic Risk Assessment Analysis
FR	Federal Register
FRE	Fire Risk Evaluation
FSAR (UFSAR)	(Updated) Final Safety Analysis Report
GDC	General Design Criterion/Criteria
GE	General Electric
GL	Generic Letter
GPC	Georgia Power Company
HEP	Human Error Probability
HEPA	High Efficiency Particulate Air
HGL	Hot Gas Layer
HNP	Hatch Nuclear Plant
HPCI	High Pressure Coolant Injection
HRA	Human Reliability Analysis
HRE	High Risk Evolution
HRR	Heat Release Rate
HSD	Hot Shutdown
HSS	High Safety Significant

HVAC	Heating, Ventilation and Air Conditioning
HX	Heat Exchanger
IA	Instrument Air / Independent Assessment
IE	Internal Event
IEEE	Institute of Electrical and Electronics Engineers
IEIF	Internal Events/Internal Flooding
IF	Ignition Frequency
IMC	Inspection Manual Chapter
IN	Information Notice
ISLOCA	Inter-System Loss of Coolant Accident
KSF	Key Safety Function
LA	Licensing Action
LAR	License Amendment Request
LERF	Large Early Release Frequency
LFS	Limiting Fire Scenario
LLS	Low-Low Set
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LS / TS	Limit Switch / Torque Switch
LSS	Low Safety Significant
LWR	Light Water Reactor
MCC	Motor Control Center
MCR	Main Control Room
MEFS	Maximum Expected Fire Scenario
MHIF	Multiple High Impedance Faults
MOV	Motor Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSO	Multiple Spurious Operations
NEC	National Electric Code
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NPO	Non-Power Operations
NPP	Nuclear Power Plant
NPSH	Net Positive Suction Head
NRC, USNRC	United States Nuclear Regulatory Commission

NSCA	Nuclear Safety Capability Assessment
NSEL	Nuclear Safety Equipment List
NSPC	Nuclear Safety Performance Criteria
OMA	Operator Manual Action
OSP	Offsite Power
P&ID	Piping and Instrumentation Diagram
PB	Performance Based
PCS	Primary Control Station
PDMS	Plant Data Management System
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSW	Plant Service Water
PVC	Polyvinyl Chloride
PWR	Pressurized Water Reactor
QA	Quality Assurance
RA	Recovery Action
RAI	Request for Additional Information
RAW	Risk Achievement Worth
RB	Reactor Building
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RI-PB, RIPB	Risk-Informed, Performance-Based
RIS	Regulatory Issue Summary
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSP	Remote Shutdown Panel
RWCU	Reactor Water Clean Up
Rx	Reactor
SBLC	Standby Liquid Control
SCBA	Self-Contained Breathing Apparatus

SDC	Shutdown Cooling
SDP	Significance Determination Process
SE	Safety Evaluation
SER	Safety Evaluation Report
SFPE	Society of Fire Protection Engineers
SM	Safety Margin
SNC	Southern Nuclear Operating Company
SOP	Standard Operating Procedure
SOV	Solenoid Operated Valve
SPC	Suppression Pool Cooling
S/RV, SRV	Safety Relief Valve
SSA	Safe Shutdown Analysis
SSC	Structure, Systems, or Components
SSD	Safe Shutdown
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SUT	Start-Up Transformer
SWGR	Switchgear
TB	Turbine Building
THIEF	Thermally-Induced Electrical Failure
TRM	Technical Requirements Manual
TS	Technical Specification
V&V	Verification and Validation
XFMR	Transformer
VFDRs	Variances From Deterministic Requirements
ZOI	Zone of Influence

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) has promulgated an alternative rule for fire protection requirements at nuclear power plants, 10 CFR 50.48(c), National Fire Protection Association Standard 805 (NFPA 805). Southern Nuclear Operating Company (SNC) is implementing the Nuclear Energy Institute methodology NEI 04-02, "Guidance for Implementing a Risk-informed, Performance-based Fire Protection Program Under 10 CFR 50.48(c)" (NEI 04-02), to transition Hatch Nuclear Plant Units 1 and 2 (HNP) from its current fire protection licensing basis to the new requirements as outlined in NFPA 805. This report describes the transition methodology utilized and documents how HNP complies with the new requirements.

1.1 Background

1.1.1 NFPA 805 – Requirements and Guidance

On July 16, 2004 the NRC amended 10 CFR 50.48, Fire Protection, to add a new subsection, 10 CFR 50.48(c), which establishes new Risk-Informed, Performance-Based (RI-PB) fire protection requirements. 10 CFR 50.48(c) incorporates by reference, with exceptions, the National Fire Protection Association's NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition, as a voluntary alternative to 10 CFR 50.48 Section (b), Appendix R, and Section (f), Decommissioning.

As stated in 10 CFR 50.48(c)(3)(i), any licensee's adoption of a RI-PB program that complies with the rule is voluntary. 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).

NEI developed NEI 04-02 to assist licensees in adopting NFPA 805 and making the transition from their current fire protection licensing basis to one based on NFPA 805. The NRC issued Regulatory Guide (RG) 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants, which endorses NEI 04-02, with exceptions, in December 2009.[1]

A depiction of the primary document relationships is shown in Figure 1-1:

[1] Where referred to in this document NEI 04-02 is Revision 2 and RG 1.205 is Revision 1.

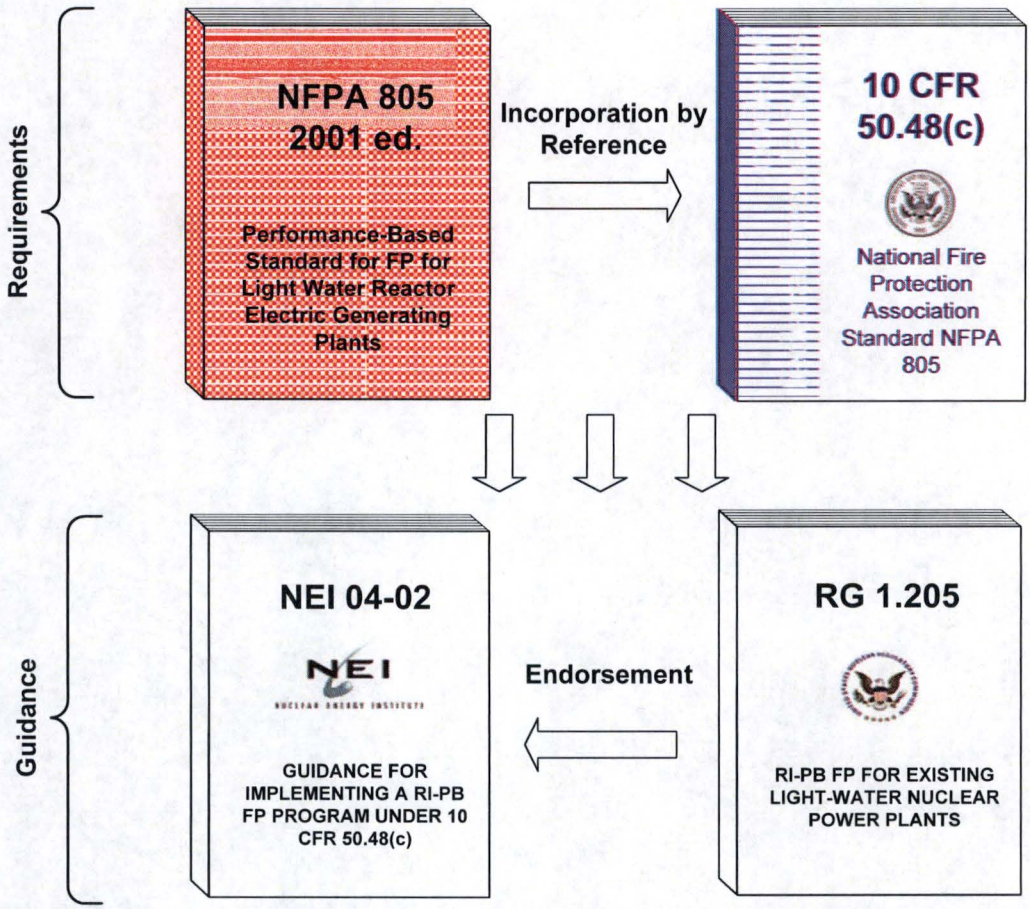

Figure 1-1 NFPA 805 Transition – Implementation Requirements/Guidance

1.1.2 Transition to 10 CFR 50.48(c)

1.1.2.1 Start of Transition

SNC submitted a letter of intent to the NRC on October 4, 2013 (ML13280A299) for HNP to adopt NFPA 805 in accordance with 10 CFR 50.48(c).

By letter dated December 2, 2013 (ML13322B259), the NRC granted a three year enforcement discretion period. In accordance with NRC Enforcement Policy, the enforcement discretion period will continue until the NRC approval of the license amendment request (LAR) is completed.

By letter dated July 6, 2016 (ML16188A341), SNC requested to extend its LAR submittal date 18 months from October 4, 2016, to April 4, 2018, by way of a Confirmatory Order (CO), in accordance with SECY-12-0031, "Enforcement Alternatives for Sites that Indicate Additional Time Required to Submit Their License Amendment Requests to Transition to 10 CFR 50.48(c) National Fire Protection Association Standard 805," dated February 24, 2012 (ML12025A349).

By letter dated October 3, 2016 (ML16223A469), NRC granted the extension by way of CO. In accordance with enforcement policy, the enforcement discretion period will continue until approval of the LAR is complete.

1.1.2.2 Transition Process

The transition to NFPA 805 includes the following high level activities:

- A new Nuclear Safety Capability Assessment (NSCA)
- A new Fire Probabilistic Risk Assessment (PRA) using NUREG/CR-6850, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, as guidance and a revision to the Internal Events PRAs to support the Fire PRAs
- Completion of activities required to transition the pre-transition Licensing Basis to 10 CFR 50.48(c) as specified in NEI 04-02 and RG 1.205

1.2 Purpose

The purpose of the Transition Report is as follows:

1) Describe the process implemented to transition the current fire protection program to comply with the additional requirements of 10 CFR 50.48(c)
2) Summarize the results of the transition process
3) Explain the bases for conclusions that the fire protection program complies with 10 CFR 50.48(c) requirements
4) Describe the new fire protection licensing basis
5) Describe the configuration management processes used to manage post-transition changes to the station and the fire protection program, and resulting impact on the licensing basis

2.0 OVERVIEW OF EXISTING FIRE PROTECTION PROGRAM

2.1 Current Fire Protection Licensing Basis

HNP Units 1 and 2 were licensed to operate on October 13, 1974 and June 13, 1978, respectively. As a result, the HNP fire protection program is based on compliance with 10 CFR 50.48(a), 10 CFR 50.48(b), and the following License Conditions:

Unit 1

2.C(3)

Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained in the updated Fire Hazards Analysis and Fire Protection Program for the Edwin I. Hatch Nuclear Plant, Units 1 and 2, which was originally submitted by letter dated July 22, 1986. Southern Nuclear may make changes to the fire protection program without prior Commission approval only if the changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Unit 2

2.C(3)(a)

Southern Nuclear shall implement and maintain in effect all provision of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained in the updated Fire Hazards Analysis and Fire Protection Program for the Edwin I. Hatch Nuclear Plant Units 1 and 2, which was originally submitted by letter from GPC to the Commission dated July 22, 1986. Southern Nuclear may make changes to the fire protection program without Commission approval only if the changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

2.2 NRC Acceptance of the Fire Protection Licensing Basis

The fire protection program for HNP Unit 1 was originally evaluated by the NRC and found to comply with General Design Criterion (GDC) 3, Fire Protection, of Appendix A to 10 CFR 50, when the original Unit 1 Safety Evaluation Report (SER) was issued on May 11, 1973, as supplemented by SER issued on May 9, 1974. The fire protection program for HNP Unit 2 was originally evaluated by the NRC and found to comply conditionally with GDC 3 and Appendix A to Branch Technical Position (BTP) Auxiliary Power Conversion Systems Branch (APCSB) 9.5-1 when the original Unit 2 SER was issued on June 13, 1978.

At the NRC's request, a reevaluation of the HNP fire protection program was conducted. The HNP Fire Protection Program Report (FPPR) provided a Fire Hazard Analysis and a detailed comparison of the existing and proposed plant fire protection features to the guidelines of Appendix A to BTP APCSB 9.5-1. Georgia Power Company (GPC) submitted the HNP FPPR for Units 1 and 2 on October 27, 1976.

As the fire protection program requirements were clarified, modifications were proposed for improvements to the fire protection program. The modification proposals, as well as the GPC responses to NRC requests for additional information on the reevaluation were provided in amendments to the FPPR, dated September 6, 1977, December 13, 1977, February 15, 1978, March 28, 1978, April 11, 1978, April 18, 1978, and May 19, 1978.

On October 4, 1978, the NRC issued the Fire Protection Safety Evaluation Report for HNP Units 1 and 2. It stated that the fire protection program as then designed and installed met GDC 3 and was acceptable. It further stated that with the completion of scheduled modifications, the program would meet the guidelines of Appendix A to BTP 9.5.1.

Effective February 17, 1981, the NRC amended the Code of Federal Regulations (CFR) to include a new part to 10 CFR 50.48 and a new appendix to 10 CFR 50, Appendix R. This action was taken to upgrade fire protection at nuclear power plants licensed to operate prior to January 1, 1979. 10 CFR 50.48(b) states that those plants whose fire protection features, proposed or implemented, have been accepted by the NRC staff as satisfying the provisions of Appendix A to BTP APCSB 9.5-1 will be required to comply with the requirements of Sections III.G, III.J, and III.O of Appendix R.

Subsequent to the issuance of 10 CFR 50.48 and 10 CFR 50 Appendix R, GPC initiated an evaluation to determine the extent of compliance with Sections III.G, III.J, and III.O of 10 CFR 50 Appendix R. In addition, Paragraph III.G.3 of 10 CFR 50 Appendix R requires that alternative or dedicated shutdown capability be provided in the event that the criteria of Paragraphs III.G.2 or III.G.3(b) are not met. Consequently, GPC also initiated an alternative shutdown capability analysis. Section III.L of 10 CFR 50 Appendix R identifies criteria for alternative and dedicated shutdown capability and Generic Letter 81-12, dated February 20, 1981, provided additional guidance concerning the staff position on safe shutdown capability. The only areas identified as requiring dedicated alternative shutdown capability to satisfy the requirements of Paragraphs III.G.3 and III.L of 10 CFR 50 Appendix R were the cable spreading room and the control room. Based on modifications proposed, the NRC approved the alternative shutdown capability for HNP per SER dated February 11, 1983.

The NRC also approved the following exemptions for HNP:

- An exemption from the 10 CFR 50 Appendix R, Section III.G.3 requirement to provide a fixed fire suppression system in the following fire area in which alternative shutdown capability is provided:
  - Main Control Room – Fire Area 0024

  This exemption was provided in a safety evaluation dated November 16, 1981.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement to provide automatic fire suppression systems and adequate separation between redundant shutdown divisions in the following fire areas:
  - Unit 1 4160V Transformer Room – Fire Area 1019
  - Unit 1 West 600V Switchgear Room – Fire Area 1016

  This exemption was provided in a safety evaluation dated April 18, 1984.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement for redundant shutdown divisions to be separated by complete 3-hour fire rated barriers in the following fire areas:
    - Unit 1 Control Building Working Floor El. 112 Feet – Fire Area 0001
    - Unit 1 West DC Switchgear Room – Fire Area 1018
    - Unit 1 East DC Switchgear Room – Fire Area 1020
    - Unit 1 East 600V Switchgear Room – Fire Area 1017
    - Unit 2 4160V Transformer Room – Fire Area 2019
    - Unit 2 West DC Switchgear Room – Fire Area 2018
    - Unit 2 East DC Switchgear Room – Fire Area 2020
    - Unit 2 West 600V Switchgear Room – Fire Area 2016
    - Unit 2 East 600V Switchgear Room – Fire Area 2017

  This exemption was provided in a safety evaluation dated April 18, 1984.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant systems be separated by either a 3-hour fire rated barrier or a 1-hour rated fire barrier with area-wide automatic fire detection and suppression in the following fire areas:
    - Unit 1 Reactor Building North of Column Line R7 – Fire Area 1205
    - Unit 1 Reactor Building South of Column Line R7 – Fire Area 1203
    - Unit 2 Reactor Building North of Column Line R19 – Fire Area 2203
    - Unit 2 Reactor Building South of Column Line R19 – Fire Area 2205

  This exemption was provided in a safety evaluation dated April 18, 1984.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement to provide a complete area-wide automatic fire suppression system in the following fire area:
    - Unit 2 Control Building Health Physics Area – Fire Area 0014

  This exemption was provided in a safety evaluation dated April 18, 1984.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2. requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers in the following fire area:
    - Unit 2 Control Building Switchgear Hallway – Fire Area 2014

  This exemption was provided in a safety evaluation dated April 18, 1984.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers in the following fire areas:
    - Unit 1 Station Battery Room 1A – Fire Area 1004

- Unit 1 Station Battery Room 1B – Fire Area 1005
- Unit 2 Station Battery Room 2A – Fire Area 2004
- Unit 2 Station Battery Room 2B – Fire Area 2005

This exemption was provided in a safety evaluation dated April 18, 1984.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers in the following fire zones:
  - Unit 2 Turbine Building Condenser Bay – Fire Zone 2101K

  This exemption was provided in a safety evaluation dated April 18, 1984.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers in the following fire areas / zones:
  - Unit 1 East Cableway – Fire Area 1104
  - Unit 2 Turbine Building East Cableway – Fire Area 2104
  - Unit 2 Turbine Building West Cableway – Fire Zone 2101I

  This exemption was provided in a safety evaluation dated April 18, 1984.
- An exemption from the 10CFR 50 Appendix R, Section III.G.2 requirement that an area-wide automatic fire suppression system be installed in the following fire area:
  - Diesel Generator Building Switchgear Room 2G – Fire Area 2409

  This exemption was provided in a safety evaluation dated April 18, 1984.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers or a 1-hour fire rated barrier with area-wide automatic fire suppression and detection systems in the following fire zone:
  - Common Control Building Corridor – Fire Zone 0014K

  This exemption was provided in a safety evaluation dated April 18, 1984.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that an area-wide automatic fire suppression system be installed in the following fire area:
  - Intake Structure – Fire Area 0501

  This exemption was provided in a safety evaluation dated April 18, 1984.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers in the following fire areas:
  - Control Building East Corridor and HP Cold Lab – Fire Area 0007

- Unit 1 Turbine Building – Fire Area 1101
- Unit 2 Turbine Building – Fire Area 2101
- Unit 2 East Cableway – Fire Area 2104

This exemption was provided in a safety evaluation dated April 18, 1984.

- Deviations from the NFPA Standards 13, 14, 15, and 72E regarding:
  - Sprinkler piping hanger design, selection, and spacing criteria
  - Closed head directional spray nozzles and multibushing reductions
  - Positioning of sprinkler heads and fire detectors

  These deviations were provided in a safety evaluation dated April 18, 1984.
- An exemption from the 10 CFR 50 Appendix R, Section III.J requirement that emergency lighting units with at least an 8-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto for the following areas in the plant:
  - Main Control Room – Fire Zone 0024C
  - All Yard Fire Areas

  This exemption was provided in a safety evaluation dated January 2, 1987.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that cables and equipment be protected by a 1-hour fire rated barrier within the suppression system / water curtain boundary in the following fire areas:
  - Unit 1 Reactor Building North of Column Line R7 – Fire Area 1205
  - Unit 1 Reactor Building South of Column Line R7 – Fire Area 1203
  - Unit 2 Reactor Building North of Column Line R19 – Fire Area 2203
  - Unit 2 Reactor Building South of Column Line R19 – Fire Area 2205

  This exemption was provided in a safety evaluation dated January 2, 1987.
- An exemption from the 10 CFR 50 Appendix R, Sections III.G.2(a) and III.G.2(b) requirements to provide barriers between redundant pathways so that a fire will not lead to loss of control of the High Pressure Coolant Injection (HPCI) system in the following fire areas:
  - Unit 1 Reactor Building North of Colum Line R7 – Fire Area 1205
  - Unit 2 Reactor Building South of Column Line R19 – Fire Area 2205

  This exemption was provided in a safety evaluation dated January 2, 1987.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.1(a) requirement that repairs should not be used to maintain hot shutdown in the following fire area:
  - Control Complex – Fire Area 0024

  This exemption was provided in a safety evaluation dated January 2, 1987

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2b requirement that redundant shutdown divisions be separated by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards in the following fire area:
  - Intake Structure – Fire Area 0501

  This exemption was provided in a safety evaluation dated January 2, 1987.

3.0 TRANSITION PROCESS

3.1 Background

Section 4.0 of NEI 04-02 describes the process for transitioning from compliance with the current fire protection licensing basis to the new requirements of 10 CFR 50.48(c). NEI 04-02 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) Submit a LAR;
5) Complete transition activities that can be completed prior to the receipt of the License Amendment;
6) Receive a Safety Evaluation; and
7) Complete implementation of the new licensing basis, including completion of modifications identified in Attachment S.

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 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 RI-PB methods. Consistent with the guidance in NEI 04-02, SNC has implemented the NFPA 805 Section 2.2 process by first determining the extent to which its current fire protection program supports findings of deterministic compliance with the requirements in NFPA 805. RI-PB methods are being applied to the requirements for which deterministic compliance could not be shown.

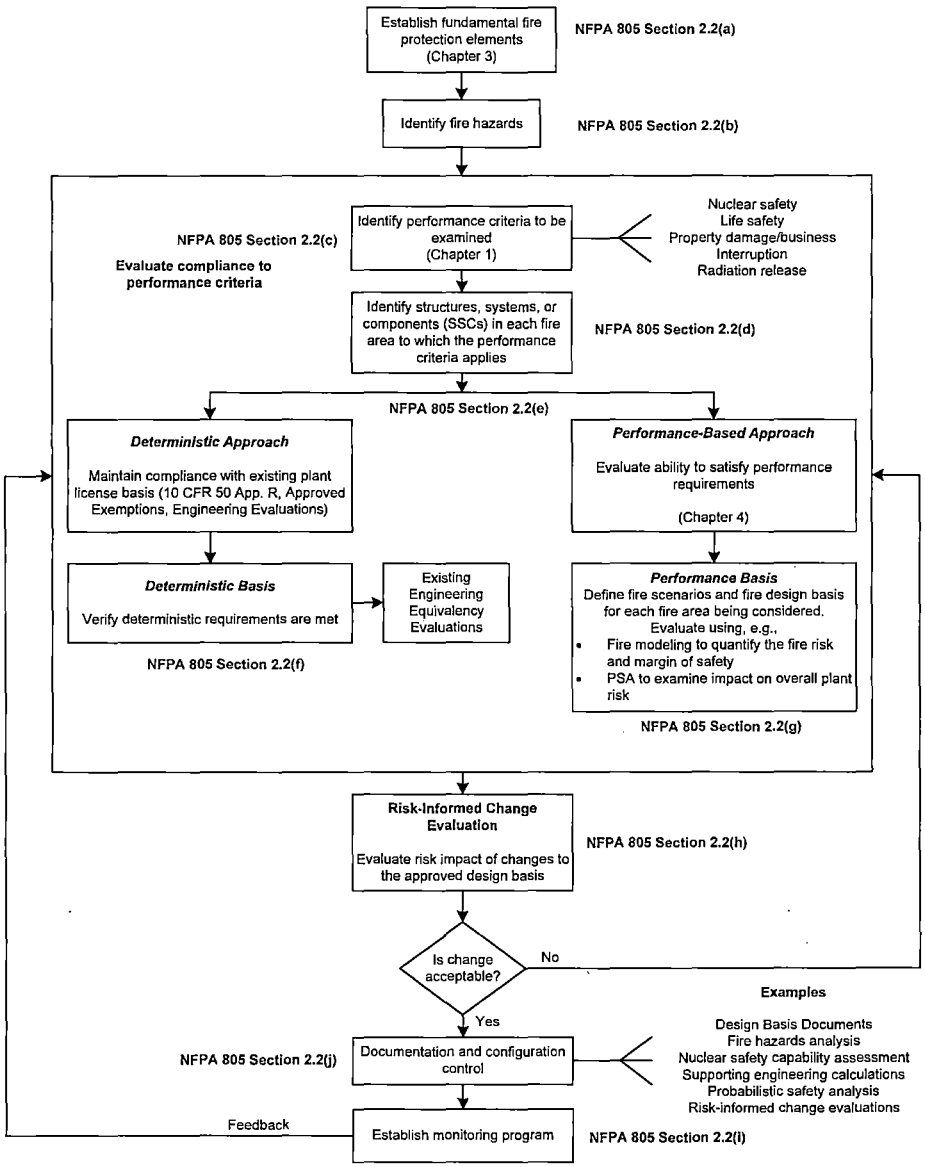

Figure 3-1 NFPA 805 Process [NEI 04-02 Figure 3-1 based on Figure 2-2 of NFPA 805][2]

3.3 NEI 04-02 – NFPA 805 Transition Process

NFPA 805 contains technical processes and requirements for a RI-PB fire protection program. NEI 04-02 was developed to provide guidance on the overall process (programmatic, technical, and licensing) for transitioning from a traditional fire protection licensing basis to a new RI-PB method based upon NFPA 805, as shown in Figure 3-2.

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[2] Note: 10 CFR 50.48(c) does not incorporate by reference Life Safety and Plant Damage/Business Interruption goals, objectives and criteria. See 10 CFR 50.48(c) for specific exceptions to the incorporation by reference of NFPA 805.

Section 4.0 of NEI 04-02 describes the detailed process for assessing a fire protection program for compliance with NFPA 805, as shown in Figure 3-2.

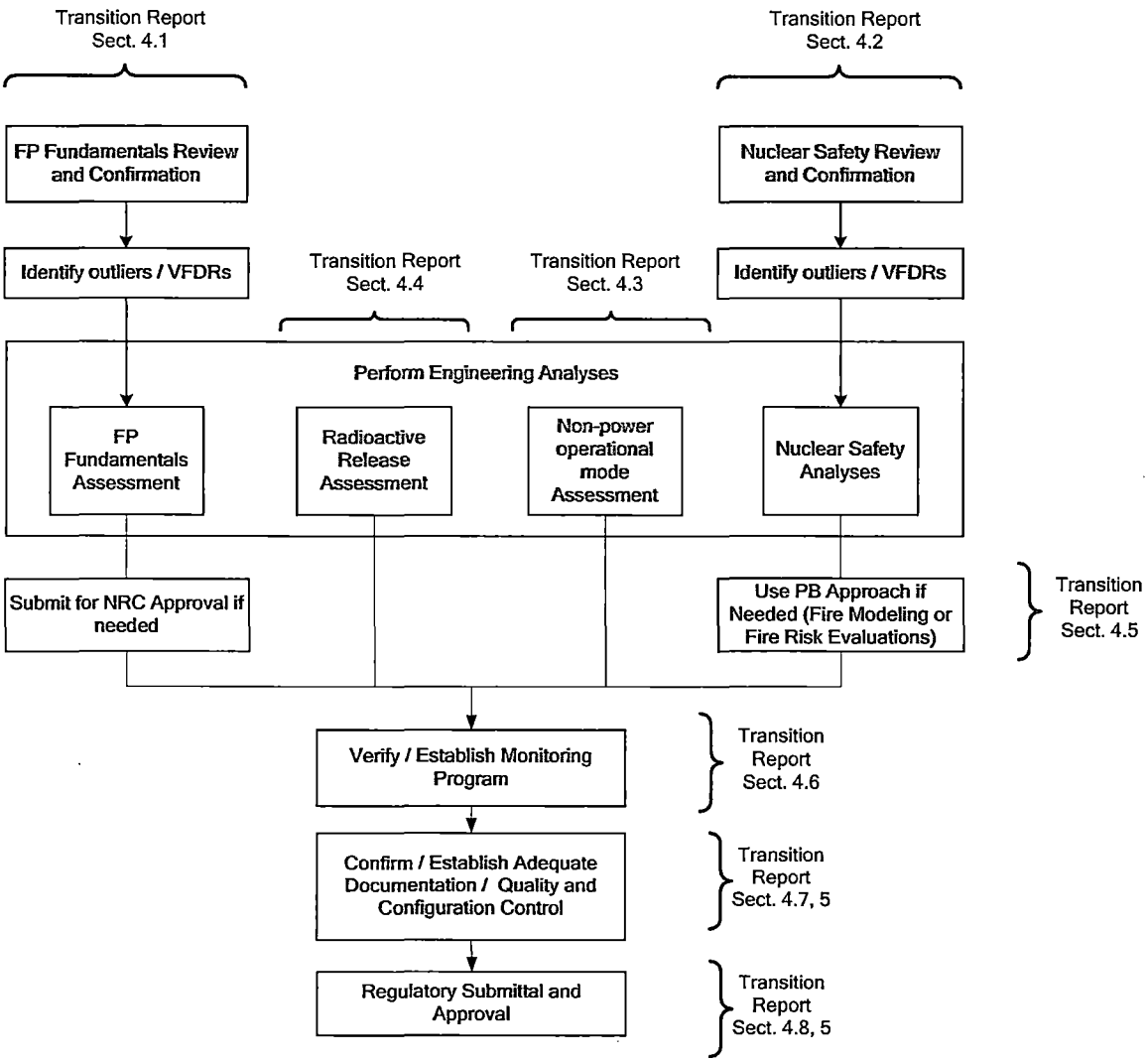

Figure 3-2 Transition Process (Simplified) [based on NEI 04-02 Figure 4-1]

3.4 NFPA 805 Frequently Asked Questions (FAQs)

The NRC 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 RG 1.205, NEI 04-02, and the requirements of NFPA 805. The NFPA 805 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 Regulatory Issues Summary (RIS) 2007-19, Process for Communicating Clarifications of Staff Positions Provided in RG 1.205 Concerning Issues Identified during the Pilot Application of NFPA Standard 805, dated August 20, 2007 (ML071590227).

Under the FAQ Process, transition issues are submitted to the NEI NFPA 805 Task Force for review, and subsequently presented to the NRC during public FAQ meetings. Once the NEI NFPA 805 Task Force and NRC reach agreement, the NRC issues a memorandum to indicate that the FAQ is acceptable. NEI 04-02 will be revised to incorporate the approved FAQs. This is an on-going revision process that will continue through the transition of NFPA 805 plants. Final closure of the FAQs will occur when future revisions of RG 1.205, endorsing the related revisions of NEI 04-02, are approved by the NRC. It is expected that additional FAQs will be written and existing FAQs will be revised as plants continue NFPA 805 transition after the Pilot Plant Safety Evaluations.

Attachment H contains the list of approved FAQs not yet incorporated into the endorsed revision of NEI 04-02. These FAQs have been used to clarify the guidance in RG 1.205, NEI 04-02, and the requirements of NFPA 805 and in the preparation of this LAR.

4.0 COMPLIANCE WITH NFPA 805 REQUIREMENTS

4.1 Fundamental Fire Protection Program and Design Elements

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

4.1.1 Overview of Evaluation Process

The comparison of the HNP Fire Protection Program to the requirements of NFPA 805 Chapter 3 was performed and documented in Calculation SMNH-16-086, "NFPA 805 Chapter 3 Fundamental Fire Protection Program and Design Elements Review." Calculation SMNH-16-086 used the guidance contained in NEI 04-02, Section 4.3.1 and Appendix B-1 (See Figure 4-1).

Each section and subsection of NFPA 805 Chapter 3 was reviewed against the current fire protection program. Upon completion of the activities associated with the review, the following compliance statement(s) was used:

- Complies - For those sections/subsections determined to meet the specific requirements of NFPA 805
- Complies with Clarification - For those sections/subsections determined to meet the requirements of NFPA 805 with clarification
- Complies by previous NRC approval - For those sections/subsections where the specific NFPA 805 Chapter 3 requirements are not met but previous NRC approval of the configuration exists.
- Complies with use of Existing Engineering Equivalency Evaluations (EEEEs) - For those sections/subsections determined to be equivalent to the NFPA 805 Chapter 3 requirements as documented by engineering analysis
- Complies, with Required Action – For those sections/subsections that will comply upon completion of a modification or other "implementation item," such as a procedure change or a work request. When this compliance statement is used, applicable implementation items are identified in the compliance basis, and also appear in Attachment S.
- Submit for NRC Approval - For those sections/subsections for which approval is sought in this LAR submittal in accordance with 10 CFR 50.48(c)(2)(vii). A summary of the bases of acceptability is provided (See Attachment L for details).

In some cases, multiple compliance statements have been assigned to a specific NFPA 805 Chapter 3 section/subsection. Where this is the case, each compliance/compliance basis statement clearly references the corresponding requirement of NFPA 805 Chapter 3.

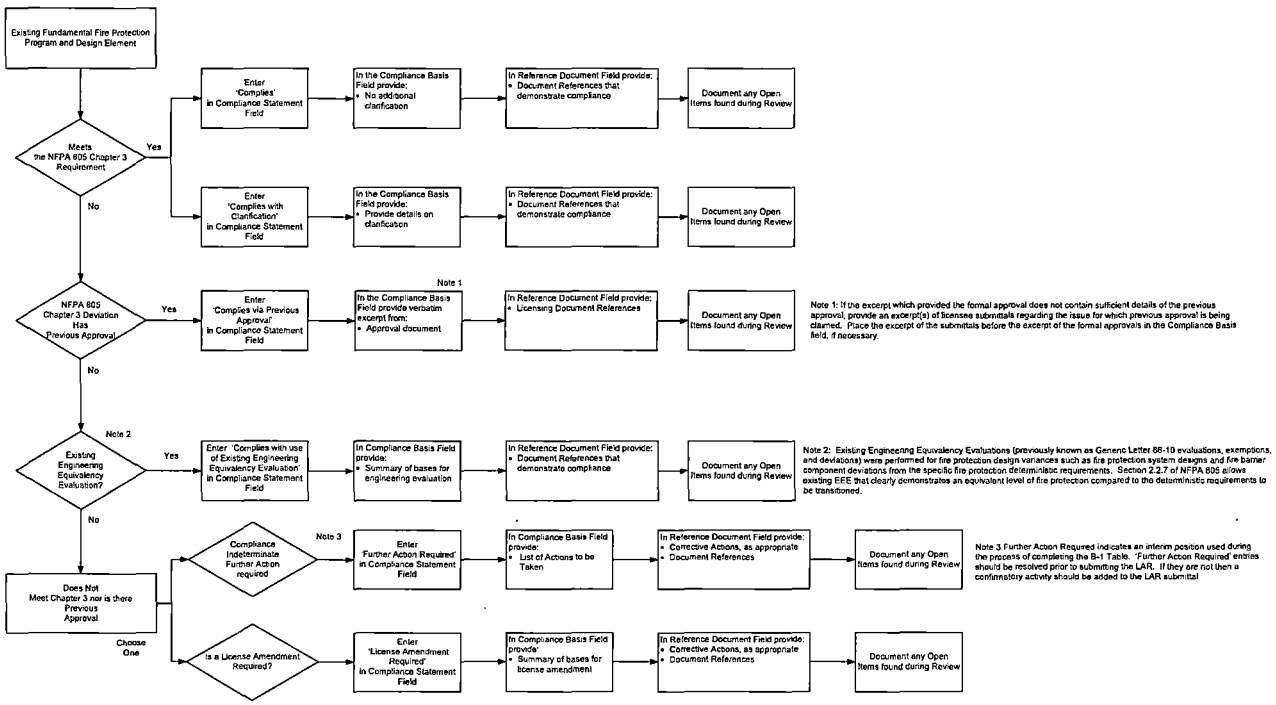

Figure 4-1 - Fundamental Fire Protection Program and Design Elements Transition Process [Based on NEI 04-02 Figure 4-2][3]

[3] Figure 4-1 depicts the process used during the transition and therefore contains elements (i.e., open items) that represent interim resolutions. Additional detail on the transition of EEEEs is included in Section 4.2.2.

4.1.2 Results of the Evaluation Process

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

Attachment A contains the NEI 04-02 Table B-1, Transition of Fundamental Fire Protection Program and Design Elements. This table provides the compliance basis for the requirements in NFPA 805 Chapter 3. Except as identified in Section 4.1.2.3, Attachment A demonstrates that the fire protection program at HNP either:

- Complies directly with the requirements of NFPA 805 Chapter 3,
- Complies with clarification with the requirements of NFPA 805 Chapter 3,
- Complies through the use of existing engineering equivalency evaluations which are valid and of appropriate quality, or
- Complies with a previously NRC approved alternative to NFPA 805 Chapter 3 and therefore the specific requirement of NFPA 805 Chapter 3 is supplanted.
- Complies, with Required Action, with applicable implementation items identified in the compliance basis, and also appear in Attachment S.

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

NFPA 805 Section 3.1 states in part, "Previously approved alternatives from the fundamental protection program attributes of this chapter by the AHJ take precedence over the requirements contained herein." In some cases, prior NRC approval of an NFPA 805 Chapter 3 program attribute may be unclear. SNC requests that the NRC concur with their finding of prior approval for the following sections of NFPA 805 Chapter 3:

- None.

4.1.2.3 NFPA 805 Chapter 3 Requirements Not Met and 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:

- 3.2.3(1) – Approval is requested for the use of EPRI performance-based fire protection inspection, testing, and maintenance frequencies
- 3.3.4 – Approval is requested for thermal insulation materials
- 3.3.5.1 – Approval is requested for wiring above suspended ceilings
- 3.3.5.2 – Approval is requested for PVC coated conduit and flexible conduit in excess of 3 ft. lengths
- 3.5.2 and 3.5.10 – Approval is requested for the lack of check valves in the fire water tanks discharge piping
- 3.5.3 – Approval is requested for fire pump controller NFPA 20 compliance
- 3.5.5 – Approval is requested for fire pump separation

- 3.5.16 - Approval is requested for alternate use of fire protection water
- 3.6.1 – Approval is requested for the current installation of Class II standpipe and hose systems in specific areas in lieu of Class III standpipe and hose systems

The specific deviation and a discussion of how the alternative satisfies 10 CFR 50.48(c)(2)(vii) requirements are provided in Attachment L. SNC requests NRC approval of these performance-based methods.

4.1.3 Definition of Power Block and Plant

Where used in NFPA 805 Chapter 3 the terms "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 pre-transition licensing basis.

SNC assembled a list of HNP structures that are required to meet the nuclear safety and radioactive release criteria described in Section 1.5 of NFPA 805.

These structures are listed in Attachment I and define the "power block" and "plant".

4.2 Nuclear Safety Performance Criteria

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 provides a systematic process for determining the extent to which the pre-transition licensing basis meets these criteria and for identifying any necessary fire protection program changes. NEI 04-02, Appendix B-2 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

The Nuclear Safety Capability Assessment (NSCA) Methodology review consists of four processes:

- Establishing compliance with NFPA 805 Section 2.4.2
- Establishing the Safe and Stable Conditions for the Plant
- Establishing Recovery Actions
- Evaluating Multiple Spurious Operations

The methodology for demonstrating reasonable assurance that a fire during non-power operational (NPO) modes will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition is an additional requirement of 10 CFR 50.48(c) and is addressed in Section 4.3.

4.2.1.1 Compliance with NFPA 805 Section 2.4.2

Overview of Process

NFPA 805 Section 2.4.2 Nuclear Safety Capability Assessment states:

> *"The purpose of this section is to define the methodology for performing a nuclear safety capability assessment. The following steps shall be performed:*
> *(1) Selection of systems and equipment and their interrelationships necessary to achieve the nuclear safety performance criteria in Chapter 1*
> *(2) Selection of cables necessary to achieve the nuclear safety performance criteria in Chapter 1*
> *(3) Identification of the location of nuclear safety equipment and cables*
> *(4) Assessment of the ability to achieve the nuclear safety performance criteria given a fire in each fire area"*

Establishing compliance with NFPA 805 Section 2.4.2 was done by evaluating the NSCA methodology against the guidance provided in NEI 00-01 Revision 2, Chapter 3, as discussed in Appendix B-2 of NEI 04-02. This guidance was used because it is endorsed as an acceptable methodology in NRC RG 1.205 and due to feedback received as a result of NRC requests for additional information on other post-pilot plant LARs.

The methodology is depicted in Figure 4-2 and consisted of the following activities:

- Each specific section of NFPA 805 2.4.2 was correlated to the corresponding section of Chapter 3 of NEI 00-01 Revision 2. Based upon the content of the NEI 00-01 methodology statements, a determination was made of the applicability of the section to the station.
- The plant-specific methodology was compared to applicable sections of NEI 00-01 and one of the following alignment statements and its associated basis were assigned to the section:
  - Aligns
  - Aligns with intent
  - Not in Alignment
  - Not in Alignment, but Prior NRC Approval
  - Not in Alignment, but no adverse consequences
  - For those sections that do not align, an assessment was made to determine if the failure to maintain strict alignment with the guidance in NEI 00-01 could have adverse consequences. Since NEI 00-01 is a guidance document, portions of its text could be interpreted as 'good practice' or intended as an example of an efficient means of performing the analyses. If the section has no adverse consequences, these sections of NEI 00-01 can be dispositioned without further review.

The comparison of the HNP NSCA methodology to NEI 00-01 Chapter 3 (NEI 04-02 Table B-2) was performed and documented in SNC Calculation SENH-16-005, "Nuclear Safety Capability Assessment Methodology Review (Table B-2)."

Results from Evaluation Process

The method used to perform the NSCA with respect to selection of systems and equipment, selection of cables, and identification of the location of equipment and cables, either meets the NRC endorsed guidance from NEI 00-01, Chapter 3 directly or met the intent of the endorsed guidance with adequate justification as documented in Attachment B. Plant documents in Attachment B that identify the bases for alignment with NEI 00-01 will be maintained post-transition to ensure NFPA 805 program compliance. See Plant Modifications Committed in Table S-2 and Implementation Items in Table S-3 of Attachment S.

NEI 00-01, Chapter 3 contains guidance criteria for identifying required and important to safe shutdown components. These specific guidance criteria are not applicable to plants transitioning to NFPA 805; therefore, they were not addressed for HNP.

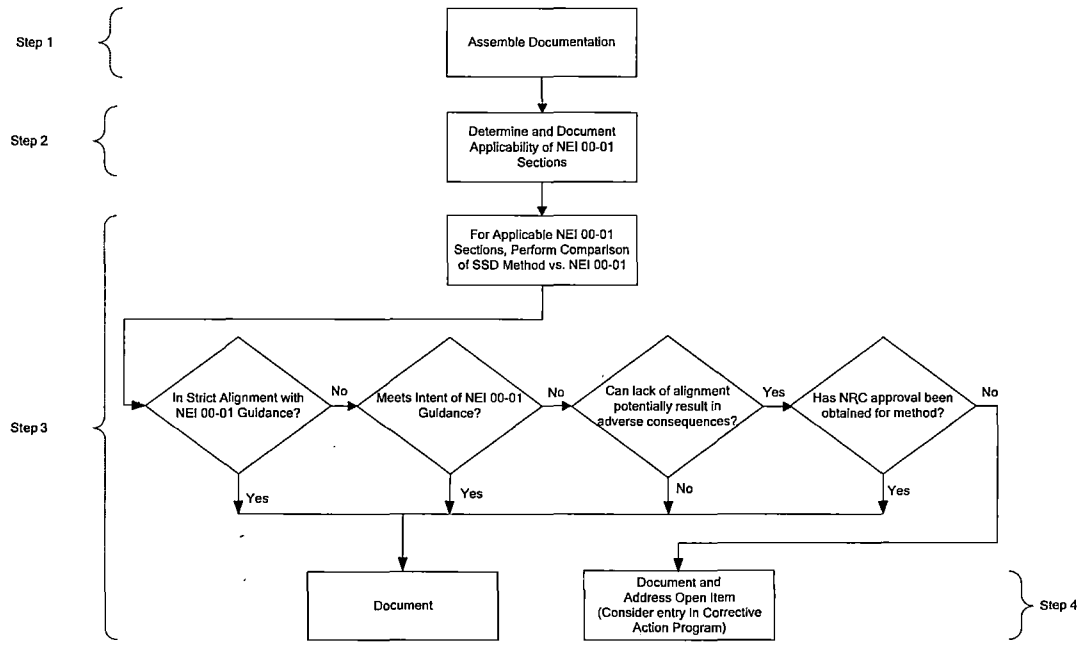

Figure 4-2 – Summary of Nuclear Safety Methodology Review Process (FAQ 07-0039)

4.2.1.2 Safe and Stable Conditions for the Plant

Overview of Process

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

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

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

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

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

Results

Based on HNP Calculation SENH-15-009, "Nuclear Safety Capability Assessment Report", the NFPA 805 licensing basis for HNP is to achieve and maintain the fuel in a "safe and stable" condition assuming a fire occurs during Mode 1 (Power Operation), Mode 2 (Startup), of Mode 3 (Hot Shutdown), when the motor control center (MCC) breakers for the RHR shutdown cooling suction valves are de-energized. The analyses demonstrate that each fire area can maintain safe and stable hot shutdown conditions for a 24 hour coping period.

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

- At-Power analysis, Modes 1-3. This analysis is discussed in Section 4.2.4.
- Non-Power analysis, which includes portions of Mode 3 (when the MCC breakers for the RHR shutdown cooling suction valves are energized), Mode 4, and Mode 5. This analysis is discussed in Section 4.3.

To establish and maintain safe and stable conditions, RPV inventory makeup is provided by HPCI, RCIC, RHR or Core Spray systems. Decay Heat Removal is provided by RHR and RHR Service Water (RHRSW) with heat transfer to the ultimate heat sink. RPV pressure control is accomplished with the SRVs.

Required AC buses are powered from Offsite Power (OSP) or Onsite Emergency Diesel Generators (EDG). Required DC buses are maintained from battery chargers with battery backup.

The minimum shift operating staff are capable of performing Recovery Actions to establish and maintain safe and stable plant conditions. Hatch and SNC procedures provide for the establishment of the Emergency Response Organization (ERO), which can provide normal operations staff with support to maintain Hot Shutdown conditions for an extended period of time.

Methods to Maintain Safe and Stable Conditions

The following describes methods to maintain the Safe and Stable conditions:

1. Reactivity Control is established and maintained by a reactor scram and insertion of control rods. Subcritical conditions are achieved for all reactor operating modes and maintaining subcritical conditions after a reactor scram is a passive function. There are no additional actions required for reactivity control to sustain safe and stable conditions beyond 24 hours.
2. HNP has design features and procedures to ensure that an adequate source of inventory is provided for RPV level control in sustained Mode 3 (Hot Shutdown) conditions. Makeup water will be provided from the Condensate Storage Tank (CST) and subsequently transitioned to the suppression pool. Adequate CST water capacity is available to maintain RPV level until conditions are met to transfer the RCIC, HPCI, RHR or CS pump suction to the suppression pool. Inventory required for RPV makeup when using low pressure injection is recirculated from the suppression pool. There are no additional actions required for makeup inventory to sustain safe and stable conditions beyond 24 hours.
3. RPV Inventory and Pressure Control is maintained using high pressure systems, HPCI and RCIC, or low pressure injection with RHR in LPCI mode or the Core Spray system. For low pressure injection, the SRVs are credited for pressure reduction. Adequate nitrogen sources are available to the SRVs to establish safe and stable conditions. Additional actions may be required to provide nitrogen from backup onsite or offsite supplies to sustain safe and stable conditions beyond 24 hours. Actions to replenish nitrogen supplies are considered routine, are covered by existing procedures and can be anticipated in ample time.
4. Core decay heat in Mode 3 (Hot Shutdown) will be rejected to the torus through the SRV tailpipes. Long term cooling is maintained using the RHR system in Alternate Shutdown Cooling mode or Suppression Pool Cooling mode with at least 2 SRVs available to provide core flow. The RHR Service Water system rejects decay heat to the ultimate heat sink. There are no additional actions required for Decay Heat Removal to sustain safe and stable conditions beyond 24 hours.
5. Exhaust steam from the RCIC/HPCI turbine is discharged to the barometric condensers and then returned to the suppression pool. Although this increases the suppression pool temperature, the peak pool temperature and pressure are within the design limits of the containment, and adequate Net Positive Suction Head (NPSH) for credited pumps is assured. There are no additional actions required for Suppression Pool cooling to sustain safe and stable conditions beyond 24 hours.
6. HNP is equipped with five diesel generators supplying standby power to three 4.16kV essential buses for each unit. Diesel generator 1B is shared between Unit 1 and 2; supplying power to either Bus 1F or 2F. The Fuel Oil Supply Storage and Transfer Subsystem provides a supply of fuel oil to the diesel engines. The storage capacity of the tanks is sufficient to operate four diesel

generators for 7 days at 3250 KW power operation. Day tanks contain capacity for 2 hours of full-load operations. In the event that fuel oil supply approaches exhaustion after seven days, actions to replenish fuel oil are considered routine, are covered by existing procedures and can be anticipated in ample time.

7. Battery chargers are credited with maintaining DC Station batteries at rated voltage. Should AC charging sources be lost, recovery actions may be required to recover sufficient chargers. Each Station Service battery is adequately sized to support the required loads for two hours without recharging. With battery chargers aligned to required DC buses, there are no additional actions required to sustain safe and stable conditions beyond 24 hours.
8. A separate 125-V diesel building battery is furnished for each diesel generator and its associated 4-kV bus. Each of the diesel building batteries is adequately sized to support the required loads for two hours without recharging. With battery chargers aligned, there are no additional actions required to sustain safe and stable conditions beyond 24 hours.
9. The PSW system features two strainers per unit to remove suspended matter from the water leaving the PSW pumps. The operation of the strainers includes an automatic backwash feature. Additionally, plant procedures ensure that operators monitor differential pressure across the strainers. Additional action may be required to manually backwash PSW strainers to sustain safe and stable conditions beyond 24 hours.
10. The RHRSW system features four strainers per unit to remove suspended matter from the water leaving the RHRSW pumps. The design of the RHRSW strainers provides for manual backwash. Plant procedures ensure that operators monitor differential pressure across the strainers during RHRSW operation. Additional action may be required to manually backwash RHRSW strainers to sustain safe and stable conditions beyond 24 hours.
11. The NSCA includes the required process monitoring instruments for the operation of each credited system. Recovery actions may be required to mitigate the loss of required instrumentation to establish safe and stable conditions. There are no additional actions required for process monitoring to sustain safe and stable conditions beyond 24 hours.

Assessment of Risk

A qualitative assessment of the risk of failure of the actions necessary to sustain safe and stable conditions beyond 24 hours has determined the risk to be very low because the requisite inventory and manpower for maintaining systems operable is not time critical. This is based on the following:

- There are a limited number of actions required to maintain safe and stable and the actions are provided within plant procedures.
- The length of time for depletion of commodities such as fuel oil and nitrogen.
- The ERO will be available and will provide sufficient resources for assessment of fire damage and completion of support activities and repairs to equipment

necessary to maintain hot standby for an extended period, transition to cold shutdown, or return to power operations as dictated by the plant fire event.

4.2.1.3 Establishing Recovery Actions

Overview of Process

NEI 04-02 and RG 1.205 suggest that a licensee submit a summary of its approach for addressing the transition of OMAs as recovery actions in the LAR (Regulatory Position 2.2.1 and NEI-04-02, Section 4.6). As a minimum, NEI 04-02 suggests that the assumptions, criteria, methodology, and overall results be included for the NRC to determine the acceptability of the licensee's methodology.

The discussion below provides the methodology used to transition pre-transition OMAs and to determine the population of post-transition recovery actions. This process is based on FAQ 07-0030 (ML110070485) and consists of the following steps:

- Step 1: Clearly define the primary control station(s) and determine which pre-transition OMAs are taken at primary control station(s) (Activities that occur in the Main Control Room are not considered pre-transition OMAs). Activities that take place at primary control station(s) or in the Main Control Room are not recovery actions, by definition.
- Step 2: Determine the population of recovery actions that are required to resolve variances from deterministic requirements (VFDRs) (to meet the risk acceptance criteria or maintain a sufficient level of defense-in-depth).
- Step 3: Evaluate the additional risk presented by the use of recovery actions required to demonstrate the availability of a success path
- Step 4: Evaluate the feasibility of the recovery actions
- Step 5: Evaluate the reliability of the recovery actions

Results

The review results are documented in SENH-15-009, "Nuclear Safety Capability Assessment Report," SENH-16-007, "Recovery Action Feasibility Assessment," and SMNH-16-093, "Fire Risk Evaluation Report,". Refer to Attachment G for the detailed evaluation process and summary of the results from the process.

4.2.1.4 Evaluation of Multiple Spurious Operations

Overview of Process

NEI 04-02 suggests that a licensee submit a summary of its approach for addressing potential fire-induced 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 NRC to determine the acceptability of the licensee's methodology. The methodology utilized to address MSOs for HNP is summarized below.

As part of the NFPA 805 transition project, a review and evaluation of HNP's susceptibility to fire-induced MSOs was performed. The process was conducted consistent with NEI 04-02 and RG 1.205, as supplemented by FAQ 07-0038 Revision 3

(ML110140242). The BWR Owners Group Generic MSO list included in NEI 00-01 was utilized.

The approach outlined in Figure 4-3 (based on FAQ 07-0038) is one acceptable method to address fire-induced MSOs. This method used insights from the Fire PRA developed in support of transition to NFPA 805 and consists of the following:

- Identifying potential MSOs of concern.
- Conducting an expert panel to assess plant specific vulnerabilities (e.g., per NEI 00-01, Rev. 1 Section F.4.2).
- Updating the Fire PRA model and NFPA 805 NSCA to include the MSOs of concern.
- Evaluating for NFPA 805 Compliance.
- Documenting Results.

This process is intended to support the transition to a new licensing basis. Post-transition changes would use the RI-PB 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).

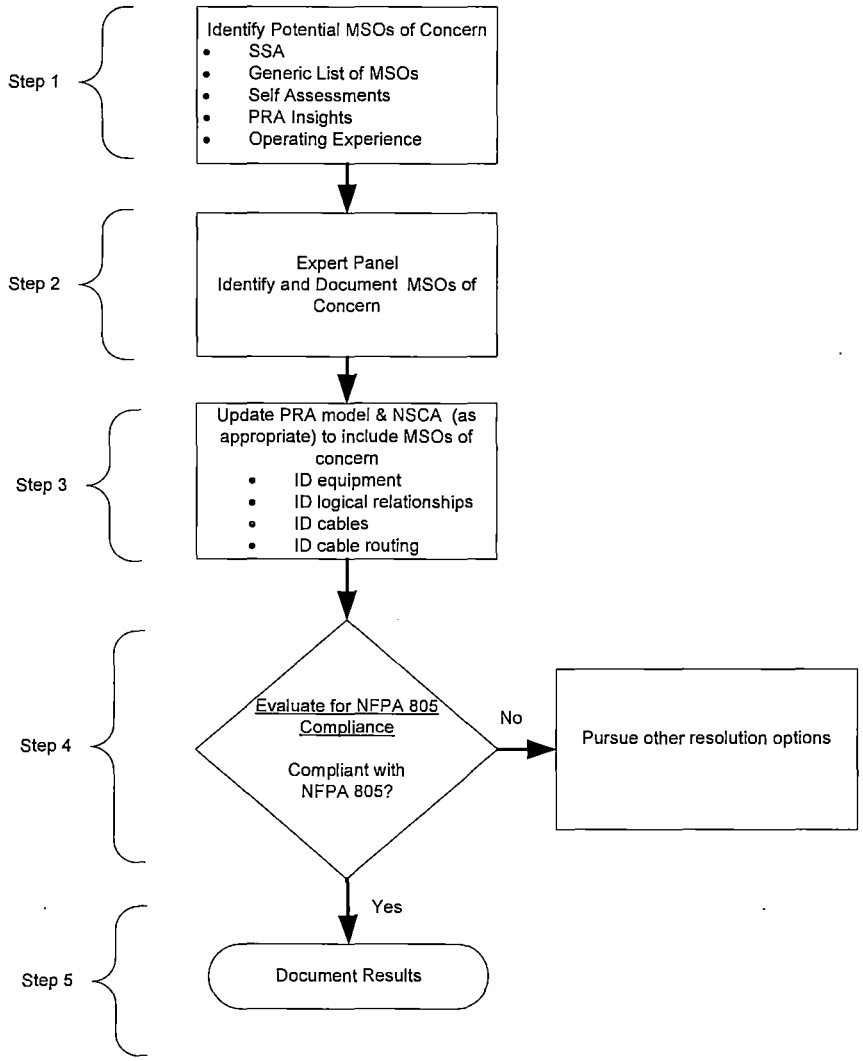

Figure 4-3 – Multiple Spurious Operations – Transition Resolution Process (Based on FAQ 07-0038)

Results

Refer to Attachment F for the process used and the results.

4.2.2 Existing Engineering Equivalency Evaluation Transition

Overview of Evaluation Process

The EEEEs that support compliance with NFPA 805 Chapter 3 or Chapter 4 (both those that existed prior to the transition and those that were created during the transition) were reviewed using the methodology contained in NEI 04-02. The methodology for performing the EEEE review included the following determinations:

- The EEEE is not based solely on quantitative risk evaluations,
- The EEEE is an appropriate use of an engineering equivalency evaluation,
- The EEEE is of appropriate quality,

- The standard license condition is met,
- The EEEE is technically adequate,
- The EEEE reflects the plant as-built condition, and
- The basis for acceptability of the EEEE remains valid

In accordance with the guidance in RG 1.205, Regulatory Position 2.3.2 and NEI 04-02, as clarified by FAQ 08-0054, Demonstrating Compliance with Chapter 4 of NFPA 805, EEEEs that demonstrate that a fire protection system or feature is "adequate for the hazard" are summarized in the LAR as follows:

- If not requesting specific approval for "adequate for the hazard" EEEEs, then the EEEE was referenced where required and a brief description of the evaluated condition was provided.
- If requesting specific NRC approval for "adequate for the hazard" EEEEs, then EEEE was referenced where required to demonstrate compliance and was included in Attachment L for NRC review and approval.

In all cases, the reliance on EEEEs to demonstrate compliance with NFPA 805 requirements was documented in the LAR.

Results

The review results for EEEEs are documented in Calculation SMNH-16-089, "Existing Engineering Equivalency Evaluation Review".

In accordance with the guidance provided in RG 1.205, Regulatory Position 2.3.2, NEI 04-02, as clarified by FAQ 08-0054, Demonstrating Compliance with Chapter 4 of NFPA 805, EEEEs used to demonstrate compliance with Chapters 3 and 4 of NFPA 805 are referenced in the Attachments A and C as appropriate.

None of the transitioning EEEEs require NRC approval.

4.2.3 Licensing Action Transition

Overview of Evaluation Process

The existing licensing actions (exemptions / deviations / safety evaluations) review was performed in accordance with NEI 04-02. The methodology for the licensing action review included the following:

- Determination of the bases for acceptability of the licensing action.
- Determination that these bases for acceptability are still valid and required for NFPA 805.

Results

Attachment K contains the detailed results of the Licensing Action Review. The licensing action review is documented in SNC Calculation SMNH-16-090, "Existing Licensing Action Review".

The following licensing actions will be transitioned into the NFPA 805 fire protection program as previously approved (NFPA 805 Section 2.2.7). These licensing actions are considered compliant under 10 CFR 50.48(c).

- None

The following licensing actions are no longer necessary and will not be transitioned into the NFPA 805 fire protection program:

- An exemption from the 10 CFR 50 Appendix R, Section III.G.3 requirement to provide a fixed fire suppression system in the following fire area in which alternative shutdown capability is provided:
  - Main Control Room – Fire Area 0024

  This exemption is not being credited for compliance with NFPA 805 because Fire Area 0024 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2, and no credit is taken for prior approval of the deterministic compliance strategy for this fire area.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement to provide automatic fire suppression systems and adequate separation between redundant shutdown divisions in the following fire areas:
  - Unit 1 4160V Transformer Room – Fire Area 1019
  - Unit 1 West 600V Switchgear Room – Fire Area 1016

  This exemption is not being credited for compliance with NFPA 805 because Fire Areas 1016 and 1019 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these areas. Additionally, the subject boundaries have been demonstrated to be adequate for the hazard in EEEEs.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement for redundant shutdown divisions to be separated by complete 3-hour fire rated barriers in the following fire areas:
  - Unit 1 Control Building Working Floor El. 112 Feet – Fire Area 0001
  - Unit 1 West DC Switchgear Room – Fire Area 1018
  - Unit 1 East DC Switchgear Room – Fire Area 1020
  - Unit 1 East 600V Switchgear Room – Fire Area 1017
  - Unit 2 4160V Transformer Room – Fire Area 2019
  - Unit 2 West DC Switchgear Room – Fire Area 2018
  - Unit 2 East DC Switchgear Room – Fire Area 2020
  - Unit 2 West 600V Switchgear Room – Fire Area 2016
  - Unit 2 East 600V Switchgear Room – Fire Area 2017

This exemption is no longer required because the subject boundaries have been demonstrated to be adequate for the hazard in EEEEs.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant systems be separated by either a 3-hour fire rated barrier or a 1-hour rated fire barrier with area-wide automatic fire detection and suppression system for the following fire areas:
  - Unit 1 Reactor Building North of Column Line R7 – Fire Area 1205
  - Unit 1 Reactor Building South of Column Line R7 – Fire Area 1203
  - Unit 2 Reactor Building North of Column Line R19 – Fire Area 2203
  - Unit 2 Reactor Building South of Column Line R19 – Fire Area 2205

  This exemption is not being credited for compliance with NFPA 805 because Fire Areas 1203, 1205, 2203, and 2205 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these areas. Additionally, the subject boundaries have been demonstrated to be adequate for the hazard in EEEEs.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement to provide a complete area-wide automatic fire suppression system in the following fire area:
  - Unit 2 Control Building Health Physics Area – Fire Area 0014

  This exemption is not being credited for compliance with NFPA 805 because the fire risk evaluation concludes that Fire Area 0014 is a performance-based area in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for this area.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2. requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers in the following fire area:
  - Unit 2 Control Building Switchgear Hallway – Fire Area 2014

  This exemption is not being credited for compliance with NFPA 805 because Fire Areas 0014 and 2014 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these areas. Additionally, the subject boundaries have been demonstrated to be adequate for the hazard in EEEEs.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers in the following fire areas:
  - Unit 1 Station Battery Room 1A – Fire Area 1004
  - Unit 1 Station Battery Room 1B – Fire Area 1005
  - Unit 2 Station Battery Room 2A – Fire Area 2004

- Unit 2 Station Battery Room 2B – Fire Area 2005

This exemption is no longer required because the subject boundaries have been demonstrated to be adequate for the hazard in EEEEs.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers in the following fire zones:
  - Unit 2 Turbine Building Condenser Bay – Fire Zone 2101K

This exemption is no longer required because the subject boundaries have been demonstrated to be adequate for the hazard in EEEEs.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers in the following fire areas / zones:
  - Unit 1 East Cableway – Fire Area 1104
  - Unit 2 Turbine Building East Cableway – Fire Area 2104
  - Unit 1 Turbine Building West Cableway – Fire Zone 1101I
  - Unit 2 Turbine Building West Cableway – Fire Zone 2101I

This exemption is not being credited for compliance with NFPA 805 because Fire Areas 1101, 1104, 2101, and 2104 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these areas. Additionally, the subject boundaries have been demonstrated to be adequate for the hazard in EEEEs.

- An exemption from the 10CFR 50 Appendix R, Section III.G.2 requirement that an area-wide automatic fire suppression system be installed in the following fire area:
  - Diesel Generator Building Switchgear Room 2G – Fire Area 2409

This exemption is not being credited for compliance with NFPA 805 because Fire Area 2409 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for this area.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers or a 1-hour fire rated barrier with area-wide automatic fire suppression and detection systems in the following fire zone:
  - Common Control Building Corridor – Fire Zone 0014K

This exemption is not being credited for compliance with NFPA 805 because Fire Areas 0014 and 2101 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these areas. Additionally,

the subject boundaries have been demonstrated to be adequate for the hazard in EEEEs.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that an area-wide automatic fire suppression system be installed in the following fire area:
  - Intake Structure – Fire Area 0501

  This exemption is not being credited for compliance with NFPA 805 because Fire Area 0501 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for this area.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that redundant shutdown divisions be separated by complete 3-hour fire rated barriers in the following fire areas:
  - Control Building East Corridor and HP Cold Lab – Fire Area 0007
  - Unit 1 Turbine Building – Fire Area 1101
  - Unit 2 Turbine Building – Fire Area 2101
  - Unit 2 East Cableway – Fire Area 2104

  This exemption is not being credited for compliance with NFPA 805 because Fire Areas 0007, 1101, 2101, and 2104 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these areas. Additionally, the subject boundaries have been demonstrated to be adequate for the hazard in EEEEs.
- Deviations from the NFPA Standards 13, 14, 15, and 72E regarding:
  - Sprinkler piping hanger design, selection, and spacing criteria
  - Closed head directional spray nozzles and multibushing reductions
  - Positioning of sprinkler heads and fire detectors

  This exemption is no longer required because the subject deviations have been demonstrated to be adequate for the hazard in EEEEs.
- An exemption from the 10 CFR 50 Appendix R, Section III.J requirement that emergency lighting units with at least an 8-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto for the following areas in the plant:
  - Main Control Room – Fire Zone 0024C
  - All Yard Fire Areas

  This exemption is no longer required because there are no deterministic requirements for emergency lighting contained within NFPA 805; therefore, although adequate levels of emergency lighting must be maintained on site, the exemption from the requirements of 10 CFR 50 Appendix R is not required.

- An exemption from the 10 CFR 50 Appendix R, Section III.G.2 requirement that cables and equipment be protected by a 1-hour fire rated barrier within the suppression system / water curtain boundary in the following fire areas:
  - Unit 1 Reactor Building North of Column Line R7 – Fire Area 1205
  - Unit 1 Reactor Building South of Column Line R7 – Fire Area 1203
  - Unit 2 Reactor Building North of Column Line R19 – Fire Area 2203
  - Unit 2 Reactor Building South of Column Line R19 – Fire Area 2205

  This exemption is not being credited for compliance with NFPA 805 because Fire Areas 1203, 1205, 2203, and 2205 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these areas.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.1(a) requirement that repairs should not be used to maintain hot shutdown in the following fire area:
  - Control Complex – Fire Area 0024

  This exemption is not being credited for compliance with NFPA 805 because Fire Area 0024 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for this area.
- An exemption from the 10 CFR 50 Appendix R, Sections III.G.2(a) and III.G.2(b) requirements to provide barriers between redundant pathways so that a fire will not lead to loss of control of the High Pressure Coolant Injection (HPCI) system in the following fire areas:
  - Unit 1 Reactor Building North of Colum Line R7 – Fire Area 1205
  - Unit 2 Reactor Building South of Column Line R19 – Fire Area 2205

  This exemption is not being credited for compliance with NFPA 805 because the fire risk evaluation concludes that Fire Areas 1205 and 2205 are performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these areas.
- An exemption from the 10 CFR 50 Appendix R, Section III.G.2b requirement that redundant shutdown divisions be separated by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards in the following fire area:
  - Intake Structure – Fire Area 0501

  This exemption is not being credited for compliance with NFPA 805 because Fire Area 0501 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for this area.

Since the exemptions and deviations are no longer necessary, in accordance with the requirements of 10 CFR 50.48(c)(3)(i), SNC requests that the exemptions and deviations listed in Attachment K be rescinded as part of the LAR process. It is SNC'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. See Attachment O, Orders and Exemptions.

4.2.4 Fire Area Transition

Overview of Evaluation Process

The Fire Area Transition (NEI 04-02 Table B-3) was performed using the methodology contained in NEI 04-02 and FAQ 08-0054. The methodology for performing the Fire Area Transition, depicted in Figure 4-4, is outlined as follows:

Step 1 - Assembled documentation. Gathered industry and plant-specific fire area analyses and licensing basis documents.

Step 2 – Documented fulfillment of nuclear safety performance criteria.

- Assessed accomplishment of nuclear safety performance goals. Documented the method of accomplishment, in summary level form, for the fire area.
- Documented evaluation of effects of fire suppression activities. Documented the evaluation of the effects of fire suppression activities on the ability to achieve the nuclear safety performance criteria.
- Performed licensing action reviews. Performed a review of the licensing aspects of the selected fire area and document the results of the review. See Section 4.2.3.
- Performed existing engineering equivalency evaluation reviews. Performed a review of existing engineering equivalency evaluations (or created new evaluations) documenting the basis for acceptability. See Section 4.2.2.
- Pre-transition OMA reviews. Performed a review of pre-transition OMAs to determine those actions taking place outside of the main control room or outside of the primary control station(s). See Section 4.2.1.3.

Step 3 – VFDR Identification and characterization and resolution considerations. Identified variances from the deterministic requirements of NFPA 805, Section 4.2.3. Documented variances as either a separation issue or a degraded fire protection system or feature. Developed VFDR problem statements to support resolution.

Step 4 – Performance-Based evaluations (Fire Modeling or Fire Risk Evaluations (FREs)) See Section 4.5.2 for additional information.

Step 5 – Final Disposition.

- Documented final disposition of the VFDRs in Attachment C (NEI 04-02 Table B-3).
- For recovery action compliance strategies, ensured the manual action feasibility analysis of the required recovery actions was completed. Note: if a recovery action cannot meet the feasibility requirements established per NEI 04-02, then alternate means of compliance was considered.
- Documented the post transition NFPA 805 Chapter 4 compliance basis.

Step 6 – Documented required fire protection systems and features. Reviewed the NFPA 805 Section 4.2.3 compliance strategies (including fire area licensing actions and engineering evaluations) and the NFPA 805 Section 4.2.4 compliance strategies (including simplifying deterministic assumptions) to determine the scope of fire protection systems and features 'required' by NFPA 805 Chapter 4. The 'required' fire protection systems and features are subject to the applicable requirements of NFPA 805 Chapter 3.

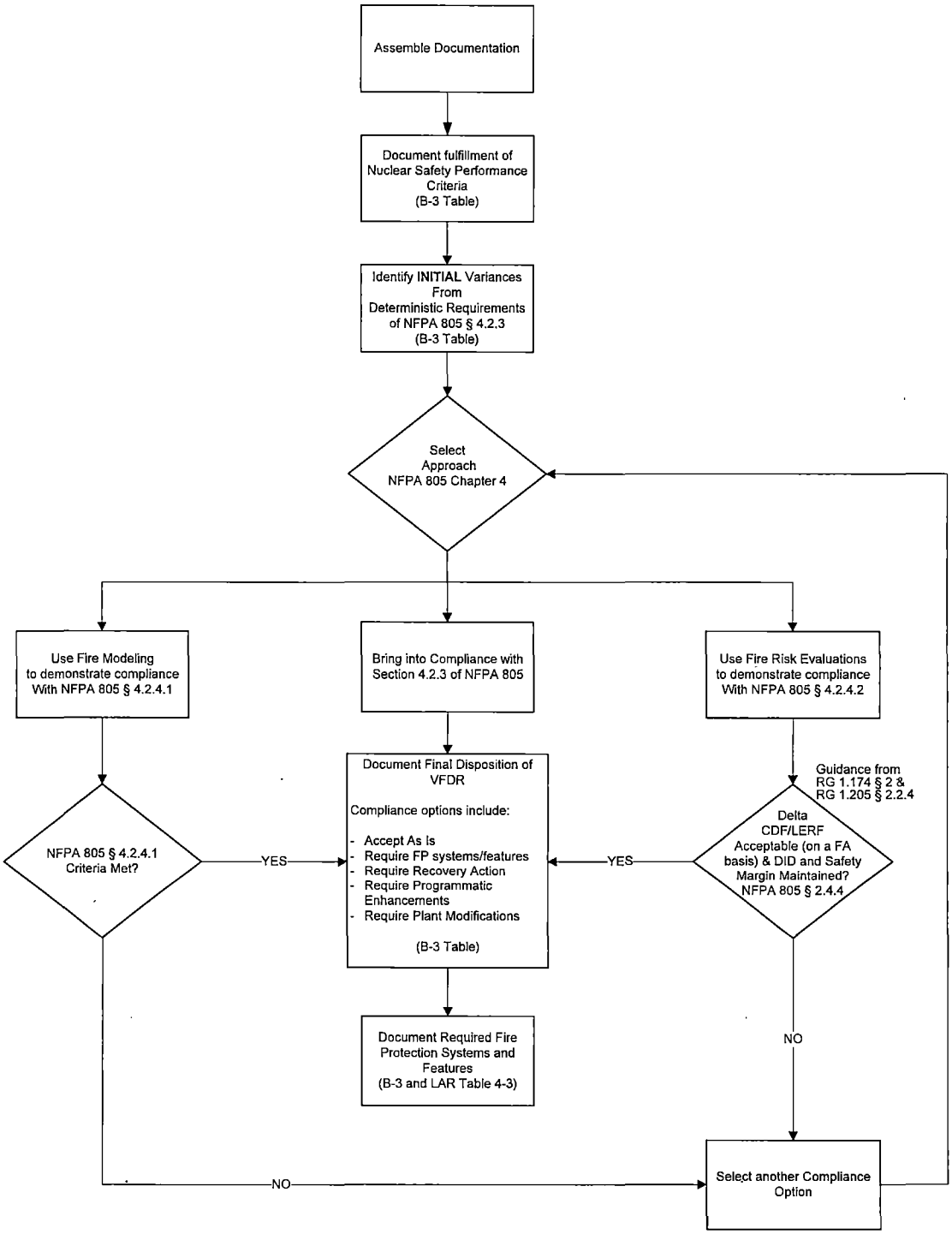

Figure 4-4 – Summary of Fire Area Review [Based on FAQ 08-0054 Revision 1]

Results of the Evaluation Process

Attachment C contains the results of the Fire Area Transition review (NEI 04-02 Table B-3). On a fire area basis, Attachment C summarizes compliance with Chapter 4 of NFPA 805.

NEI 04-02 Table B-3 includes the following summary level information for each fire area:

- Regulatory Basis – NFPA 805 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 Nuclear Safety Capability Assessment Documents are provided.
- Fire Suppression Activities Effect on Nuclear Safety Performance Criteria – A summary of the method of accomplishment is provided.
- Licensing Actions – Specific references to exemption requests, deviations, and safety evaluations that will remain part of the post-transition licensing basis. A brief description of the condition and the basis for acceptability of the licensing action is provided.
- EEEE – Specific references to EEEE that rely on determinations of "adequate for the hazard" that will remain part of the post-transition licensing basis. A brief description of the condition and the basis for acceptability is provided.
- VFDRs – Specific variances from the deterministic requirements of NFPA 805 Section 4.2.3. Refer to Section 4.5.2 for a discussion of the performance-based approach.

4.3 Non-Power Operational Modes

4.3.1 Overview of Evaluation Process

HNP implemented the process outlined in NEI 04-02 and FAQ 07-0040, Clarification on Non-Power Operations. The goal (as depicted in Figure 4-5) is to ensure that contingency plans are established when the plant is in a Non-Power Operational (NPO) mode where the risk is intrinsically high. 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 NPO modes involved the following steps:

- Reviewed the existing Outage Management Processes
- Identified Equipment/Cables:
  - Reviewed plant systems to determine success paths that support each of the defense-in-depth Key Safety Functions (KSFs), and
  - Identified cables required for the selected components and determined their routing.
- Performed Fire Area Assessments (identify pinch points – plant locations where a single fire may damage all success paths of a KSF).

- Managed pinch-points associated with fire-induced vulnerabilities during the outage.

The process is depicted in Figures 4-5 and 4-6. The results are presented in Section 4.3.2.

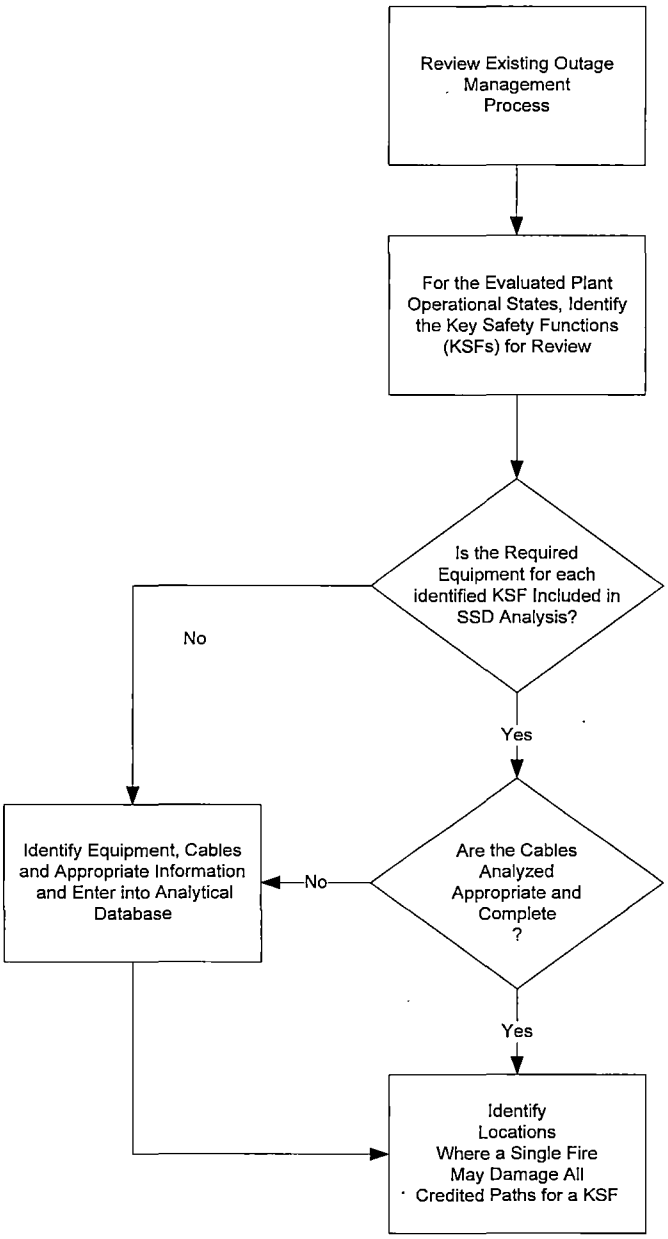

Figure 4-5 Review POSs, KSFs, Equipment, and Cables, and Identify Pinch Points

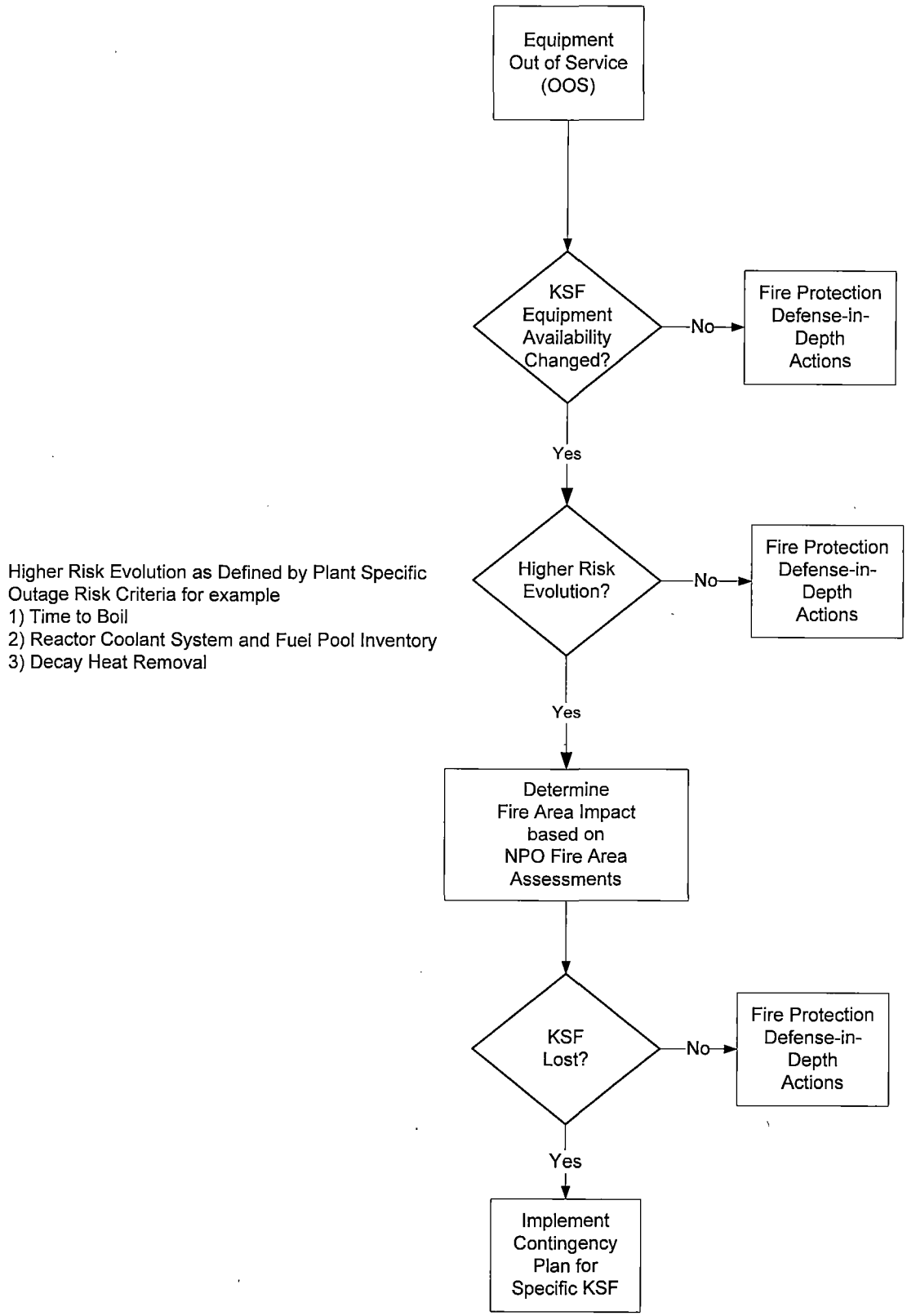

Figure 4-6 Manage Pinch Points

4.3.2 Results of the Evaluation Process

Based on FAQ 07-0040, the Plant Operating States (POS) considered for equipment and cable selection are defined in HNP Calculation SENH-16-002, "Non-Power Operations (NPO) Report." Systems and components were identified to provide the following KSFs:

1. Decay Heat Removal Capability
2. Electrical Power Availability
3. Inventory Control
4. Reactivity Control
5. Spent Fuel Pool Cooling
6. Containment Integrity
7. Support System (e.g., PSW, HVAC, etc.)

The process for the selection and treatment of components was consistent with the methodology in the NSCA. For those components not already in the Fire Data Manager (FDM) or those with a functional state for NPO differing from that in the At-Power Analysis, circuit analysis and routing was performed as described in the plant's NSCA methodology.

Any fire area experiencing fire damage which eliminates all success paths for an NPO KSF was considered a "pinch point". The strategies listed below are permissible methods for restoring a NPO KSF or preventing its loss.

HNP Calculation SENH-16-002, "Non-Power Operations (NPO) Report" contains the NPO fire area assessment, the identified pinch points, and the credited strategies consistent with FAQ 07-0040 to reduce fire risk. The list of actions and generic recommendations specified in the report considers the following from FAQ 07-0040:

- 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 configuration changes (e.g., removing power from equipment once it is placed in desired position).
- Provision of additional fire patrols at periodic intervals or other appropriate compensatory measures (such as surveillance cameras) during increased vulnerability.
- Use of actions by operators to mitigate potential losses of KSFs.
- Identification and monitoring in situ ignition sources for "fire precursors" (e.g., equipment temperatures).
- Reschedule the work to a period with lower risk or higher defense-in-depth.

Attachment D provides additional detail for the HNP NPO transition.

Implementation of the NPO fire area assessment results into the HNP outage management process will be completed as part of the NFPA 805 Program Implementation. The following implementation items are associated with the NPO transition:

- Applicable HNP procedures will be revised to contain guidance developed by the NPO review. This will include revising procedures as necessary to ensure that FP program, operational controls, administrative controls, and housekeeping requirements are established and maintained to the extent practical during periods of higher shutdown risk. (See Attachment S, Table S-3, Implementation Item IMP-12)

4.4 Radioactive Release Performance Criteria

4.4.1 Overview of Evaluation Process

The review of the fire protection program against NFPA 805 requirements for fire suppression related radioactive release was performed using the methodology contained in HNP Calculation SMNH-16-091, "NFPA 805 Radioactive Release Review". The methodology consisted of the following:

- A review of fire pre-plans and fire brigade training materials to identify fire protection program elements (e.g., systems / components / procedural control actions / flow paths, etc.) that are being credited to meet the radioactive release goals, objectives, and performance criteria during all plant operating modes, including full power and non-power conditions.
- A review of engineering controls to ensure containment of gaseous and liquid effluents (e.g., smoke and fire fighting agents). This review included all plant operating modes (including full power and non-power conditions). Otherwise, provided a bounding analysis, quantitative analysis, or other analysis that demonstrates that the limitations for instantaneous release of radioactive effluents specified in the unit's Technical Specifications are met.

4.4.2 Results of the Evaluation Process

HNP Calculation SMNH-16-091, "NFPA 805 Radioactive Release Review," details the results of the screening process and review of pre-fire plans, fire brigade training materials, and engineering controls.

The radioactive release review determined the fire protection program will be compliant with the requirements of NFPA 805 and the guidance in NEI 04-02 and RG 1.205 upon completion of the implementation items identified in Attachment S, Table S-3.

The site specific review of the direct effects of fire suppression on radioactive release is summarized in Attachment E.

4.5 Fire PRA and Performance-Based Approaches

RI-PB evaluations are an integral element of an NFPA 805 fire protection program. Key parts of RI-PB evaluations include:

- A Fire PRA (discussed in Section 4.5.1 and Attachments U, V, and W).
- NFPA 805 Performance-Based Approaches (discussed in Section 4.5.2).

4.5.1 Fire PRA Development and Assessment

In accordance with the guidance in RG 1.205, a Fire PRA model was developed for HNP in compliance with the requirements of Part 4 "Requirements for Fires At Power PRA," of the ASME and ANS combined PRA Standard, ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Application," (hereafter referred to as Fire PRA Standard). The Fire PRA model is based on the Internal Events/Internal Flooding (IEIF) model. SNC had peer reviews conducted for both the IEIF and Fire PRA models by independent industry analysts in accordance with RG 1.200 Revision 2, NEI 05-04, and NEI 07-12, and the ASME standard prior to a risk-informed submittal. In addition, SNC had F&O closure reviews conducted for all open suggestion and finding level F&Os in accordance with Appendix X of NEI 05-04/07-12/12-13. The resulting fire risk assessment model is used as the analytical tool to perform Fire Risk Evaluations during the transition process.

Section 4.5.1.1 describes the Internal Events PRA model. Section 4.5.1.2 describes the Fire PRA model. Section 4.5.1.3 describes the results and resolution of the peer review of the Fire PRA, and Section 4.5.1.4 describes insights gained from the Fire PRA.

4.5.1.1 Internal Events PRA

The HNP base internal events U1R4V1 and U2R4V2 PRA models were the starting point for the Fire PRA. The HNP Full-Power Internal Events (including Internal Flooding) PRA has undergone a RG 1.200, Revision 2, Peer Review against the ASME PRA standard ASME/ANS RA-Sa-2009 by a team of knowledgeable industry (vendor and utility) personnel. The Peer review was performed in November 2009 using the NEI 05-04 process, the ASME/ANS PRA Standard and Regulatory Guide 1.200, Rev. 2. The Peer Review was a full-scope review of the Technical Elements of the internal events, at-power PRA. In the course of this review, 40 new Facts and Observations (F&Os) were prepared, including one "Best Practices" and 25 "Findings".

The F&Os resulting from the IEIF PRA peer review were closed in July 2017 using the February 2017 version of Appendix X of NEI 05-04/07-12/12-13 (ADAMS ML17079A427). Four findings remained open after this review. Attachment U provides the discussion on Full Power Internal Events quality and addresses the applicability of each remaining finding for the Fire PRA application.

4.5.1.2 Fire PRA

The internal events PRA was modified to capture the effects of fire both as an initiator of an event and as a potential failure mode of affected circuits and individual targets. The Unit 1 and Unit 2 Fire PRA Models were developed using the guidance for Fire PRA development in NUREG/CR-6850, including Supplement 1 and subsequent frequently

asked questions. The FPRA methodology also incorporated the latest guidance from NUREG-2169, NUREG/CR-7150, NUREG-2178, and NUREG-1921 R1. Attachment V lists the calculations associated with each of the FPRA model tasks. H-RIE-FIREPRA-U00-014, "Hatch Fire PRA Task 14, Fire Risk Quantification," is the summary calculation for the Hatch Fire PRA models and contains the model results and model files.

The Fire PRA quality and results are discussed in the subsequent sections and in Attachments V and W, respectively.

Fire Models Utilization in the Application

Fire modeling was performed as part of the Fire PRA development (NFPA 805 Section 4.2.4.2). RG 1.205, Regulatory Position 4.2 and Section 5.1.2 of NEI 04-02, provide guidance to identify fire models that are acceptable to the NRC for plants implementing a risk-informed, performance-based licensing basis.

The acceptability of the use of the fire models used in the development of the HNP Fire PRA is included in Attachment J.

4.5.1.3 Results of Fire PRA Peer Review

The HNP Fire PRA was peer reviewed against the requirements of ASME/ANS RA-Sa-2009, Part 4 using the guidance in RG 1.200 Rev. 2 and NEI 07-12. The review was issued by the BWROG in June of 2016. The summary of the peer review findings exhibited the following statistics for the evaluation of elements to the combined Fire PRA Standard. For the Hatch 1/2 FPRA, 93.7% of the applicable SRs were assessed at Capability Category II or higher, including 83.5% of applicable SRs being assessed at Capability Category III. The HNP 1/2 FPRA had 1.2% of the applicable SRs assessed at the Capability Category level 1 and concluding that 5.1% of the applicable SRs be assessed as not MET. The F&Os resulting from the Fire PRA peer review were closed using the February 2017 version of Appendix X of NEI 05-04/07-12/12-13 (ADAMS ML17079A427). No open findings remain.

Attachment V provides the discussion on Fire PRA quality.

4.5.1.4 Risk Insights

Risk insights were documented as part of the development of the Fire PRA. The total plant fire CDF/LERF was derived using the NUREG/CR-6850 methodology for Fire PRA development and is useful in identifying the areas of the plant where fire risk is greatest. A review of the fire initiating events that represent 1% of the calculated fire risk or greater are included in Attachment W.

4.5.2 Performance-Based Approaches

NFPA 805 outlines the approaches for performing performance-based analyses. As specified in Section 4.2.4, there are generally two types of analyses performed for the performance-based approach:

- Fire Modeling (NFPA 805 Section 4.2.4.1).
- Fire Risk Evaluation (NFPA 805 Section 4.2.4.2).

4.5.2.1 Fire Modeling Approach

The fire modeling approach was not utilized for the transition.

4.5.2.2 Fire Risk Approach

Overview of Evaluation Process

The Fire Risk Evaluations were completed as part of the HNP 1/2 NFPA 805 transition. These Fire Risk Evaluations were developed using the process described below. This methodology is based upon the requirements of NFPA 805, industry guidance in NEI 04-02, and RG 1.205. These are summarized in Table 4-1.

Table 4-1 Fire Risk Evaluation Guidance Summary Table

Document	Section(s)	Topic
NFPA 805	2.2(h), 4.2.4, A.2.2(h), A.2.4.4, D.5	Change Evaluation (2.2(h), 2.2.9, 2.4.4 A.2.2(h), A.2.4.4, D.5) Risk of Recovery Actions (4.2.4) Use of Fire Risk Evaluation (4.2.4.2)
NEI 04-02 Revision 2	4.4, 5.3, Appendix B, Appendix I, Appendix J	Change Evaluation, Change Evaluation Forms (App. I), No specific discussion of Fire Risk Evaluation
RG 1.205 Revision 1	C.2.2.4, C.2.4, C.3.2	Risk Evaluations (C.2.2.4) Recovery Actions (C.2.4)

During the transition to NFPA 805, variances from the deterministic approach in Section 4.2.3 of NFPA 805 were evaluated using a Fire Risk Evaluation per Section 4.2.4.2 of NFPA 805. A Fire Risk Evaluation was performed for each fire area containing variances from the deterministic requirements of Section 4.2.3 of NFPA 805 (VFDRs).

If the Fire Risk Evaluation meets the acceptance criteria, this is confirmation that a success path effectively remains free of fire damage and that the performance-based approach is acceptable per Section 4.2.4.2 of NFPA 805.

The Fire Risk Evaluation process consists of the following steps (Figure 4-7 depicts the Fire Risk Evaluation process used during transition. This is generally based on FAQ 08-0054 Revision 1:

Step 1 – Preparation for the Fire Risk Evaluation.

- Definition of the Variances from the Deterministic Requirements. The definition of the VFDR includes a description of problem statement and the section of NFPA 805 that is not met, type of VFDR (e.g., separation issue or degraded fire protection system), and proposed evaluation per applicable NFPA 805 section.
- Preparatory Evaluation – Fire Risk Evaluation Team Review. Using the information obtained during the development of the NEI 04-02 B-3 Table and the Fire PRA, a team review of the VFDR was performed. Depending on the scope and complexity of the VFDR, the team may include the Safe shutdown/NSCA Engineer, the Fire Protection Engineer, and the Fire PRA Engineer. The purpose and objective of this team review was to address the following;

- Review of the Fire PRA modeling treatment of VFDR
- Ensure discrepancies were captured and resolved

Step 2 – Performed the Fire Risk Evaluation

- The Evaluator coordinated as necessary with the Safe shutdown/NSCA Engineer, Fire Protection Engineer and Fire PRA Engineer to assess the VFDR using the Fire Risk Evaluation process to perform the following:
  - Change in Risk Calculation with consideration for additional risk of recovery actions and required fire protection systems and features due to fire risk.
  - Fire area change in risk summary

Step 3 – Reviewed the Acceptance Criteria

- The acceptance criteria for the Fire Risk Evaluation consist of two parts. One is quantitatively based and the other is qualitatively based. The quantitative figures of merit are ΔCDF and ΔLERF. The qualitative factors are defense-in-depth and safety margin.
  - Risk Acceptance Criteria. The transition risk evaluation was measured quantitatively for acceptability using the ΔCDF and ΔLERF criteria from RG 1.174, as clarified in RG 1.205 Regulatory Position 2.2.4.
  - Defense-in-Depth. A review of the impact of the change on defense-in-depth was performed, using the guidance NEI 04-02. NFPA 805 defines defense-in-depth as:
    - Preventing fires from starting
    - Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting damage
    - Providing 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.

    In general, the defense-in-depth requirement was considered to be satisfied if the proposed change does not result in a substantial imbalance among these elements (or echelons).

    The review of defense-in-depth was qualitative and addressed each of the elements with respect to the proposed change. Defense-in-depth was performed on a fire area basis.

    Fire protection features and systems relied upon to ensure defense-in-depth were identified as a result of the assessment of defense-in-depth.
  - Safety Margin Assessment. A review of the impact of the change on safety margin was performed. An acceptable set of guidelines for making that assessment is summarized below. Other equivalent acceptance guidelines may also be used.
    - Codes and standards or their alternatives accepted for use by the NRC are met, and

- Safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met, or provides sufficient margin to account for analysis and data uncertainty.

The requirements related to safety margins for the change analysis are described for each of the specific analysis types used in support of the FRE.

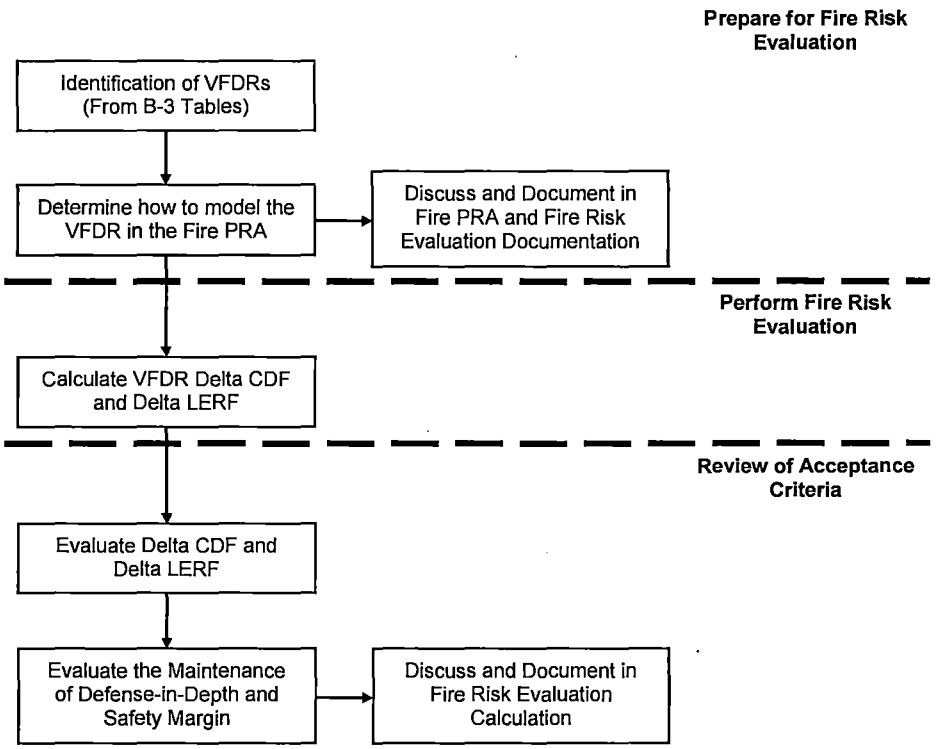

Figure 4-7 – Fire Risk Evaluation Process (NFPA 805 Transition) [Based on FAQ 08-0054 Revision 1]

Results of Evaluation Process

Disposition of VFDRs

The HNP NSCA and the NFPA 805 transition project activities have identified a number of variances from the deterministic requirements of NFPA 805 Section 4.2.3. These variances were dispositioned using the fire risk evaluation process.

Each variance dispositioned using a Fire Risk Evaluation was assessed against the Fire Risk Evaluation acceptance criteria of ΔCDF and ΔLERF; and maintenance of defense-in-depth and safety margin criteria from Section 5.3.5 of NEI 04-02 and RG 1.205. The results of these calculations are summarized in Attachment C.

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

Risk Change Due to NFPA 805 Transition

In accordance with the guidance in RG 1.205, Section C.2.2.4, Risk Evaluations, risk increases or decreases for each fire area using Fire Risk Evaluations and the overall plant should be provided. Note that the risk increase due to the use of recovery actions was included in the risk change for transition for each fire area.

RG 1.205 Section C.2.2.4.2 states in part

> *"The total increase or decrease in risk associated with the implementation of NFPA 805 for the overall plant should be calculated by summing the risk increases and decreases for each fire area (including any risk increases resulting from previously approved recovery actions). The total risk increase should be consistent with the acceptance guidelines in Regulatory Guide 1.174. Note that the acceptance guidelines of Regulatory Guide 1.174 may require the total CDF, LERF, or both, to evaluate changes where the risk impact exceeds specific guidelines. If the additional risk associated with previously approved recovery actions is greater than the acceptance guidelines in Regulatory Guide 1.174, then the net change in total plant risk incurred by any proposed alternatives to the deterministic criteria in NFPA 805, Chapter 4 (other than the previously approved recovery actions), should be risk neutral or represent a risk decrease."*

The risk increases and decreases are provided in Attachment W.

4.6 Monitoring Program

4.6.1 Overview of NFPA 805 Requirements and NEI 04-02 Guidance on the NFPA 805 Fire Protection System and Feature 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."*

As part of the transition review, the adequacy of the inspection and testing program to address fire protection systems and equipment within the plant inspection and the compensatory measures programs should be reviewed. In addition, the adequacy of the plant corrective action program in determining the causes of equipment and programmatic failures and minimizing their recurrence should also be reviewed as part of the transition to a risk-informed, performance-based licensing basis.

4.6.2 Overview of Post-Transition NFPA 805 Monitoring Program

This section describes the process that will be utilized to implement the post-transition NFPA 805 monitoring program. The monitoring program will be implemented after the safety evaluation issuance as part of the fire protection program transition to NFPA 805. See Attachment S, Implementation Item IMP-14. The monitoring process is comprised of four phases.

- Phase 1 – Scoping
- Phase 2 – Screening Using Risk Criteria
- Phase 3 – Risk Target Value Determination
- Phase 4 – Monitoring Implementation

Figure 4-8 provides detail on the Phase 1 and 2 processes.

The results of these phases will be documented in the HNP Monitoring Program Engineering Evaluation developed during implementation.

Phase 1 – Scoping

In order to meet the NFPA 805 requirements for monitoring, the following categories of SSCs and programmatic elements will be reviewed during the implementation phase for inclusion in the NFPA 805 monitoring program:

- Structures, Systems, and Components required to comply with NFPA 805, specifically:
  - Fire protection systems and features
    - Required by the Nuclear Safety Capability Assessment
    - Modeled in the Fire PRA
    - Required by Chapter 3 of NFPA 805
  - Nuclear Safety Capability Assessment equipment
    - Nuclear safety equipment
    - Fire PRA equipment
    - NPO equipment
  - Structures, Systems, and Components relied upon to meet radioactive release criteria
- Fire Protection Programmatic Elements

Phase 2 – Screening Using Risk Criteria

The equipment from Phase 1 scoping will be screened to determine the appropriate level of NFPA 805 monitoring. As a minimum, the SSCs identified in Phase 1 should be part of an inspection and test program and system/program health program. If not in the current program, the SSCs will be added in order to assure that the criteria can be met reliably.

The following screening process will be used to determine those SSCs that may require additional monitoring beyond normal inspection and test program and system/program health reporting and will be documented in the HNP Monitoring Program Engineering Evaluation to be developed during implementation.

1. Fire Protection Systems and Features

Those fire protection systems and features identified in Phase 1 are candidates for additional monitoring in the NFPA 805 program commensurate with risk significance.

Risk significance is determined at the component, programmatic element, and/or functional level on an individual fire area basis. Compartments smaller than fire areas

may be used provided the compartments are independent (i.e., share no fire protection SSCs). If compartments smaller than fire areas are used the basis will be documented in the HNP Monitoring Program Engineering Evaluation to be developed during implementation.

The Fire PRA is used to establish the risk significance based on the following screening criteria:

Risk Achievement Worth (RAW) of the monitored parameter ≥ 2.0

(AND) either

Core Damage Frequency (CDF) x (RAW) ≥ 1.0E-7 per year

(OR)

Large Early Release Frequency (LERF) x (RAW) ≥ 1.0E-8 per year

CDF, LERF, and $RAW_{(monitored\ parameter)}$ are calculated for each fire area. The 'monitored parameter' will be established at a level commensurate with the amenability of the parameter to risk measurement (e.g., a fire barrier may be more conducive to risk measurement than an individual barrier penetration).

Fire protections systems and features that meet or exceed the criteria identified above are considered High Safety Significant (HSS) will be included in the HNP NFPA 805 Monitoring Program, which is a new program similar to the Maintenance Rule, and will be documented in the HNP Monitoring Program Engineering Evaluation. The remaining required fire protection systems and features will be monitored via the existing inspection and test procedures and/or in the existing system / program health reporting as described NMP-ES-002, "System Monitoring And Health Reporting", and NMP-ES-009-002, "Engineering Programs - Health Reports And Notebooks".

2. Nuclear Safety Capability Assessment Equipment

Required NSCA equipment, except the NPO scope, identified in Phase 1 will be screened for safety significance using the Fire PRA and the Maintenance Rule guidelines differentiating HSS equipment from Low Safety Significant (LSS) equipment. The screening will also ensure that the Maintenance Rule functions are consistent with the required functions of the NSCA equipment.

HSS NSCA equipment not currently monitored in the Maintenance Rule will be included in the Maintenance Rule. All NSCA equipment that are not HSS are considered LSS and need not be included in the monitoring program.

For non-power operational modes, the qualitative use of fire prevention to manage fire risk during Higher Risk Evolutions does not lend itself to quantitative risk measurement. Therefore, fire risk management effectiveness is monitored programmatically similar to combustible material controls and other fire prevention programs. Additional monitoring beyond inspection and test programs and system/program health reporting is not considered necessary.

3. SSCs Relied upon for Radioactive Release Criteria

The evaluations performed to meet the radioactive release performance criteria are qualitative in nature. The SSCs relied upon to meet the radioactive release performance criteria are not amenable to quantitative risk measurement. Additionally, since 10 CFR Part 20 limits (which are lower than releases due to core damage and containment breach) for radiological effluents are not being exceeded, equipment relied upon to meet the radioactive release performance criteria is considered inherently low risk. Therefore, additional monitoring beyond inspection and test programs and system/program health reporting is not considered necessary.

4. Fire Protection Programmatic Elements

Monitoring of programmatic elements is required in order to "assess the performance of the fire protection program in meeting the performance criteria". These programs form the bases for many of the analytical assumptions used to evaluate compliance with NFPA 805 requirements Programmatic aspects include:

- Transient combustible control; transient exclusion zones
- Hot work control; administrative controls
- Impairment and compensatory measures including program compliance and effectiveness
- Fire brigade effectiveness

Monitoring of programmatic elements is more qualitative in nature since the programs do not lend themselves to the numerical methods of reliability and availability. Therefore, monitoring is conducted using the existing system and program health programs. Fire protection health reports, self-assessments, regulator and insurance company reports provide inputs to the monitoring program.

Phase 3 – Risk Target Value Determination

Phase 3 establishes the target values for reliability and availability for the fire protection systems and features that met or exceeded the screening criteria and the HSS NSCA equipment established in Phase 2.

Target values for reliability and availability for the fire protection systems and features are established at the component level, program level, or functionally through the use of the pseudo system or 'performance monitoring group' concept. The actual action level is determined based on the number of component, program or functional failures within a sufficiently bounding time period (~2-3 operating cycles). In addition, the EPRI Technical Report (TR) 1006756, "Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features" may be used as input for establishing reliability targets, action levels, and monitoring frequency.

Since the HSS NSCA equipment have been identified using the Maintenance Rule guidelines, the associated equipment specific performance criteria will be established as in the Maintenance Rule, provided the criteria are consistent with Fire PRA assumptions.

When establishing the action level threshold for reliability and availability, the action level will be no lower than the Fire PRA assumptions. Adverse trends and

unacceptable levels of availability, reliability, and performance will be reviewed against established action levels. The monitoring program failure criteria and action level targets will be documented in the HNP Monitoring Program Engineering Evaluation to be developed during implementation.

Note that fire protection systems and features, NSCA equipment, SSCs required to meet the radioactive release criteria, and fire protection program elements that do not meet the screening criteria in Phase 2 will be included in the existing inspection and test programs and the system and program health programs. Reliability and availability criteria will not be assigned.

Phase 4 – Monitoring Implementation

Phase 4 is the implementation of the monitoring program, once the monitoring scope and criteria are established. Monitoring consists of periodically gathering, trending, and evaluating information pertinent to the performance, and/or availability of the equipment and comparing the results with the established goals and performance criteria to verify that the goals are being met. Results of monitoring activities will be analyzed in a timely manner to assure that appropriate action is taken. The corrective action process will be used to address performance of fire protection and nuclear safety SSCs that do not meet performance criteria.

For fire protection systems and features and NSCA HSS equipment that are monitored, unacceptable levels of availability, reliability, and performance will be reviewed against the established action levels. If an action level is triggered, corrective action in accordance with NMP-GM-002, "Corrective Action Program," will be initiated to identify the negative trend. A corrective action plan will then be developed to ensure the performance returns to the established level.

When applicable, a sensitivity study can be performed to determine the margin below the action level that still provides acceptable Fire PRA results to help prioritize corrective actions if the action level is reached.

A periodic assessment will be performed (e.g., at a frequency of approximately every two to three operating cycles), taking into account, where practical, industry wide operating experience. This will be conducted as part of self-assessments performed in accordance with NMP-AD-027, "NRC Inspection Preparation and Response." Issues that will be addressed include:

- Review systems with performance criteria. Do performance criteria still effectively monitor the functions of the system? Do the criteria still monitor the effectiveness of the fire protection and NSCA systems?
- Have the supporting analyses been revised such that the performance criteria are no longer applicable or new fire protection and NSCA SSCs, programmatic elements and/or functions need to be in scope?
- Based on the performance during the assessment period, are there any trends in system performance that should be addressed that are not being addressed?

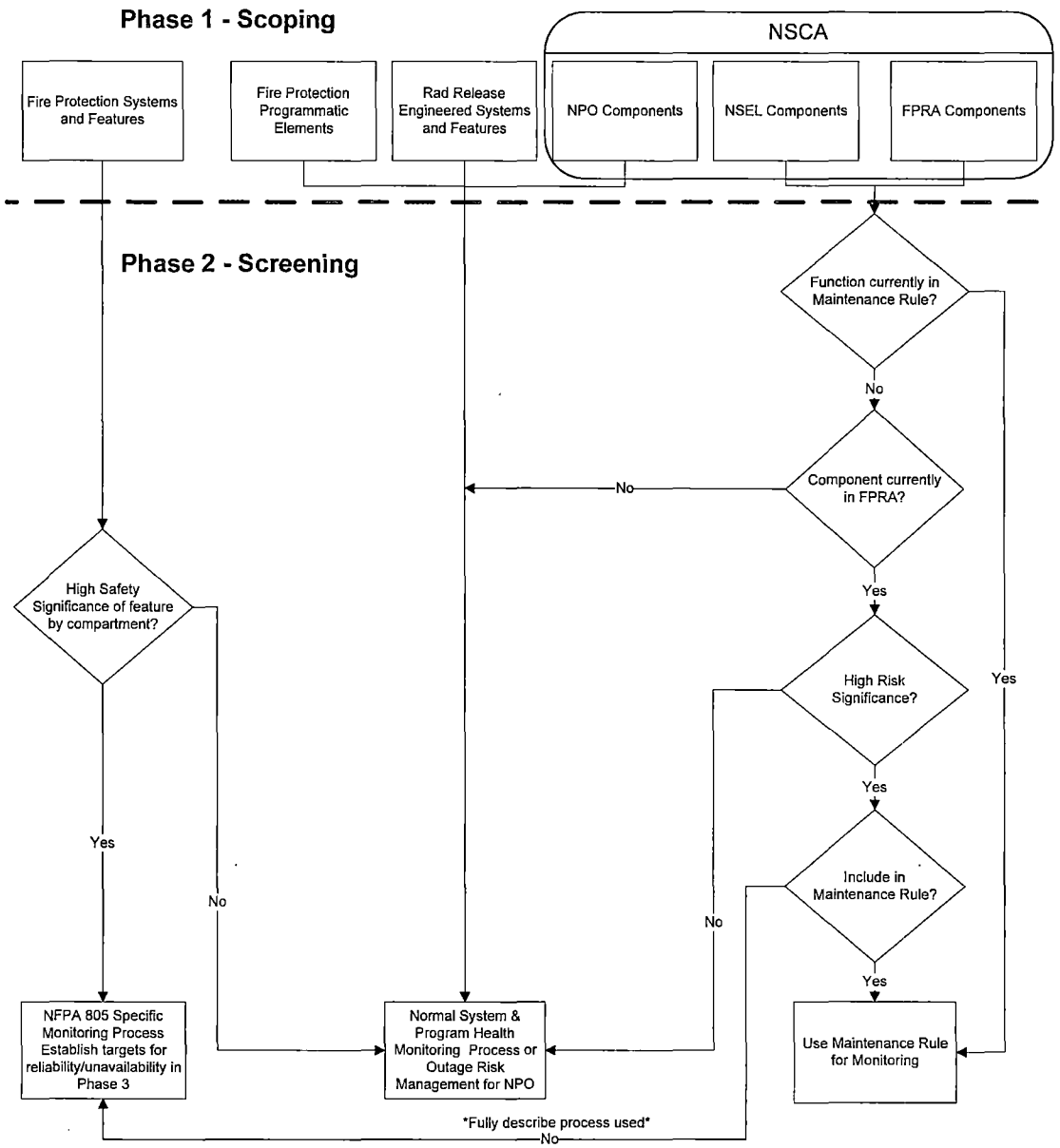

Figure 4-8 – NFPA 805 Monitoring – Scoping and Screening

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, HNP has documented analyses to support compliance with 10 CFR 50.48(c). The analyses are being performed in accordance with SNC's processes for ensuring assumptions are clearly defined, that results are easily

understood, that results are clearly and consistently described, and that sufficient detail is provided to allow future review of the entire analyses.

Analyses, as defined by NFPA 805 Section 2.4, performed to demonstrate compliance with 10 CFR 50.48(c) will be maintained for the life of the plant and organized to facilitate review for accuracy and adequacy. Note these analyses do not include items such as periodic tests, hot work permits, fire impairments, etc.

The Fire Protection Design Basis Document described in Section 2.7.1.2 of NFPA 805 and necessary supporting documentation described in Section 2.7.1.3 of NFPA 805 will be created as part of transition to 10 CFR 50.48(c) to ensure program implementation following receipt of the safety evaluation. See Attachment S, Table S-3, Implementation Item IMP-15. Appropriate cross references will be established to supporting documents as required by SNC processes. Figure 4-9 depicts the planned post-transition documentation and relationships.

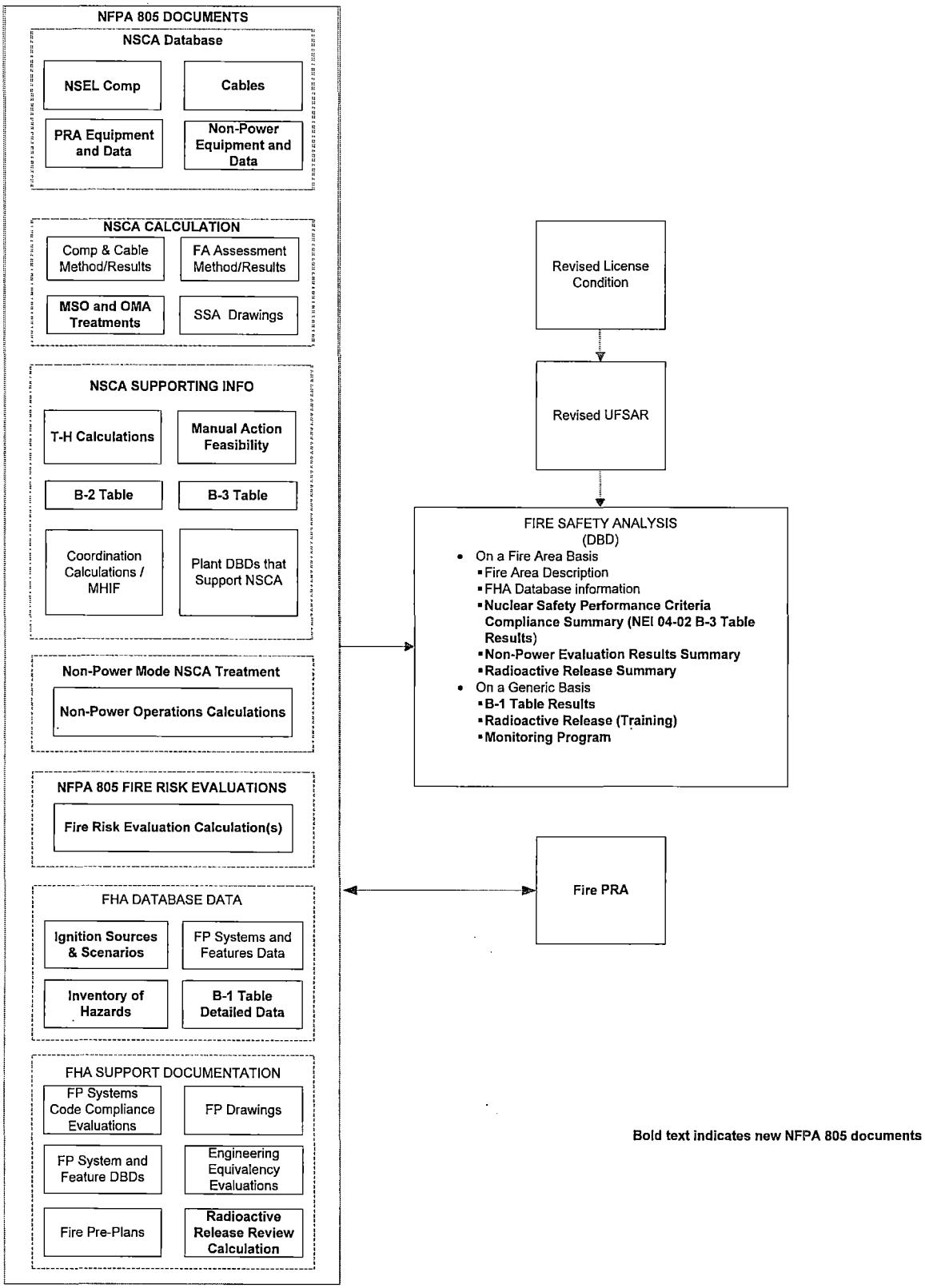

Figure 4-9 – NFPA 805 Planned Post-Transition Documents and Relationships

4.7.2 Compliance with Configuration Control Requirements in Section 2.7.2 and 2.2.9 of NFPA 805

Program documentation established, revised, or utilized in support of compliance with 10 CFR 50.48(c) is subject to SNC configuration control processes that meet the requirements of Section 2.7.2 of NFPA 805. This includes the appropriate procedures and configuration control processes for ensuring that changes impacting the fire protection program are reviewed appropriately. The RI-PB post transition change process methodology is based upon the requirements of NFPA 805, and industry guidance in NEI 04-02, and RG 1.205. These requirements are summarized in Table 4-2.

Table 4-2 Change Evaluation Guidance Summary Table

Document	Section(s)	Topic
NFPA 805	2.2(h), 2.2.9, 2.4.4, A.2.2(h), A.2.4.4, D.5	Change Evaluation
NEI 04-02	5.3, Appendix B, Appendix I, Appendix J	Change Evaluation, Change Evaluation Forms (Appendix I)
RG 1.205	C.2.2.4, C.3.1, C.3.2, C.4.3	Risk Evaluation, Standard License Condition, Change Evaluation Process, Fire PRA

The Plant Change Evaluation Process consists of the following 4 steps and is depicted in Figure 4-10:

- Defining the Change
- Performing the Preliminary Risk Screening:
- Performing the Risk Evaluation
- Evaluating the Acceptance Criteria

Change Definition

The Change Evaluation process begins by defining the change or altered condition to be examined and the baseline configuration as defined by the Design Basis and Licensing Basis (NFPA 805 Licensing Basis post-transition).

1. The baseline is defined as that plant condition or configuration that is consistent with the Design Basis and Licensing Basis (NFPA 805 Licensing Basis post-transition).
2. The changed or altered condition or configuration that is not consistent with the Design Basis and Licensing Basis is defined as the proposed alternative.

Preliminary Risk Review

Once the definition of the change is established, a screening is then performed to identify and resolve minor changes to the fire protection program. This screening is consistent with fire protection regulatory review processes in place at nuclear plants under traditional licensing bases. This screening process is modeled after the NEI 02-03 process. This process will address most administrative changes (e.g., changes to the combustible control program, organizational changes, etc.).

The characteristics of an acceptable screening process that meets the "assessment of the acceptability of risk" requirement of Section 2.4.4 of NFPA 805 are:

- The quality of the screen is sufficient to ensure that potentially greater than minimal risk increases receive detailed risk assessments appropriate to the level of risk.
- The screening process must be documented and be available for inspection by the NRC.
- The screening process does not pose undue evaluation or maintenance burden.

If any of the above is not met, proceed to the Risk Evaluation step.

Risk Evaluation

The screening is followed by engineering evaluations that may include fire modeling and risk assessment techniques. The results of these evaluations are then compared to the acceptance criteria. Changes that satisfy the acceptance criteria of NFPA 805 Section 2.4.4 and the license condition can be implemented within the framework provided by NFPA 805. Changes that do not satisfy the acceptance criteria cannot be implemented within this framework. The acceptance criteria require that the resultant change in CDF and LERF be consistent with the license condition. The acceptance criteria also include consideration of defense-in-depth and safety margin, which would typically be qualitative in nature.

The risk evaluation involves the application of fire modeling analyses and risk assessment techniques to obtain a measure of the changes in risk associated with the proposed change. In certain circumstances, an initial evaluation in the development of the risk assessment could be a simplified analysis using bounding assumptions provided the use of such assumptions does not unnecessarily challenge the acceptance criteria discussed below.

Acceptability Determination

The Change Evaluations are assessed for acceptability using the ΔCDF (change in core damage frequency) and ΔLERF (change in large early release frequency) criteria from the license condition. The proposed changes are also assessed to ensure they are consistent with the defense-in-depth philosophy and that sufficient safety margins were maintained.

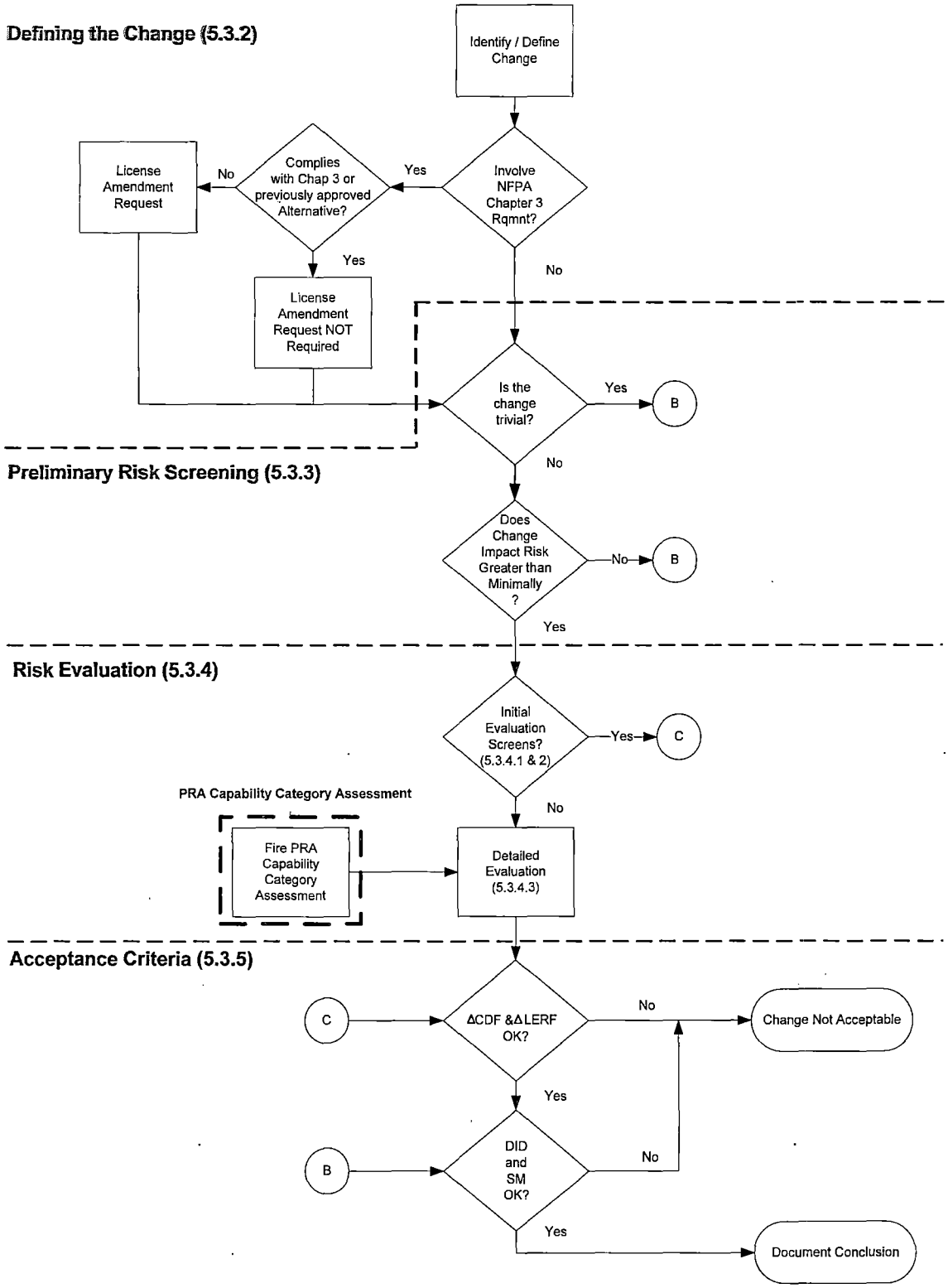

Figure 4-10 Plant Change Evaluation [NEI 04-02 Figure 5-1]
Note references in Figure refer to NEI 04-02 Sections

The HNP Fire Protection Program configuration is defined by the program documentation. To the greatest extent possible, the existing configuration control processes for modifications, calculations and analyses, and Fire Protection Program License Basis Reviews, will be utilized to maintain configuration control of the Fire Protection program documents. The configuration control procedures which govern the various HNP documents and databases that currently exist will be revised to reflect the new NFPA 805 licensing bases requirements.

Several NFPA 805 document types such as: NSCA Supporting Information, Non-Power Mode NSCA Treatment, etc., generally require new control procedures and processes to be developed since they are new documents and databases created as a result of the transition to NFPA 805. The new procedures will be modeled after the existing processes for similar types of documents and databases. System level design basis documents will be revised to reflect the NFPA 805 role that the system components now play.

The process for capturing the impact of proposed changes to the plant on the Fire Protection Program will continue to be a multiple step review. The first step of the review is an initial screening for process users to determine if there is a potential to impact the Fire Protection Program as defined under NFPA 805 through a series of screening questions/checklists contained in one or more procedures depending upon the configuration control process being used. Reviews that identify potential Fire Protection Program impacts will be sent to qualified individuals (Fire Protection, Safe Shutdown/NSCA, Fire PRA) to ascertain the program impacts, if any. If Fire Protection Program impacts are determined to exist as a result of the proposed change, the issue would be resolved by one of the following:

- Deterministic Approach: Comply with NFPA 805 Chapter 3 and 4.2.3 requirements.
- Performance-Based Approach: Utilize the NFPA 805 change process developed in accordance with NEI 04-02, RG 1.205, and the NFPA 805 fire protection license condition to assess the acceptability of the proposed change. This process would be used to determine if the proposed change could be implemented "as-is" or whether prior NRC approval of the proposed change is required.

This process follows the requirements in NFPA 805 and the guidance outlined in RG 1.174 which requires the use of qualified individuals, procedures that require calculations be subject to independent review and verification, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when errors are discovered. See Attachment S, Table S-3, Implementation Item IMP-16 and IMP-17.

4.7.3 Compliance with Quality Requirements in Section 2.7.3 of NFPA 805

Fire Protection Program Quality

SNC will maintain the existing Fire Protection Quality Assurance program.

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.

Fire PRA Quality

Configuration control of the Fire PRA model will be maintained by integrating the Fire PRA model into the existing processes used to ensure configuration control of the internal events PRA model. This process complies with Section 1-5 of the ASME PRA Standard and ensures that HNP maintains an as-built, as-operated PRA model of the plant. The process has been peer reviewed. Quality assurance of the Fire PRA is assured via the same processes applied to the internal events model.

This process follows the guidance outlined in RG 1.174 which requires the use of qualified individuals, procedures that require calculations be subject to independent review and verification, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when errors are discovered. Although the entire scope of the formal 10 CFR 50 Appendix B program is not applied to the PRA models or processes in general, often parts of the program are applied as a convenient method of complying with the requirements of RG 1.174. For instance, the procedure which addresses independent review of calculations for 10 CFR 50 Appendix B is applied to the PRA model calculations, as well.

With respect to Quality Assurance Program requirements for independent reviews of calculations and evaluations, those existing requirements for Fire Protection Program documents will remain unchanged. SNC specifically requires that the calculations and evaluations in support of the NFPA 805 LAR, exclusive of the Fire PRA, be performed within the scope of the QA program which requires independent review as defined by SNC procedures. As recommended by NUREG/CR-6850, the sources of uncertainty in the Fire PRA were identified and specific parameters were analyzed for sensitivity in support of the NFPA 805 Fire Risk Evaluation process.

Specifically, with regard to uncertainty, an uncertainty and sensitivity matrix was developed and included with the Hatch Fire PRA Task 15, Uncertainty and Sensitivity Analysis (Calculation H-RIE-FIREPRA-U00-015). In addition, sensitivity to uncertainty associated with specific Fire PRA parameters was quantitatively addressed in this report.

While the removal of conservatism inherent in the Fire PRA is a long-term goal, the Fire PRA results were deemed sufficient for evaluating the risk associated with this application. While SNC continues to strive toward a more "realistic" estimate of fire risk, use of mean values continues to be the best estimate of fire risk. During the Fire Risk Evaluation process, the uncertainty and sensitivity associated with specific Fire PRA parameters were considerations in the evaluation of the change in risk relative to the applicable acceptance thresholds.

Specific Requirements of NFPA 805 Section 2.7.3

The following discusses how the requirements of NFPA 805 Section 2.7.3 were met during the transition process. Post-transition, SNC will perform work in accordance with NFPA 805 Section 2.7.3 requirements.

NFPA 805 Section 2.7.3.1 – Review

Analyses, calculations, and evaluations performed in support of compliance with 10 CFR 50.48(c) are performed in accordance with SNC 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) were 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 applied 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 use and apply engineering analysis and numerical methods in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by Section 2.7.3.4 of NFPA 805.

During the transition to 10 CFR 50.48(c), work was performed in accordance with the quality requirements of Section 2.7.3 of NFPA 805. Personnel who used and applied engineering analysis and numerical methods (e.g., fire modeling) in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by NFPA 805 Section 2.7.3.4.

Post-transition, for personnel performing fire modeling or Fire PRA development and evaluation, SNC will develop and maintain qualification requirements for individuals assigned various tasks. Position Specific Guides will be developed to identify and document required training and mentoring to ensure individuals are appropriately qualified per the requirements of NFPA 805 Section 2.7.3.4 to perform assigned work. See Attachment S, Table S-3, Implementation Item IMP-18.

NFPA 805 Section 2.7.3.5 – Uncertainty Analysis

Uncertainty analyses were performed as required by 2.7.3.5 of NFPA 805 and the results were considered in the context of the application. This is of particular interest in fire modeling and Fire PRA development. Note: 10 CFR 50.48(c)(2)(iv) states that NFPA 805 Section 2.7.3.5 is not required for the deterministic approach because conservatism is included in the deterministic criteria.

4.8 Summary of Results

4.8.1 Results of the Fire Area Review

A summary of the NFPA 805 compliance basis and the required fire protection systems and features is provided in Attachment C, Table C-2. The table provides the following information from the NEI 04-02 Table B-3:

- Fire Area / Fire Zone: Fire Area/Zone Identifier.

- Description: Fire Area/Zone Description.
- NFPA 805 Regulatory Basis: Post-transition NFPA 805 Chapter 4 compliance basis.
- Required Fire Protection System / Feature: Detection / suppression required in the Fire Area based on NFPA 805 Chapter 4 compliance. Other required features may include Electrical Raceway Fire Barrier Systems (ERFBS), fire barriers, etc. The documentation of required fire protection systems and features does not include the documentation of the fire area boundaries. Fire area boundaries are required and documentation of the fire area boundaries has been performed as part of reviews of engineering evaluations, licensing actions, or as part of the reviews of the NEI 04-02 Table B-1 process. The basis for the requirement of the fire protection system / feature is designated as follows:
    - S – Separation Criteria: Systems/Features required for Chapter 4 Separation Criteria in Section 4.2.3
    - L – Licensing Action Criteria: Systems/Features required for acceptability of NRC approved Licensing Action (i.e., Exemptions/Deviations/Safety Evaluations) (Section 2.2.7)
    - E – EEEE: Systems/Features required for acceptability of Existing Engineering Equivalency (Section 2.2.7)
    - R – Risk Criteria: Systems/Features required to meet the Risk Criteria for the Performance-Based Approach (Section 4.2.4)
    - D – Defense-in-depth Criteria: Systems/Features required to maintain adequate balance of Defense-in-Depth for a Performance-Based Approach (Section 4.2.4)

Attachment W contains the results of the Fire Risk Evaluations, additional risk of recovery actions, and the change in risk on a fire area basis.

4.8.2 Plant Modifications and Items to be Completed During the Implementation Phase

Planned modifications, studies, and evaluations to comply with NFPA 805 are described in Attachment S.

The Fire PRA model represents the as-built, as-operated and maintained plant as it will be configured at the completion of the transition to NFPA 805. The Fire PRA model includes credit for the planned implementation of the modifications identified in Attachment S. Following installation of modifications and the as-built installation details, additional refinements surrounding the modification may need to be incorporated into the Fire PRA model. However, these changes are not expected to be significant. No other significant plant changes are outstanding with respect to their inclusion in the Fire PRA model. See Attachment S, Table S-3, Implementation Item IMP-19.

4.8.3 Supplemental Information –Other Licensee Specific Issues

There are no HNP specific issues that warrant additional treatment in this section.

5.0 REGULATORY EVALUATION

5.1 Introduction – 10 CFR 50.48

On July 16, 2004 the NRC amended 10 CFR 50.48, Fire Protection, to add a new subsection, 10 CFR 50.48(c), which establishes alternative fire protection requirements. 10 CFR 50.48 endorses, with exceptions, NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition (NFPA 805), as a voluntary alternative for demonstrating compliance with 10 CFR 50.48 Section (b), Appendix R, and Section (f), Decommissioning.

The voluntary adoption of 10 CFR 50.48(c) by HNP does not eliminate the need to comply with 10 CFR 50.48(a) and 10 CFR 50, Appendix A, GDC 3, Fire Protection. The NRC addressed the overall adequacy of the regulations during the promulgation of 10 CFR 50.48(c) (Reference FR Notice 69 FR 33536 dated June 16, 2004, ML041340086).

> *"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 Section 1.5.1 criteria include provisions for ensuring that reactivity control, inventory and pressure control, decay heat removal, vital auxiliaries, and process monitoring are achieved and maintained.*
>
> *This methodology specifies a process to identify the fire protection systems and features required to achieve the nuclear safety performance criteria in Section 1.5 of NFPA 805. 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 must meet any applicable requirements of NFPA 805, Chapter 3. Having identified the required fire protection systems and features, the licensee selects either a deterministic or performance-based approach to demonstrate that the performance criteria are satisfied. This process satisfies the GDC 3 requirement to design and locate SSCs important to safety to minimize the probability and effects of fires and explosions." (Reference FR Notice 69 FR 33536 dated June 16, 2004, ML041340086)*

The new rule provides actions that may be taken to establish compliance with 10 CFR 50.48(a), which requires each operating nuclear power plant to have a fire protection program plan that satisfies GDC 3, as well as specific requirements in that section. The transition process described in 10 CFR 50.48(c)(3)(ii) provides, in pertinent parts, that a licensee intending to adopt the new rule must, among other things, "modify the fire protection plan required by paragraph (a) of that section to reflect the licensee's decision to comply with NFPA 805." Therefore, to the extent that the

contents of the existing fire protection program plan required by 10 CFR 50.48(a) are inconsistent with NFPA 805, the fire protection program plan must be modified to achieve compliance with the requirements in NFPA 805. All other requirements of 10 CFR 50.48 (a) and GDC 3 have corresponding requirements in NFPA 805.

A comparison of the current requirements in Appendix R with the comparable requirements in Section 3 of NFPA 805 shows that the two sets of requirements are consistent in many respects. This was further clarified in NEI 04-02, 10 CFR 50.48(a) and GDC 3 clarification (ML081400292). The following tables 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.

10 CFR 50.48(a)

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

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 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 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 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 NEI 04-02 B-1 and B-3 Tables
(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 3.3.1 and 3.4 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 NEI 04-02 B-1 and B-3 Tables
(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 NEI 04-02 B-3 Table
(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 (Analyses, as defined by NFPA 805 2.4, performed to demonstrate compliance with this standard) be maintained for the life of the plant. See the Quality Assurance Topical Report for HNP Record Retention.

Table 5-1 10 CFR 50.48(a) – Applicability/Compliance Reference	
10 CFR 50.48(a) Section(s)	**Applicability/Compliance Reference**
(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. HNP is licensed under 10 CFR 50.

General Design Criterion 3

Table 5-2 GDC 3 – Applicability/Compliance Reference	
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 NEI 04-02 B-1 and B-3 Tables
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 3.3.2, 3.3.3, 3.3.4, 3.11.4 NEI 04-02 B-1 Table
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 NEI 04-02 B-1 and B-3 Tables
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 NEI 04-02 Table B-3

10 CFR 50.48(c)

Table 5-3 10 CFR 50.48(c) – Applicability/Compliance Reference	
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.	General Information. NFPA 805 (2001 edition) is the edition used.
(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 used.
(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 LAR.

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

10 CFR 50.48(c) Section(s)	Applicability/Compliance Reference
(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 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 methodology.
(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 methodology.
(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 complies with a flame propagation test that was found acceptable to the NRC as documented in Attachment A.
(vi) Water supply and distribution. The italicized exception to Section 3.6.4 is not endorsed. Licensees who wish to use the exception to Section 3.6.4 must submit a request for a license amendment in accordance with paragraph (c)(2)(vii) of this section.	HNP "Complies by Previous NRC Approval" as documented in Attachment A.
(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 use of performance-based methods for NFPA 805 Chapter 3 is requested. See Attachment L.
(3) *Compliance with NFPA 805.*	See below

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

10 CFR 50.48(c) Section(s)	Applicability/Compliance Reference
(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 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 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: (i) Satisfy the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (ii) Maintain safety margins; and (iii) Maintain fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).	No risk-informed or performance-based alternatives to compliance with NFPA 805 (per 10 CFR 50.48(c)(4)) were utilized. See Attachment P.

5.2 Regulatory Topics

5.2.1 License Condition Changes

The current HNP fire protection license conditions 2.C(3) for Unit 1 and 2.C(3)a for Unit 2 is being replaced with the standard license condition based upon Regulatory Position 3.1 of RG 1.205, as shown in Attachment M.

5.2.2 Technical Specifications

HNP conducted a review of the Technical Specifications to determine which Technical Specifications are required to be revised, deleted, or superseded. HNP determined that the changes to the Technical Specifications and applicable justification listed in Attachment N are adequate for the HNP 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. A review was also performed to ensure that compliance with the physical protection requirements, security orders, and adherence to those commitments applicable to the plant are maintained. A discussion of affected orders and exemptions is included in Attachment O.

5.3 Regulatory Evaluations

5.3.1 No Significant Hazards Consideration

A written evaluation of the significant hazards consideration of a proposed license amendment is required by 10 CFR 50.92. 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:

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

This evaluation is contained in Attachment Q.

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. HNP has evaluated the proposed amendment and determined that it involves no significant hazards consideration.

5.3.2 Environmental Consideration

Pursuant to 10 CFR 51.22(b), an evaluation of the 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 R. 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 Revision to the FSAR

After the approval of the LAR, in accordance with 10 CFR 50.71(e), the HNP FSAR will be revised. See Attachment S, Table S-3, Implementation Item IMP-16. The format and content will be consistent with NEI 04-02 FAQ 12-0062.

5.5 Transition Implementation Schedule

The following schedule for transitioning HNP to the new fire protection licensing basis requires NRC approval of the LAR in accordance with the following schedule:

- Implementation of new NFPA 805 fire protection program to include procedure changes, process updates, and training to affected plant personnel. This will occur 365 days after NRC approval. See Implementation Items in Table S-3 of Attachment S.
- Modifications will be completed by the startup of the second refueling outage (for each unit) after the issuance of the SE. See Table S-2 of Attachment S.

6.0 REFERENCES

The following references were used in the development of the Transition Report. Additional references are in the NEI 04-02 Tables in the various Attachments.

1. NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition
2. NEI 00-01, Guidance for Post Fire Safe Shutdown Circuit Analysis, Revision 2, May 2009
3. NEI 00-01, Guidance for Post Fire Safe Shutdown Circuit Analysis, Revision 3, October 2011
4. NEI 04-02, Guidance for Implementing A Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c), Revision 2, April 2008
5. Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, Revision 1, December 2009
6. NUREG/CR-6850, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, September 2005
7. Regulatory Issue Summary 2007-19, Process for Communicating Clarifications of Staff Positions Provided in Regulatory Guide 1.205 Concerning Issues Identified During the Pilot Application of National Fire Protection Associated Standard 805, August 20, 2007 (ML0611660105)
8. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition
9. NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management, December 1991
10. ASME/ANS RA-Sa-2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications
11. Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 2, March 2009
12. Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 2, May 2011
13. EPRI Technical Report 1006756, Fire Protection Equipment Surveillance Optimization and Maintenance Guide, July 2003
14. EPRI Technical Report 1010068, Aggregations of Quantitative Risk Assessment Results, December 2005
15. Letter, SNC to NRC, Letter of Intent to Adopt the 2001 Edition of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", October 4, 2013

16. Letter, NRC to SNC, Edwin I. Hatch Nuclear Plant, Units 1 and 2 – Request for Enforcement Discretion in Accordance with the Interim Enforcement Policy for Fire Protection Issues During Transition to National Fire Protection Standard NFPA 805, December 2, 2013
17. Updated Final Safety Analysis Report, Revision 36
18. Technical Specifications Unit Nos. 1 and 2
19. SENH-16-005, Nuclear Safety Capability Assessment Methodology Review (Table B-2), Version 1
20. SENH-16-006, Fire Area Review Report (NEI 04-02 Table B-3), Version 1
21. SMNH-16-086, NFPA 805 Chapter 3 Fundamental Fire Protection Program and Design Elements Review (Table B-1), Version 1
22. SMNH-16-089, Existing Engineering Equivalency Evaluation Review, Version 1
23. SMNH-16-090, Existing Licensing Action Review, Version 1
24. SMNH-16-091, NFPA 805 Radioactive Release Review, Version 1
25. SMNH-16-093, Fire Risk Evaluation (FRE) Report, Version 1
26. NMP-GM-002, Corrective Action Program, Version 14.4
27. NMP-AD-027, NRC Inspection Preparation and Response, Version 8.0
28. NMP-ES-002, System Monitoring And Health Reporting, Version 21.2
29. NMP-ES-009-002, Engineering Programs - Health Reports And Notebooks, Version 24.0
30. Information Notice 92-18, Potential for Loss of Remote Shutdown Capability During a Control Room Fire, February 28, 1992
31. 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)
32. NRC Generic Letter 86-10, Supplement 1, Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area, March 25, 1994
33. 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
34. Letter, NRC to SNC, SER, October 4, 1978
35. Letter, NRC to SNC, SER, November 16, 1981
36. Letter, NRC to SNC, SER, February 11, 1983
37. Letter, NRC to SNC, SER, April 18, 1984
38. Letter, NRC to SNC, SER, January 16, 1985

39. Letter, NRC to SNC, SER, May 14, 1985

40. Letter, NRC to SNC, SER, January 2, 1987

41. Letter, NRC to SNC, SER, March 24, 1987

42. Letter, SNC to NRC, Request for Extension of Enforcement Discretion and Revised Submittal Date for 10 CFR 50.48(c) License Amendment Request, July 6, 2016

43. Letter, NRC to SNC, Confirmatory Order, October 3, 2016

44. Quality Assurance Topical Report, Version 18

45. NEI 05-04, Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard, Rev. 2, November 2008

46. NEI 07-12, Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines, Rev. 1, June 2010

47. NEI 12-13, External Hazards PRA Peer Review Process Guidelines, Rev. 0, August 2012

ATTACHMENTS

A. NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements

92 Pages Attached

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.1 General.	This chapter contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. These fire protection program elements and minimum design requirements shall not be subject to the performance-based methods permitted elsewhere in this standard. Previously approved alternatives from the fundamental protection program attributes of this chapter by the AHJ take precedence over the requirements contained herein.	N/A	N/A - Section title, no technical requirements.	N/A
3.2 Fire Protection Plan.	N/A	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.2.1 Intent.	A site-wide fire protection plan shall be established. This plan shall document management policy and program direction and shall define the responsibilities of those individuals responsible for the plan's implementation. This section establishes the criteria for an integrated combination of components, procedures, and personnel to implement all fire protection program activities.	Complies	No Additional Clarification	Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / All
3.2.2 Management Policy Direction and Responsibility.	A policy document shall be prepared that defines management authority and responsibilities and establishes the general policy for the site fire protection program.	Complies	No Additional Clarification	Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / All
3.2.2.1	The policy document shall designate the senior management position with immediate authority and responsibility for the fire protection program.	Complies	No Additional Clarification	Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / Section 3.0

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.2.2.2	The policy document shall designate a position responsible for the daily administration and coordination of the fire protection program and its implementation.	Complies	No Additional Clarification	Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / Section 3.0
3.2.2.3	The policy document shall define the fire protection interfaces with other organizations and assign responsibilities for the coordination of activities. In addition, this policy document shall identify the various plant positions having the authority for implementing the various areas of the fire protection program.	Complies	No Additional Clarification	Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / Section 3.0
3.2.2.4	The policy document shall identify the appropriate AHJ for the various areas of the fire protection program.	Complies	No Additional Clarification	Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / Section 2.1

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.2.3 Procedures.	Procedures shall be established for implementation of the fire protection program. In addition to procedures that could be required by other sections of the standard, the procedures to accomplish the following shall be established:	Complies	No Additional Clarification	Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / All Procedure NMP-ES-035-001, Fire Protection Program Implementation, Ver. 13.1 / All Procedure NMP-ES-035-GL01, Fire Protection Program Guideline, Ver. 3.0 / All
3.2.3(1)	Inspection, testing, and maintenance for fire protection systems and features credited by the fire protection program	Complies	Except as identified below, HNP complies with no additional clarification.	Fire Hazards Analysis, Rev. 36 / Appendix B Procedure NMP-ES-035-015, Performance Based Evaluations for Fire Protection Surveillances, Ver. 1.0 / All Procedure NMP-GM-006-002, Surveillance/PM Program, Ver. 1.3 / All
		Complies, with Required Action	Implementation items are identified below.	None

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

IMP-1 Fire protection program document(s) will be updated in order to establish the controls on the use of Electric Power Research Institute (EPRI) Technical Report TR-1006756, "Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features" as noted in Attachment L, Approval Request 1.

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
		Submit for NRC Approval	NRC approval of the use of performance-based methods to establish the appropriate inspection, testing, and maintenance frequencies for required fire protection systems and features is being requested in Attachment L, Approval Request 1.	None
3.2.3(2)	Compensatory actions implemented when fire protection systems and other systems credited by the fire protection program and this standard cannot perform their intended function and limits on impairment duration	Complies	No Additional Clarification	Fire Hazards Analysis, Rev. 36 / Appendix B Procedure NMP-ES-035-005, Fire Protection Alternative Compensatory Measures, Ver. 6.0 / All
3.2.3(3)	Reviews of fire protection program — related performance and trends	Complies	No Additional Clarification	Procedure NMP-ES-002, System Monitoring and Health Reporting, Ver. 21.2 / All Procedure NMP-ES-009-002, Engineering Programs - Health Reports and Notebooks, Ver. 24.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.2.3(4)	Reviews of physical plant modifications and procedure changes for impact on the fire protection program	Complies	No Additional Clarification	Procedure NMP-AD-008, Applicability Determinations, Ver. 20.0 / Attachment 4 Procedure NMP-ES-035-006, Fire Protection Program Impact Screen and Detailed Reviews, Ver. 9.0 / All
3.2.3(5)	Long-term maintenance and configuration of the fire protection program	Complies	No Additional Clarification	Procedure NMP-ES-009-002, Engineering Programs - Health Reports and Notebooks, Ver. 24.0 / All Procedure NMP-ES-035-002, Fire Protection Program Notebooks, Ver. 5.0 / All
3.2.3(6)	Emergency response procedures for the plant industrial fire brigade	Complies	No Additional Clarification	Drawing Series A-43965, Pre-Fire Plan for Powerblock Areas / All Drawing Series A-43966, Pre-Fire Plan for Non-Powerblock Areas / All Procedure NMP-ES-035-010, Fire Brigade, Ver. 5.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3 Prevention.	A fire prevention program with the goal of preventing a fire from starting shall be established, documented, and implemented as part of the fire protection program. The two basic components of the fire prevention program shall consist of both of the following:	Complies	No Additional Clarification	Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / All
3.3(1)	Prevention of fires and fire spread by controls on operational activities	Complies	No Additional Clarification	Procedure 10AC-MGR-022-0, Plant Housekeeping and Material Condition, Ver. 7.2 / All Procedure NMP-ES-035-003, Fleet Hot Work Instruction, Ver. 7.0 / All Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / All Procedure NMP-FLS-017, Compressed Gas Safety, Ver. 2.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3(2)	Design controls that restrict the use of combustible materials. The design control requirements listed in the remainder of this section shall be provided as described.	Complies	No Additional Clarification	Procedure NMP-ES-084-001, Plant Modification and Configuration Change Processes, Ver. 6.0 / All Procedure NMP-ES-084-001, Plant Modification and Configuration Change Processes, Ver. 7.0 / All
3.3.1 Fire Prevention for Operational Activities.	The fire prevention program activities shall consist of the necessary elements to address the control of ignition sources and the use of transient combustible materials during all aspects of plant operations. The fire prevention program shall focus on the human and programmatic elements necessary to prevent fires from starting or, should a fire start, to keep the fire as small as possible.	Complies	No Additional Clarification	Procedure 10AC-MGR-022-0, Plant Housekeeping and Material Condition, Ver. 7.2 / All Procedure NMP-ES-035-003, Fleet Hot Work Procedure, Ver. 7.0 / All Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / All Procedure NMP-FLS-017, Compressed Gas Safety, Ver. 2.0 / All
3.3.1.1 General Fire Prevention Activities.	The fire prevention activities shall include but not be limited to the following program elements:	Complies	No Additional Clarification	None

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.1.1(1)	Training on fire safety information for all employees and contractors including, as a minimum, familiarization with plant fire prevention procedures, fire reporting, and plant emergency alarms	Complies	No Additional Clarification	Procedure NMP-TR-401, SNC General Employee Training, Ver. 10.1 / Section 4.3.1.6
3.3.1.1(2)	Documented plant inspections including provisions for corrective actions for conditions where unanalyzed fire hazards are identified	Complies	No Additional Clarification	Procedure NMP-ES-035-009, Quarterly Fire Safety Inspection, Ver. 4.0 / All
3.3.1.1(3)	Administrative controls addressing the review of plant modifications and maintenance to ensure that both fire hazards and the impact on plant fire protection systems and features are minimized	Complies	No Additional Clarification	Procedure NMP-ES-035-006, Fire Protection Program Impact Screen and Detailed Reviews, Ver. 9.0 / All Procedure NMP-MA-050, Work Package Preparation, Ver. 5.0 / All Procedure NMP-MA-050-F07, Fire Protection Screening, Ver. 1.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.1.2 Control of Combustible Materials.	Procedures for the control of general housekeeping practices and the control of transient combustibles shall be developed and implemented. These procedures shall include but not be limited to the following program elements:	Complies	No Additional Clarification	Procedure 10AC-MGR-022-0, Plant Housekeeping and Material Condition, Ver. 7.2 / All Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / All
3.3.1.2(1)	Wood used within the power block shall be listed pressure-impregnated or coated with a listed fire-retardant application. Exception: Cribbing timbers 6 in. by 6 in. (15.2 cm by 15.2 cm) or larger shall not be required to be fire-retardant treated.	Complies	No Additional Clarification	NRC Memorandum from Klein to AFPB File, "Close-Out of National Fire Protection Association 805 FAQ 12-0070 Regarding Use of Non-Fire Treated Wood" dated / All FAQ 14-0070, Use of Non-Fire Treated Wood, Rev. 0g / All Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / Section 4.1.10
3.3.1.2(2)	Plastic sheeting materials used in the power block shall be fire-retardant types that have passed NFPA 701, Standard Methods of Fire Tests for Flame Propagation of Textiles and Films, large-scale tests, or equivalent.	Complies	No Additional Clarification	Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / Section 4.1.13

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.1.2(3)	Waste, debris, scrap, packing materials, or other combustibles shall be removed from an area immediately following the completion of work or at the end of the shift, whichever comes first.	Complies	No Additional Clarification	Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / Section 4.1.3
3.3.1.2(4)	Combustible storage or staging areas shall be designated, and limits shall be established on the types and quantities of stored materials.	Complies	No Additional Clarification	Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / All
3.3.1.2(5)	Controls on use and storage of flammable and combustible liquids shall be in accordance with NFPA 30, Flammable and Combustible Liquids Code, or other applicable NFPA standards.	Complies with Use of EEEE's	Controls on the use and storage of flammable and combustible liquids are in accordance with NFPA 30, Flammable and Combustible Liquids Code, as identified in Calculation SMNH-16-032, NFPA 30 Code Compliance Review. With the exception of NFPA 30, no NFPA standards are applicable to controls on use and storage of flammable and combustible liquids at HNP, based on the guidance in Section K.1 of NEI 04-02.	Calculation SMNH-16-032, NFPA 30 Code Compliance Review, Ver. 1 / All NFPA 30, Flammable and Combustible Liquids Code, 2008 Edition / All NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48, Rev. 2 / Section K.1

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.1.2(6)	Controls on use and storage of flammable gases shall be in accordance with applicable NFPA standards.	Complies with Use of EEEE's	Controls on use and storage of bulk hydrogen are in accordance with NFPA 50A, Standard for Gaseous Hydrogen Systems at Consumer Sites, as identified in Calculation SMNH-16-034, NFPA 50A Code Compliance Review. With the exception of NFPA 50A, no NFPA standards are applicable to controls on use and storage of flammable gases at HNP, based on the guidance in Section K.1 of NEI 04-02.	Calculation SMNH-16-034, NFPA 50A Code Compliance Review, Ver. 1 / All NFPA 50A, Standard for Gaseous Hydrogen Systems at Consumer Sites, 1973 Edition / All NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48, Rev. 2 / Section K.1 Procedure NMP-FLS-017, Compressed Gas Safety, Ver. 2.0 / All
3.3.1.3 Control of Ignition Sources.	N/A	N/A	N/A - Section title, no technical requirements. See subsections for specific compliance statements and references.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.1.3.1	A hot work safety procedure shall be developed, implemented, and periodically updated as necessary in accordance with NFPA 51B, Standard for Fire Prevention During Welding, Cutting, and Other Hot Work, and NFPA 241, Standard for Safeguarding Construction, Alteration, and Demolition Operations.	Complies with Use of EEEE's	The hot work safety procedures comply with NFPA 51B and NFPA 241, as identified in Calculation SMNH-16-035, NFPA 51B Code Compliance Review, and Calculation SMNH-16-040, NFPA 241 Code Compliance Review.	Calculation SMNH-16-035, NFPA 51B Code Compliance Review, Ver. 1 / All Calculation SMNH-16-040, NFPA 241 Code Compliance Review, Ver. 1 / All NFPA 241, Standard for Safeguarding Construction, Alteration, and Demolition Operations, 2000 Edition / All NFPA 51B, Standard for Fire Prevention during Welding, Cutting, and other Hot Work, 2003 Edition / All Procedure NMP-ES-035-003, Fleet Hot Work Procedure, Ver. 7.0 / All
3.3.1.3.2	Smoking and other possible sources of ignition shall be restricted to properly designated and supervised safe areas of the plant.	Complies	No Additional Clarification	Procedure 10AC-MGR-022-0, Plant Housekeeping and Material Condition, Ver. 7.2 / Section 4.4 Procedure NMP-ES-035-003, Fleet Hot Work Procedure, Ver. 7.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.1.3.3	Open flames or combustion-generated smoke shall not be permitted for leak or air flow testing.	Complies	No Additional Clarification	Procedure NMP-ES-035-003, Fleet Hot Work Procedure, Ver. 7.0 / Section 4.1
3.3.1.3.4	Plant administrative procedure shall control the use of portable electrical heaters in the plant. Portable fuel-fired heaters shall not be permitted in plant areas containing equipment important to nuclear safety or where there is a potential for radiological releases resulting from a fire.	Complies	No Additional Clarification	Procedure NMP-ES-035-003, Fleet Hot Work Procedure, Ver. 7.0 / Section 4.1.1.d and 4.1.2
3.3.2 Structural.	Walls, floors, and components required to maintain structural integrity shall be of noncombustible construction, as defined in NFPA 220, Standard on Types of Building Construction.	Complies	No Additional Clarification	NFPA 220, Standard on Types of Building Construction, 1999 Edition / All Edwin I. Hatch Nuclear Plant Final Safety Analysis Reports Update, Rev. 36 / HNP-1-FSAR-12.2 and HNP-2-FSAR-3

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.3 Interior Finishes.	Interior wall or ceiling finish classification shall be in accordance with NFPA 101®, Life Safety Code®, requirements for Class A materials. Interior floor finishes shall be in accordance with NFPA 101 requirements for Class I interior floor finishes.	Complies	Except as identified below, HNP complies with no additional clarification.	Nuclear Electric Insurance Limited Loss Control Manual, January 2017 Edition / Section 3.2.10 Procedure NMP-ES-084-003, Site Facility Change Process, Ver. 1.2 / All Procedure NMP-GM-011, Procurement, Receipt, and Control of Materials and Services, Ver. 26.1 / Section 3.3 Procedure NMP-MA-011, Nuclear Coatings Program, Ver. 5.1 / All Procedure NMP-MA-011-GL01, Guideline for Nuclear Coatings Program, Ver. 2.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
		Complies by Previous NRC Approval	In the response to Appendix A to Branch Technical Position APCSB 9.5-1 for Units No. 1 and 2, item IV.B.1.d Georgia Power stated: "Paint coating systems used at HNP have all had flame spread test performed in accordance with Section 2 of ANSI N101.2-1972, with the exception of some used in Unit #1 portion of control building. All tested systems had flame spread rates of forty five or less." The NRC Safety Evaluation Report dated October 4, 1978 stated: "We find that the Fire Protection Program for Edwin I. Hatch Nuclear Plant with the improvements already made by the licensee, is adequate for the present and, with the scheduled modifications, will meet the guidelines contained in Appendix A to Branch Technical Position 9.5-1 with a single acceptable alternative and meets the General Design Criterion 3 and is, therefore, acceptable." The "scheduled modifications" were reviewed and are not applicable to this requirement. The basis for approval has been reviewed. There have been no plant modifications or other changes that would invalidate the basis for approval.	Letter from Ippolito (NRC) to Whitmer (GPC) dated October 4, 1978 / Enclosure 2, Section VIII HNP Response to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" dated October 27, 1976 / Section IV.B.1.d
		Complies, with Required Action	Implementation items are identified below.	None

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

IMP-2 Plant procedures will be revised to ensure that all future interior finish installations will be in accordance with NFPA 101 requirements as required by NFPA 805 Section 3.3.3.

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.4 Insulation Materials.	Thermal insulation materials, radiation shielding materials, ventilation duct materials, and soundproofing materials shall be noncombustible or limited combustible.	Complies with Clarification	NMP-ES-035-014 states, "Temporary thermal insulation material, radiation shielding materials, ventilation duct material, and sound proofing material shall be noncombustible or limited combustible." In cases where thermal insulation is located near a fire hazard, SS6902189 requires that it shall be covered with an aluminum sheet extending at least 24 inches beyond the source of the hazard. Per SS6915002, all ductwork, except in the drywell, shall be constructed of galvanized copper bearing sheet steel. Ductwork in the drywell shall be fabricated of 18-8 AISI, type 302, stainless steel. SS6915002 states that all duct insulation shall be fiber glass or approved equal and shall have a factory applied vapor barrier facing embossed aluminum foil. Per SS2115002, all ductwork, except in the drywell, shall be constructed of galvanized copper bearing sheet steel. Ductwork in the drywell shall be ASTM A 167, stainless steel sheet.	Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / Section 4.1.19 Specification S-52300, Insulation Instruction Manual, Rev. 0 / All Specification SS2102176, Thermal Insulation Outside Primary Containment, Rev. 6 / All Specification SS2115002, Heating, Ventilating and Air Conditioning of Edwin I. Hatch Nuclear Plant Unit 2, Rev. 0 / Appendix F Specification SS6902189, Thermal Insulation - Piping and Equipment Outside Primary Containment, Rev. 0 / All Specification SS6915002, Heating, Ventilating and Air Conditioning of Edwin I. Hatch Nuclear Plant Unit 1, Rev. 1 / Section 9.0 Specification SX-28056, Insulation Instruction Manual, Rev. 1 / All
		Submit for NRC Approval	NRC approval of existing thermal insulation present in the plant is being requested in Attachment L, Approval Request 2.	None

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.5 Electrical.	N/A	N/A	N/A - Section title, no technical requirements. See subsections for specific compliance statements and references.	N/A
3.3.5.1	Wiring above suspended ceiling shall be kept to a minimum. Where installed, electrical wiring shall be listed for plenum use, routed in armored cable, routed in metallic conduit, or routed in cable trays with solid metal top and bottom covers.	Complies, with Required Action	Implementation items are identified below.	None
	IMPLEMENTATION ITEMS (See Attachment S, Table S-3):			
	IMP-3 Plant documentation will be revised to incorporate the requirements for electrical wiring above suspended ceilings for all future installations.			
		Submit for NRC Approval	NRC approval for the current configuration of wiring above suspended ceilings is being requested in Attachment L, Approval Request 3.	None

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.5.2	Only metal tray and metal conduits shall be used for electrical raceways. Thin wall metallic tubing shall not be used for power, instrumentation, or control cables. Flexible metallic conduits shall only be used in short lengths to connect components.	Complies	Except as identified below, HNP complies with no additional clarification.	Drawing A29500, Conduit and Conduit Support, Ver. 3.0 / Section 3.1 Drawing A29501, General Design Document and Details for the Installation of Nonsafety-Related Electrical Work, Ver. 2.0 / Section 6.0 Drawing B13000, Conduit & Grounding Installation Notes, Ver. 5.0 E-1-03, SNC Raceway Design Standard, Rev. 6 Specification SS-2123-009, Technical Specification for Cable Trays and Cable Tray Accessories for the Edwin I. Hatch Nuclear Plant - Unit 2, Rev. A / All
		Submit for NRC Approval	FAQ 06-0021 defines "short lengths" as approximately three feet of flexible metallic conduit. NRC approval of the use of PVC coated flexible conduit in lengths up to 6 feet and embedded non-metallic conduit is being requested in Attachment L, Approval Request 4.	FAQ 06-0021, Cable Air Drops, Rev. 0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.5.3	Electric cable construction shall comply with a flame propagation test as acceptable to the AHJ.	Complies	Except as identified below, HNP complies with no additional clarification.	Specification GPEE-8001, Power and Control Cables, Special Instrumentation and Control Cables, and Communication Cables, Ver. 7 / Section 3.1

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
		Complies by Previous NRC Approval	In the response to Appendix A to Branch Technical Position APCSB 9.5-1 for Units No. 1 and 2, item D.3.f Georgia Power stated: "All manufacturers supplying cables for Unit 2 have supplied flame test data which meets IEEE 383-1974 or IPCEA vertical tray flame test S-19-81. IEEE-383 was not in existence at the time cable was purchased for HNP-1. The cable purchased met the state-of-the-art, as it existed at that time, with respect to flame retardance. Further discretion, based upon past experience with the Southern Company's fossil fuel plants, was used in evaluating cable with respect to flame retardance. Since the issuance of IEEE-383, documentation has been received which will qualify 70% of the cable types purchased for Unit 1. These types include all of the power and general purpose control cables. The only types not meeting IEEE-383-1974 are some of the instrumentation and communication cables." The NRC Safety Evaluation Report dated October 4, 1978 stated: "We find that the Fire Protection Program for Edwin I. Hatch Nuclear Plant with the improvements already made by the licensee, is adequate for the present and, with the scheduled modifications, will meet the guidelines contained in Appendix A to Branch Technical Position 9.5-1 with a single acceptable alternative and meets the General Design Criterion 3 and is, therefore, acceptable." The basis for approval has been reviewed. There have been no plant modifications or other changes that would invalidate the basis for approval.	Letter from Ippolito (NRC) to Whitmer (GPC) dated October 4, 1978 / Enclosure 2, Section VIII HNP Response to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" dated October 27, 1976 / Section IV.B.3.f

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.6 Roofs.	Metal roof deck construction shall be designed and installed so the roofing system will not sustain a self-propagating fire on the underside of the deck when the deck is heated by a fire inside the building. Roof coverings shall be Class A as determined by tests described in NFPA 256, Standard Methods of Fire Tests of Roof Coverings.	Complies	No Additional Clarification	Drawing H-12600, Fire Protection Pump House Floor Plan & South & West Elevations, Ver. 5.0 / All Drawing H-12620, Substation and Switch House Plans and Elevations, Ver. 1 / All Drawing H-12621, Cooling Tower Switch Houses No. 1A, No. 1B, & No. 1C, Ver. 6.0 / All Drawing H-22832, Unit 2 Chill Water Equipment Building Elevations & Sections, Rev. 0 / All Drawing H25966, Cooling Tower Switch Houses No. 4, No. 5, & No. 6, Ver. 2.0 / All Drawing H-43982, Chilled Water System Bldg. Archetectural Floor Plan & Details, Rev. 0 / All Drawing H-44596, Helper Cooling Towers MCC Houses Floor Plan, Elevations & Specifications, Rev. 0 / All Drawing H46471, Powerblock Area Re-roofing Plan, Ver. 2.0 / All Drawing H46472, Powerblock Area Re-roofing Sections & Detail, Ver. 2.0 / All Plant Modification Proposal, Re-roof the Plant Hatch Powerblock Buildings including Units 1 & 2 Reactor, Turbine, Control, Hot Machine Shop, Radwaste, and Radwaste Addition Buildings / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
				Specification HC-S-06-001, Specification for Adhered Single-Ply Roofing and Flashing for Edwin I. Hatch Nuclear Plant Units 1 and 2, Ver. 3.0 / All Specification HC-S-06-004, Specification for Installation of 30-Year Decothane Roofing Membrane and Flashing for Edwin I. Hatch Nuclear Plant, Ver. 1.0 / All Specification SCDS 04-01, Specification for Membrane Roofing and Flashing for Hatch Nuclear Plant Units 1 and 2, Ver. 3.0 / All
3.3.7 Bulk Flammable Gas Storage.	Bulk compressed or cryogenic flammable gas storage shall not be permitted inside structures housing systems, equipment, or components important to nuclear safety.	Complies	No Additional Clarification	Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / Section 4.3.13

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.7.1	Storage of flammable gas shall be located outdoors, or in separate detached buildings, so that a fire or explosion will not adversely impact systems, equipment, or components important to nuclear safety. NFPA 50A, Standard for Gaseous Hydrogen Systems at Consumer Sites, shall be followed for hydrogen storage.	Complies with Use of EEEE's	Per the Plant Hatch Fire Hazards Analysis, bulk storage of combustibles is not permitted inside or adjacent to safety-related systems or structures. Hydrogen storage is in accordance with NFPA 50A as identified in Calculation SMNH-16-034, NFPA 50A Code Compliance Review.	Calculation SMNH-16-034, NFPA 50A Code Compliance Review, Ver. 1 / All Fire Hazards Analysis, Rev. 36 / Appendix A Procedure NMP-ES-084-003, Site Facility Change Process, Ver. 1.2 / All
3.3.7.2	Outdoor high-pressure flammable gas storage containers shall be located so that the long axis is not pointed at buildings.	Complies with Clarification	Bulk hydrogen storage tanks are located outdoors with their long axes parallel to power block buildings. The hydrogen storage tanks are pointed at the Williams Buildings, approximately 150 ft away, which is a non-power block building, a loss of the Williams Buildings would not affect nuclear safety performance criteria.	Drawing H11802, Fire Hazards Analysis General Building Site Plan, Ver. 14.0 / All
3.3.7.3	Flammable gas storage cylinders not required for normal operation shall be isolated from the system.	Complies	No Additional Clarification	Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / Section 4.3.5

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.8 Bulk Storage of Flammable and Combustible Liquids.	Bulk storage of flammable and combustible liquids shall not be permitted inside structures containing systems, equipment, or components important to nuclear safety. As a minimum, storage and use shall comply with NFPA 30, Flammable and Combustible Liquids Code.	Complies by Previous NRC Approval	In the response to Appendix A to Branch Technical Position APCSB 9.5-1 for Units No. 1 and 2, item III.C.1.5(a), Georgia Power stated: "The oil storage rooms are located in the northwest and southwest corners of the control building at elevation 112'. Turbine lube oil is stored in tanks located in these rooms...Fire detection is provided by an automatic dry pilot system which gives an audible alarm locally and in the main control room. Due to the high fire load in these rooms a dry pilot fixed deluge system was designed and installed as the primary fire protection system. Manual hose stations and portable CO2 fire extinguishers are provided as backup. Normally covered floor drains are provided. Doors, floors, ceilings and walls are of three hour fire rated construction. The rooms are equipped with a double three hour automatic fire door. This doorway provides access for fire fighting. A dike high enough to contain the oil is provided should the tank rupture. The ventilation ducts are designed with fire dampers located in runs that pass through fire barriers and which automatically close in the event of a fire in one of these rooms. A fire in one of these areas will not prevent the safe shutdown of the plant. The NRC Safety Evaluation Report dated October 4, 1978, which was issued in response to the GPC letter dated October 27, 1976, stated: "The licensee's Fire Hazards Analysis addresses other plant areas not specifically discussed in this report. The licensee has committed to install additional detectors, portable extinguishers, hose stations, and some additional emergency lighting as identified in the licensee's installation schedule. With the commitment made by the licensee, we find these areas to be in accordance with the guidelines of Appendix A of BTP 9.5-1, and the applicable sections of the National Fire Protection Association Code and are therefore acceptable." The scheduled modifications referenced above have	Letter from Ippolito (NRC) to Whitmer (GPC) dated October 4, 1978 / Enclosure 2, Section IV.E HNP Response to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" dated October 27, 1976 / Section III.C.1.5(a)

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
			been reviewed and are not applicable to this requirement. The basis for approval has been reviewed and there have been no plant modifications or other changes that would invalidate the basis for this approval.	
		Complies with Use of EEEE's	Storage and use of flammable and combustible liquids are in accordance with NFPA 30, Flammable and Combustible Liquids Code, as identified in Calculation SMNH-16-032, NFPA 30 Code Compliance Review.	Calculation SMNH-16-032, NFPA 30 Code Compliance Review, Ver. 1 / All NFPA 30, Flammable and Combustible Liquids Code, 1973 Edition / All NFPA 30, Flammable and Combustible Liquids Code, 2008 Edition / All
3.3.9 Transformers.	Where provided, transformer oil collection basins and drain paths shall be periodically inspected to ensure that they are free of debris and capable of performing their design function.	Complies	No Additional Clarification	PM N1Y281, Engineer to Visually Inspect Storm Drains for Breaks, Debris Buildip and General Condition / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.10 Hot Pipes and Surfaces.	Combustible liquids, including high flashpoint lubricating oils, shall be kept from coming in contact with hot pipes and surfaces, including insulated pipes and surfaces. Administrative controls shall require the prompt cleanup of oil on insulation.	Complies	No Additional Clarification	Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / Section 4.2.3.6
3.3.11 Electrical Equipment.	Adequate clearance, free of combustible material, shall be maintained around energized electrical equipment.	Complies	No Additional Clarification	NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48, Rev. 2 / Section K.5 Procedure NMP-ES-035-014, Fleet Transient Combustible Controls, Ver. 2.0 / Section 4.1.11
3.3.12 Reactor Coolant Pumps.	For facilities with non-inerted containments, reactor coolant pumps with an external lubrication system shall be provided with an oil collection system. The oil collection system shall be designed and installed such that leakage from the oil system is safely contained for off normal conditions such as accident conditions or earthquakes. All of the following shall apply.	N/A	Not Applicable - HNP has an inerted containment.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.12(1)	The oil collection system for each reactor coolant pump shall be capable of collecting lubricating oil from all potential pressurized and nonpressurized leakage sites in each reactor coolant pump oil system.	N/A	Not Applicable - HNP has an inerted containment.	N/A
3.3.12(2)	Leakage shall be collected and drained to a vented closed container that can hold the inventory of the reactor coolant pump lubricating oil system.	N/A	Not Applicable - HNP has an inerted containment.	N/A
3.3.12(3)	A flame arrestor is required in the vent if the flash point characteristics of the oil present the hazard of a fire flashback.	N/A	Not Applicable - HNP has an inerted containment.	N/A
3.3.12(4)	Leakage points on a reactor coolant pump motor to be protected shall include but not be limited to the lift pump and piping, overflow lines, oil cooler, oil fill and drain lines and plugs, flanged connections on oil lines, and the oil reservoirs, where such features exist on the reactor coolant pumps.	N/A	Not Applicable - HNP has an inerted containment.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.3.12(5)	The collection basin drain line to the collection tank shall be large enough to accommodate the largest potential oil leak such that oil leakage does not overflow the basin.	N/A	Not Applicable - HNP has an inerted containment.	N/A
3.4 Industrial Fire Brigade.	N/A	N/A	N/A - Section title, no technical requirements. See subsections for specific compliance statements and references.	N/A
3.4.1 On-Site Fire-Fighting Capability.	All of the following requirements shall apply.	N/A	N/A - Section title, no technical requirements. See subsections for specific compliance statements and references.	N/A
3.4.1(a) On-Site Fire-Fighting Capability.	A fully staffed, trained, and equipped fire-fighting force shall be available at all times to control and extinguish all fires on site. This force shall have a minimum complement of five persons on duty and shall conform with the following NFPA standards as applicable:	Complies	No Additional Clarification	Procedure NMP-ES-035-010, Fire Brigade, Ver. 5.0 / Section 4.0

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.1(a)(1)	NFPA 600, Standard on Industrial Fire Brigades (interior structural fire fighting)	Complies with Use of EEEE's	The site fire brigade complies with NFPA 600, as identified in Calculation SMNH-16-041, NFPA 600 Code Compliance Review.	Calculation SMNH-16-041, NFPA 600 Code Compliance Review, Ver. 1 / All NFPA 600, Standard on Industrial Fire Brigades, 2005 Edition / All
3.4.1(a)(2)	NFPA 1500, Standard on Fire Department Occupational Safety and Health Program	N/A	NFPA 1500, Section 1.3.2 states, "This standard shall not apply to industrial fire brigades that might also be known as emergency brigades, emergency response teams, fire teams. Plant emergency organizations, or mine emergency response teams." Therefore, NFPA 1500 does not apply to industrial fire brigades.	NFPA 1500, Standard on Fire Department Occupational Health and Safety Program, 2007 Edition / Section 1.3.2 NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48, Rev. 2 / Section K.6
3.4.1(a)(3)	NFPA 1582, Standard on Medical Requirements for Fire Fighters and Information for Fire Department Physicians	N/A	NFPA 1582, Section 1.1.4 states, "This standard shall not apply to industrial fire brigades that also can be known as emergency brigades, emergency response teams, fire teams, plant emergency organizations, or mine emergency response teams." Therefore, NFPA 1582 does not apply to industrial fire brigades.	NFPA 1582, Standard on Comprehensive Occupational Medical Program for Fire Departments, 2000 Edition / Section 1.1.4 NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48, Rev. 2 / Section K.6

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.1(b)	Industrial fire brigade members shall have no other assigned normal plant duties that would prevent immediate response to a fire or other emergency as required.	Complies	No Additional Clarification	Procedure NMP-ES-035-010, Fire Brigade, Ver. 5.0 / Section 3.0 Procedure NMP-OS-007-009, Site Specific Hatch Minimum Shift Crew Composition, Ver. 1.4 / Section 4.6
3.4.1(c)	During every shift, the brigade leader and at least two brigade members shall have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance criteria. Exception: Sufficient training and knowledge shall be permitted to be provided by an operations advisor dedicated to industrial fire brigade support.	Complies	No Additional Clarification	FAQ 13-0069, Fire Brigade Member Qualifications, Rev. 3 / All Procedure NMP-ES-035-010, Fire Brigade, Ver. 5.0 / Section 4.0 Procedure NMP-OS-007, Conduct of Operations, Ver. 13.1 / Sections 3.11, 3.12, 3.13, and 3.19 Procedure NMP-TR-415, Systems Operator Initial and Continuing Training Program, Ver. 5.0 / All
3.4.1(d)	The industrial fire brigade shall be notified immediately upon verification of a fire.	Complies	No Additional Clarification	Procedure 34AB-X43-001-1, Fire Procedure, Ver. 15.1 / Section 4.0 Procedure 34AB-X43-001-2, Fire Procedure, Ver. 18.1 / Section 4.0

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.1(e)	Each industrial fire brigade member shall pass an annual physical examination to determine that he or she can perform the strenuous activity required during manual firefighting operations. The physical examination shall determine the ability of each member to use respiratory protection equipment.	Complies	No Additional Clarification	Procedure MS-MED-009, Fire Brigade Physical Examination Procedure, Ver. 1.1 / All Procedure NMP-ES-035-010, Fire Brigade, Ver. 5.0 / Section 3.0 Procedure NMP-TR-426, Fire Training Program, Ver. 5.1 / Sections 4.2.1.k and 4.3
3.4.2 Pre-Fire Plans.	Current and detailed pre-fire plans shall be available to the industrial fire brigade for all areas in which a fire could jeopardize the ability to meet the performance criteria described in Section 1.5.	Complies, with Required Action	Current and detailed pre-fire plans for fighting fires in all areas have been instituted at HNP and are readily available. The pre-fire plans describe actions to be taken by firefighting personnel during the fire, including instructions on use of firefighting equipment. Implementation items are identified below.	Drawing Series A-43965, Pre-Fire Plan for Powerblock Areas Drawing Series A-43966, Pre-Fire Plan for Non-Powerblock Areas

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

IMP-4 Fire Brigade training materials and pre-fire plans will be revised to address the radioactive release requirements of NFPA 805 and to ensure that details regarding nuclear safety components that are present in the area are provided.

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.2.1	The plans shall detail the fire area configuration and fire hazards to be encountered in the fire area, along with any nuclear safety components and fire protection systems and features that are present.	Complies, with Required Action	The pre-fire plans at Plant Hatch detail the fire area configuration and fire hazards to be encountered for the fire area. The pre-fire plans also list any fire protection systems and features that are present in the fire area. Implementation items are identified below.	Drawing Series A-43965, Pre-Fire Plan for Powerblock Areas / All Drawing Series A-43966, Pre-Fire Plan for Non-Powerblock Areas / All Procedure NMP-ES-035-GL01, Fire Protection Program Guideline, Ver. 3.0 / Section 3.2.4.f
	IMPLEMENTATION ITEMS (See Attachment S, Table S-3): IMP-4 Fire Brigade training materials and pre-fire plans will be revised to address the radioactive release requirements of NFPA 805 and to ensure that details regarding nuclear safety components that are present in the area are provided.			
3.4.2.2	Pre-fire plans shall be reviewed and updated as necessary.	Complies	No Additional Clarification	PM NCFIREPLAN1, Review Pre-Fire Plans / All PM NCFIREPLAN2, Review Pre-Fire Plans / All Procedure NMP-ES-035-GL01, Fire Protection Program Guideline, Ver. 3.0 / Sections 3.2.4.e(1) and 3.2.4.e(3) Procedure NMP-ES-084-001, Plant Modification and Configuration Change Processes, Ver. 7.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.2.3	Pre-fire plans shall be available in the control room and made available to the plant industrial fire brigade.	Complies	No Additional Clarification	Procedure NMP-ES-084-009, Document Preparation, Update, and Transmittal Process, Ver. 1.0 / All
3.4.2.4	Pre-fire plans shall address coordination with other plant groups during fire emergencies.	Complies with Clarification	The coordination between other plant groups during fire emergencies is outlined in NMP-ES-035-010, 34AB-X43-001-1, and 34AB-X43-001-2. Pre-fire plans are not used to coordinate between plant groups during fire emergencies.	Southern Nuclear Operating Company Standard Emergency Plan, Ver. 2 / Section B.3 Procedure 34AB-X43-001-1, Fire Procedure, Ver. 15.1 / Section 4.0 Procedure 34AB-X43-001-2, Fire Procedure, Ver. 18.1 / Section 4.0 Procedure NMP-ES-035-010, Fire Brigade, Ver. 5.0 / Section 3.0
3.4.3 Training and Drills.	Industrial fire brigade members and other plant personnel who would respond to a fire in conjunction with the brigade shall be provided with training commensurate with their emergency responsibilities.	Complies	No Additional Clarification	Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / Sections 3.8, 3.9, and 3.10 Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / Section 3.0

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.3(a)(1) Plant Industrial Fire Brigade Training. All of the following requirements shall apply.	Plant industrial fire brigade members shall receive training consistent with the requirements contained in NFPA 600, Standard on Industrial Fire Brigades, or NFPA 1500, Standard on Fire Department Occupational Safety and Health Program, as appropriate.	Complies with Use of EEEE's	The fire brigade training is in accordance with NFPA 600, as identified in Calculation SMNH-16-041, NFPA 600 Code Compliance Review. Plant industrial fire brigade members receive training in accordance with Procedure NMP-TR-426, which is consistent with the requirements contained in NFPA 600. NFPA 1500 is not applicable as it is for fire departments.	Calculation SMNH-16-041, NFPA 600 Code Compliance Review, Ver. 1 / All NFPA 600, Standard on Industrial Fire Brigades, 2005 Edition / All NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c), 2 / Section K.6 Procedure NMP-TR-426, Fire Training Program, Ver. 5.1 / All
3.4.3(a)(2)	Industrial fire brigade members shall be given quarterly training and practice in fire fighting, including radioactivity and health physics considerations, to ensure that each member is thoroughly familiar with the steps to be taken in the event of a fire.	Complies	No Additional Clarification	Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / Section 4.0 Procedure NMP-TR-426, Fire Training Program, Ver. 5.1 / Section 4.5.2
3.4.3(a)(3)	A written program shall detail the industrial fire brigade training program.	Complies	No Additional Clarification	Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / All Procedure NMP-TR-426, Fire Training Program, Ver. 5.1 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.3(a)(4)	Written records that include but are not limited to initial industrial fire brigade classroom and hands-on training, refresher training, special training schools attended, drill attendance records, and leadership training for industrial fire brigades shall be maintained for each industrial fire brigade member.	Complies	No Additional Clarification	Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / Section 5.0 Procedure NMP-TR-426, Fire Training Program, Ver. 5.1 / Section 5.0
3.4.3(b) Training for Non-Industrial Fire Brigade Personnel.	Plant personnel who respond with the industrial fire brigade shall be trained as to their responsibilities, potential hazards to be encountered, and interfacing with the industrial fire brigade.	Complies	No Additional Clarification	Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / Sections 3.8, 3.9, and 3.10 Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / Sections 3.8, 3.9, 3.10, and 3.11
3.4.3(c)(1) Drills. All of the following requirements shall apply.	Drills shall be conducted quarterly for each shift to test the response capability of the industrial fire brigade.	Complies	No Additional Clarification	Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / Section 4.2.1

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.3(c)(2)	Industrial fire brigade drills shall be developed to test and challenge industrial fire brigade response, including brigade performance as a team, proper use of equipment, effective use of pre-fire plans, and coordination with other groups. These drills shall evaluate the industrial fire brigade's abilities to react, respond, and demonstrate proper fire-fighting techniques to control and extinguish the fire and smoke conditions being simulated by the drill scenario.	Complies	No Additional Clarification	Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / All
3.4.3(c)(3)	Industrial fire brigade drills shall be conducted in various plant areas, especially in those areas identified to be essential to plant operation and to contain significant fire hazards.	Complies	No Additional Clarification	Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / Sections 4.2.1 and 4.3.2
3.4.3(c)(4)	Drill records shall be maintained detailing the drill scenario, industrial fire brigade member response, and ability of the industrial fire brigade to perform as a team.	Complies	No Additional Clarification	Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / Section 5.0

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.3(c)(5)	A critique shall be held and documented after each drill.	Complies	No Additional Clarification	Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / Section 4.10
3.4.4 Fire-Fighting equipment.	Protective clothing, respiratory protective equipment, radiation monitoring equipment, personal dosimeters, and fire suppression equipment such as hoses, nozzles, fire extinguishers, and other needed equipment shall be provided for the industrial fire brigade. This equipment shall conform with the applicable NFPA standards.	Complies	No Additional Clarification	Form ENG-0246, Fire Fighting Equipment Inspection Inventory, Ver. 24.2 / All Procedure NMP-OS-007, Conduct of Operations, Ver. 13.1 / Section 3.22 Procedure NMP-TR-426, Fire Training Program, Ver. 5.1 / Section 4.15
3.4.5 Off-Site Fire Department Interface.	N/A	N/A	N/A - Section title, no technical requirements. See subsections for specific compliance statements and references.	N/A
3.4.5.1 Mutual Aid Agreement.	Off-site fire authorities shall be offered a plan for their interface during fires and related emergencies on site.	Complies	No Additional Clarification	Standard Emergency Plan Annex for Hatch Nuclear Plan Units 1 and 2, Ver. 2 / Section 2.3.1

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.5.2 Site-Specific Training.	Fire fighters from the off-site fire authorities who are expected to respond to a fire at the plant shall be offered site-specific training and shall be invited to participate in a drill at least annually.	Complies	No Additional Clarification	Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / Section 4.2.1 Procedure NMP-TR-426, Fire Training Program, Ver. 5.1 / Section 4.9
3.4.5.3 Security and Radiation Protection.	Plant security and radiation protection plans shall address off-site fire authority response.	Complies	No Additional Clarification	Procedure 62RP-RAD-048-0, Radiation Protection Response to a Radiological Fire, Ver. 3.1 / All Procedure 73EP-EIP-009-0, Nuclear Security Duties, Ver. 12.0 / Section 4.2 Procedure NMP-ES-035, Fire Protection Program, Ver. 6.0 / Section 3.9 and 3.10 Procedure NMP-TR-425, Fire Drill Program, Ver. 8.0 / Section 3.0

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.4.6 Communications.	An effective emergency communications capability shall be provided for the industrial fire brigade.	Complies	No Additional Clarification	Form ENG-0246, Fire Fighting Equipment Inspection Inventory, Ver. 24.2 / All Edwin I. Hatch Nuclear Plant Final Safety Analysis Reports Update, Rev. 36 / HNP-2-FSAR-9.5.2 Procedure DI-FPX-02-0693, Fire Fighting Equipment Inspection, Ver. 13.0 / All Procedure NMP-ES-035-010, Fire Brigade, Ver. 5.0 / Section 4.0
3.5 Water Supply.	N/A	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	N/A
3.5.1	A fire protection water supply of adequate reliability, quantity, and duration shall be provided by one of the two following methods.	N/A	N/A - General statement, no technical requirements.	N/A
3.5.1(a)	Provide a fire protection water supply of not less than two separate 300,000-gal (1,135,500-L) supplies.	N/A	N/A - Option (b) is used to demonstrate compliance with Section 3.5.1.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.1(b)	Calculate the fire flow rate for 2 hours. This fire flow rate shall be based on 500 gpm (1892.5 L/min) for manual hose streams plus the largest design demand of any sprinkler or fixed water spray system(s) in the power block as determined in accordance with NFPA 13, Standard for the Installation of Sprinkler Systems, or NFPA 15, Standard for Water Spray Fixed Systems for Fire Protection. The fire water supply shall be capable of delivering this design demand with the hydraulically least demanding portion of fire main loop out of service.	Complies	No Additional Clarification	Calculation SMNH-16-110, Fire Protection Water Main Loop Adequacy Determination, Ver. 1 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.2	The tanks shall be interconnected such that fire pumps can take suction from either or both. A failure in one tank or its piping shall not allow both tanks to drain. The tanks shall be designed in accordance with NFPA 22, Standard for Water Tanks for Private Fire Protection. Exception No. 1: Water storage tanks shall not be required when fire pumps are able to take suction from a large body of water (such as a lake), provided each fire pump has its own suction and both suctions and pumps are adequately separated. Exception No. 2: Cooling tower basins shall be an acceptable water source for fire pumps when the volume is sufficient for both purposes and water quality is consistent with the demands of the fire service.	Complies with Use of EEEE's	The fire water storage tanks are designed in accordance with NFPA 22 as identified in Calculation SMNH-16-030, NFPA 22 Code Compliance Review.	Calculation SMNH-16-030, NFPA 22 Code Compliance Review, Ver. 1 / All Drawing H-11033 Sheet 1, Fire Protection- P&ID Pumphouse Layout, Ver. 51.0 / All NFPA 22, Standard for Water Tanks for Private Fire Protection, 1976 Edition / All
		Submit for NRC Approval	NRC approval of the lack of check valves in the fire water system between the fire water tanks and the fire pumps is being requested in Attachment L, Approval Request 5.	Drawing H-11033 Sheet 1, Fire Protection- P&ID Pumphouse Layout, Ver. 51.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.3	Fire pumps, designed and installed in accordance with NFPA 20, Standard for the Installation of Stationary Pumps for Fire Protection, shall be provided to ensure that 100 percent of the required flow rate and pressure are available assuming failure of the largest pump or pump power source.	Complies with Use of EEEE's	With the exception of the electric fire pump controller, fire pump design and installation is in accordance with NFPA 20, as identified in Calculation SMNH-16-029, NFPA 20 Code Compliance Review.	Calculation SMNH-16-029, NFPA 20 Code Compliance Review, Ver. 1 / All Calculation SMNH-16-110, Fire Protection Water Main Loop Adequacy Determination, Ver. 1 / All NFPA 20, Standard for Installation of Centrifugal Fire Pumps, 1972 Edition / All
		Submit for NRC Approval	NRC approval of the current configuration of the electric fire pump controller is being requested in Attachment L, Approval Request 6.	None
3.5.4	At least one diesel engine-driven fire pump or two more seismic Category I Class 1E electric motor-driven fire pumps connected to redundant Class 1E emergency power buses capable of providing 100 percent of the required flow rate and pressure shall be provided.	Complies	No Additional Clarification	Calculation SMNH-16-110, Fire Protection Water Main Loop Adequacy Determination, Ver. 1

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.5	Each pump and its driver and controls shall be separated from the remaining fire pumps and from the rest of the plant by rated fire barriers.	Complies	Except as identified below, HNP complies with no additional clarification.	Drawing H11846, Edwin I. Hatch Nuclear Plant Unit No. 1 & 2 Fire Hazards Analysis Diesel Generator Bldg., Ver. 3.0 / All Drawing H11848, Edwin I. Hatch Nuclear Plant Unit No. 1 & 2 Fire Hazards Analysis Fire Protection Pump House, Rev. 0 / All
		Submit for NRC Approval	NRC approval of the lack of fire rated barriers between the fire pumps and the rest of the plant is being requested in Attachment L, Approval Request 7.	Drawing H11848, Edwin I. Hatch Nuclear Plant Unit No. 1 & 2 Fire Hazards Analysis Fire Protection Pump House, Rev. 0 / All
3.5.6	Fire pumps shall be provided with automatic start and manual stop only.	Complies	No Additional Clarification	Drawing H-13384, Single Line Diagram 4160V Bus 1E, R22-S005 & Bus 1F, R22-S006, Ver. 30.0 / All Fire Hazards Analysis, Rev. 36 / Section 4.7 A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All

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

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
		Submit for NRC Approval	NRC approval of the lack of check valves in the fire water system installed at each water supply source is being requested in Attachment L, Approval Request 5.	Drawing H-11033 Sheet 1, Fire Protection- P&ID Pumphouse Layout, Ver. 51.0 / All
3.5.11	Means shall be provided to isolate portions of the yard fire main loop for maintenance or repair without simultaneously shutting off the supply to both fixed fire suppression systems and fire hose stations provided for manual backup. Sprinkler systems and manual hose station standpipes shall be connected to the plant fire protection water main so that a single active failure or a crack to the water supply piping to these systems can be isolated so as not to impair both the primary and backup fire suppression systems.	Complies	No Additional Clarification	Drawing H-11033 Sheet 1, Fire Protection- P&ID Pumphouse Layout, Ver. 51.0 / All Drawing H-11033 Sheet 2, Fire Protection- P&ID Yard Layout MP - House and Yard Mains, Ver. 27.0 / All Drawing H-11033 Sheet 3, Fire Protection P&ID Yard Layout - Cooling Tower, Rev. 30 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.12	Threads compatible with those used by local fire departments shall be provided on all hydrants, hose couplings, and standpipe risers. Exception: Fire departments shall be permitted to be provided with adapters that allow interconnection between plant equipment and the fire department equipment if adequate training and procedures are provided.	Complies	No Additional Clarification	Fire Hazards Analysis, Rev. 36 / Section 4.7

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.13	Headers fed from each end shall be permitted inside buildings to supply both sprinkler and standpipe systems, provided steel piping and fittings meeting the requirements of ANSI B31.1, Code for Power Piping, are used for the headers (up to and including the first valve) supplying the sprinkler systems where such headers are part of the seismically analyzed hose standpipe system. Where provided, such headers shall be considered an extension of the yard main system. Each sprinkler and standpipe system shall be equipped with an outside screw and yoke (OS&Y) gate valve or other approved shutoff valve.	Complies by Previous NRC Approval	The requirements of this section related to steel piping and fittings comprising interior supply headers meeting the requirements of ANSI B31.1 and a seismically analyzed standpipe system design have been assessed at HNP. Position E.3.d in Appendix A to BTP APCSB 9.5-1 did not require provisions to supply water at least to standpipes in the event of a safe shutdown earthquake (SSE) for plants which were issued construction permits prior to July 1, 1976. In the response to Appendix A to Branch Technical Position APCSB 9.5-1 for Units No. 1 and 2, item IV.A.4 Georgia Power stated: "The fire protection systems for HNP were not designed as seismic Category I systems. The water systems are designed such that no active failure will prevent water from being provided to both hose stations and sprinkler or deluge systems that receive water from the fire protection water tanks." The NRC Safety Evaluation Report dated October 4, 1978, which was issued in response to the GPC letter dated October 27, 1976, stated: "We have reviewed the design criteria and bases for the water suppression systems and conclude that these systems meet the guidelines of Appendix A to Branch Technical Position 9.5.1 and are in accord with the applicable portions of the National Fire Protection Association (NFPA) Codes, and are, therefore, acceptable." The basis for approval has been reviewed. There have been no plant modifications or other changes that would invalidate the basis for approval.	Letter from Ippolito (NRC) to Whitmer (GPC) dated October 4, 1978 / Enclosure 2, Section II.A HNP Response to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" dated October 27, 1976 / Section IV.A.4

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.14	All fire protection water supply and fire suppression system control valves shall be under a periodic inspection program and shall be supervised by one of the following methods. (a) Electrical supervision with audible and visual signals in the main control room or other suitable constantly attended location. (b) Locking valves in their normal position. Keys shall be made available only to authorized personnel. (c) Sealing valves in their normal positions. This option shall be utilized only where valves are located within fenced areas or under the direct control of the owner/operator.	Complies	No Additional Clarification	A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All Procedure 42SV-FPX-035-1, Fire Protection Valve Cycling Surveillance, Ver. 5.9 / All Procedure 42SV-FPX-035-2, Fire Protection Valve Cycling Surveillance, Ver. 3.13 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.15	Hydrants shall be installed approximately every 250 ft (76 m) apart on the yard main system. A hose house equipped with hose and combination nozzle and other auxiliary equipment specified in NFPA 24, Standard for the Installation of Private Fire Service Mains and Their Appurtenances, shall be provided at intervals of not more than 1000 ft (305 m) along the yard main system. Exception: Mobile means of providing hose and associated equipment, such as hose carts or trucks, shall be permitted in lieu of hose houses. Where provided, such mobile equipment shall be equivalent to the equipment supplied by three hose houses.	Complies	No Additional Clarification	Calculation SMNH-16-031, NFPA 24 Code Compliance Review, Ver. 1 / All Drawing H11150 SH. 1, Fire Protection - Piping Yard Underground System, Ver. 39.0 / All Drawing H11150 SH. 2, Fire Protection - Piping - Yard Underground System, Ver. 10 / All Drawing H11150 SH. 3, Fire Protection - Piping Yard Underground System, Ver. 11.0 / All Drawing H11150 SH. 4, Fire Protection - Piping - Yard Underground System, Rev. 5 / All NFPA 24, Standard for Outside Protection, 1973 Edition / All Procedure 42SV-FPX-022-0, Fire Hydrants and Hydrant Hose Houses, Ver. 2.9 / Attachment 1 Procedure DI-FPX-02-0693, Fire Fighting Equipment Inspection, Ver. 13.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.5.16	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.	Complies, with Required Action	Implementation items are identified below.	None
	IMPLEMENTATION ITEMS (See Attachment S, Table S-3): IMP-5 Site documentation will be revised to ensure that fire protection water is not used for non-fire-protection purposes except to provide backup to nuclear safety systems, provided an approved analysis concludes that the combined fire and nuclear safety flow demands can be maintained.			
		Submit for NRC Approval	NRC approval of non-fire-protection use of the fire protection water supply system is being requested in Attachment L, Approval Request 8.	None

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.6 Standpipe and Hose Stations.	N/A	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	N/A
3.6.1	For all power block buildings, Class III standpipe and hose systems shall be installed in accordance with NFPA 14, Standard for the Installation of Standpipe, Private Hydrant, and Hose Systems.	Complies with Use of EEEE's	The standpipe and hose system is designed in accordance with NFPA 14 as identified in Calculation SMNH-16-027, NFPA 14 Code Compliance Review. Calculation SMNH-16-047 documents the equivalency of hose stations that do not meet NFPA 14 requirements. Calculation SMNH-16-067 documents that the hose station coverage in the Intake Structure provides an equivalent level of protection.	Calculation SMNH-16-027, NFPA 14 Code Compliance Review, Ver. 1 / All Calculation SMNH-16-047, Engineering Evaluation of Fire Hose Station Coverage in the Cable Spreading Room, Unit 2 Chiller Room, and Unit 1 Stack Monitoring Room, Ver. 1 / All Calculation SMNH-16-067, Hydraulic Calculation and Evaluation of Hose Stations and Water Spray / Sprinkler System - Intake Structure, Ver. 1 / All NFPA 14, Standard for the Installation of Standpipe and Hose Systems, 1983 Edition / All
		Submit for NRC Approval	NRC approval of the 1.5 inch hose connections in the Reactor Buildings is being requested in Attachment L, Approval Request 9.	None

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.6.2	A capability shall be provided to ensure an adequate water flow rate and nozzle pressure for all hose stations. This capability includes the provision of hose station pressure reducers where necessary for the safety of plant industrial fire brigade members and off-site fire department personnel.	Complies	No Additional Clarification	Calculation SMNH 00-011, Hose Nozzle Pressure Drop Analysis, Ver. 0 / All
3.6.3	The proper type of hose nozzle to be supplied to each power block area shall be based on the area fire hazards. The usual combination spray/straight stream nozzle shall not be used in areas where the straight stream can cause unacceptable damage or present an electrical hazard to fire-fighting personnel. Listed electrically safe fixed fog nozzles shall be provided at locations where high-voltage shock hazards exist. All hose nozzles shall have shutoff capability and be able to control water flow from full open to full closed.	Complies	No Additional Clarification	Procedure 42SV-FPX-024-0, Fire Hose Stations Appendix B Areas, Ver. 5.0 / Attachment 2 Procedure 52SV-FPX-004-0, Non-Appendix B Fire Hose Stations, Ver. 2.0 / Attachment 10 Training Document S-FP-PP-10400-07.1, Introduction to Fire Fighting, Rev. 7.1 / Slides 39 and 40

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.6.4	Provisions shall be made to supply water at least to standpipes and hose stations for manual fire suppression in all areas containing systems and components needed to perform the nuclear safety functions in the event of a safe shutdown earthquake (SSE).	Complies by Previous NRC Approval	The supply of water to standpipes and hose stations for manual fire suppression in all areas containing systems and components needed to perform the nuclear safety functions in the event of a safe shutdown earthquake has been assessed. Position E.3.d in Appendix A to BTP APCSB 9.5-1 did not require provisions to supply water at least to standpipes in the event of a safe shutdown earthquake (SSE) for plants which were issued construction permits prior to July 1, 1976. In the response to Appendix A to Branch Technical Position APCSB 9.5-1 for Units No. 1 and 2, item IV.A.4 Georgia Power stated: "The fire protection systems for HNP were not designed as seismic Category I systems. The water systems are designed such that no active failure will prevent water from being provided to both hose stations and sprinkler or deluge systems that receive water from the fire protection water tanks." The NRC Safety Evaluation Report dated October 4, 1978, which was issued in response to the GPC letter dated October 27, 1976, stated: "We have reviewed the design criteria and bases for the water suppression systems and conclude that these systems meet the guidelines of Appendix A to Branch Technical Position 9.5.1 and are in accord with the applicable portions of the National Fire Protection Association (NFPA) Codes, and are, therefore, acceptable." The basis for approval has been reviewed. There have been no plant modifications or other changes that would invalidate the basis for approval.	Letter from Ippolito (NRC) to Whitmer (GPC) dated October 4, 1978 / Enclosure 2, Section II.A HNP Response to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" dated October 27, 1976 / Section IV.A.4

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.6.5	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.	Complies by Previous NRC Approval	Seismic required hose stations have been assessed at HNP. Position E.3.d in Appendix A to BTP APCSB 9.5-1 did not require provisions to supply water at least to standpipes in the event of a safe shutdown earthquake (SSE) for plants which were issued construction permits prior to July 1, 1976. In the response to Appendix A to Branch Technical Position APCSB 9.5-1 for Units No. 1 and 2, item IV.A.4 Georgia Power stated: "The fire protection systems for HNP were not designed as seismic Category I systems. The water systems are designed such that no active failure will prevent water from being provided to both hose stations and sprinkler or deluge systems that receive water from the fire protection water tanks." The NRC Safety Evaluation Report dated October 4, 1978, which was issued in response to the GPC letter dated October 27, 1976, stated: "We have reviewed the design criteria and bases for the water suppression systems and conclude that these systems meet the guidelines of Appendix A to Branch Technical Position 9.5.1 and are in accord with the applicable portions of the National Fire Protection Association (NFPA) Codes, and are, therefore, acceptable." The basis for approval has been reviewed. There have been no plant modifications or other changes that would invalidate the basis for approval.	Letter from Ippolito (NRC) to Whitmer (GPC) dated October 4, 1978 / Enclosure 2, Section II.A HNP Response to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" dated October 27, 1976 / Section IV.A.4

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.7 Fire Extinguishers.	Where provided, fire extinguishers of the appropriate number, size, and type shall be provided in accordance with NFPA 10, Standard for Portable Fire Extinguishers. Extinguishers shall be permitted to be positioned outside of fire areas due to radiological conditions.	Complies, with Required Action	Fire extinguishers are provided in accordance with NFPA 10 as identified in Calculation SMNH-16-024, NFPA 10 Code Compliance Review. Additional fire extinguishers will be installed in the control, turbine, and reactor buildings to ensure compliance with Class B fire size and placement requirements of NFPA 10. (See Attachment S, Table S-2, Item 4.)	Calculation SMNH-16-024, NFPA 10 Code Compliance Review, Ver. 1 / All Drawing Series A-43965, Pre-Fire Plan for Powerblock Areas / All Drawing Series A-43966, Pre-Fire Plan for Non-Powerblock Areas / All NFPA 10, Standard for Portable Fire Extinguishers, 1975 Edition / All
3.8 Fire Alarm and Detection Systems.	N/A	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.8.1 Fire Alarm.	Alarm initiating devices shall be installed in accordance with NFPA 72, National Fire Alarm Code®. Alarm annunciation shall allow the proprietary alarm system to transmit fire-related alarms, supervisory signals, and trouble signals to the control room or other constantly attended location from which required notifications and response can be initiated. Personnel assigned to the proprietary alarm station shall be permitted to have other duties. The following fire-related signals shall be transmitted:	Complies with Use of EEEE's	Alarm initiating devices comply with NFPA 72D and NFPA 72E as identified in Calculation SMNH-16-036, NFPA 72D Code Compliance Review and Calculation SMNH-16-037, NFPA 72E Code Compliance Review. The proprietary alarm system transmits fire-related alarms, supervisory signals, and trouble signals to the supervisory fire protection system. The system information is also transmitted to the master panel located in the constantly attended main control room. Per A-42162, fire-related alarms such as plant suppression and detection are monitored by the centralized electronic monitoring and supervisory fire protection system, commonly referred to as the XL3. The basic communication component of the system is the TRX, an addressable electronic module. The TRX modules report to the XL3 slave panels, which transmit system information to the master panel located in the constantly attended main control room. For each panel, indication is provided for a "Trouble" condition. Additionally, "Trouble" is also indicated on an alphanumeric display which provides the device number and location plus the diagnosis of the trouble to operators.	Calculation SMNH-16-036, NFPA 72D Code Compliance Review, Ver. 1 / All Calculation SMNH-16-037, NFPA 72E Code Compliance Review, Ver. 1 / All NFPA 72D, Standard for the Installation, Maintenance and Use of Proprietary Protectuve Signaling Systems, 1979 Edition / All NFPA 72E, Standard on Automatic Fire Detectors, 1982 Edition / All A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All
		Complies, with Required Action	Implementation items are identified below.	None

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

IMP-6 Inspection, testing, and maintenance procedures will be revised to include semi-annual testing of tank water level devices on both fire protection water storage tanks.

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.8.1(1)	Actuation of any fire detection device	Complies	No Additional Clarification	A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All
3.8.1(2)	Actuation of any fixed fire suppression system	Complies	No Additional Clarification	A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All
3.8.1(3)	Actuation of any manual fire alarm station	N/A	The detection systems credited in the NFPA 805 fire protection program do not employ manual fire alarm stations.	N/A
3.8.1(4)	Starting of any fire pump	Complies	No Additional Clarification	A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All
3.8.1(5)	Actuation of any fire protection supervisory device	Complies	No Additional Clarification	A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.8.1(6)	Indication of alarm system trouble condition	Complies	No Additional Clarification	A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All
3.8.1.1	Means shall be provided to allow a person observing a fire at any location in the plant to quickly and reliably communicate to the control room or other suitable constantly attended location.	Complies	No Additional Clarification	Edwin I. Hatch Nuclear Plant Final Safety Analysis Reports Update, Rev. 36 / HNP-2-FSAR-9.5.2
3.8.1.2	Means shall be provided to promptly notify the following of any fire emergency in such a way as to allow them to determine an appropriate course of action:	N/A	N/A - No technical requirements. See subsections for specific compliance statements and references.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.8.1.2(1)	General site population in all occupied areas	Complies	No Additional Clarification	Standard Emergency Plan Annex for Hatch Nuclear Plan Units 1 and 2, Ver. 2 / Section 4.3.1 Southern Nuclear Operating Company Standard Emergency Plan, Ver. 2 / Section E.2.1 and J.1 Procedure 34AB-X43-001-1, Fire Procedure, Ver. 15.1 / Section 4.4 Procedure 34AB-X43-001-2, Fire Procedure, Ver. 18.1 / Section 4.4
3.8.1.2(2)	Members of the industrial fire brigade and other groups supporting fire emergency response	Complies	No Additional Clarification	Standard Emergency Plan Annex for Hatch Nuclear Plan Units 1 and 2, Ver. 2 / Section 4.3.1 Southern Nuclear Operating Company Standard Emergency Plan, Ver. 2 / Sections E.2.1 and J.1 Procedure 34AB-X43-001-1, Fire Procedure, Ver. 15.1 / Section 4.4 Procedure 34AB-X43-001-2, Fire Procedure, Ver. 18.1 / Section 4.4

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.8.1.2(3)	Off-site fire emergency response agencies. Two independent means shall be available (e.g., telephone and radio) for notification of off-site emergency services.	Complies	No Additional Clarification	Standard Emergency Plan Annex for Hatch Nuclear Plan Units 1 and 2, Ver. 2 / Section 5.3 Procedure 34SO-R51-001-0, Communications System, Ver. 3.0 / All
3.8.2 Detection.	If automatic fire detection is required to meet the performance or deterministic requirements of Chapter 4, then these devices shall be installed in accordance with NFPA 72, National Fire Alarm Code, and its applicable appendixes.	Complies with Use of EEEE's	Automatic fire detection is installed in accordance with NFPA 72E as identified in Calculation SMNH-16-037, NFPA 72E Code Compliance Review. Calculation SMNH-16-044 documents the acceptability of ionization spot-type smoke detectors that are located near air supply registers contrary to Section 4.4.1 of NFPA 72.	Calculation SMNH-16-037, NFPA 72E Code Compliance Review, Ver. 1 / All Calculation SMNH-16-044, Engineering Evaluation of Smoke Detector Locations, Ver. 1 / All NFPA 72E, Standard on Automatic Fire Detectors, 1982 Edition / All
3.9 Automatic and Manual Water-Based Fire Suppression Systems.	N/A	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.9.1	If an automatic or manual water-based fire suppression system is required to meet the performance or deterministic requirements of Chapter 4, then the system shall be installed in accordance with the appropriate NFPA standards including the following:	N/A	N/A - No technical requirements. See subsections for specific compliance statements and references.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.9.1(1)	NFPA 13, Standard for the Installation of Sprinkler Systems	Complies with Use of EEEE's	Sprinkler systems are installed in accordance with NFPA 13 as identified in Calculation SMNH-16-026, NFPA 13 Code Compliance Review. Evaluation SMNH-16-046 documents the acceptability of sprinkler K-factors and temperatures, branch line and sprinkler spacing and spray pattern obstructions that do not meet NFPA 13. Evaluation SMNH-16-049 documents the adequacy of the application density for the pre-action sprinkler system (2U43112W02) protecting the Unit 2 Area Under Main Condenser (Fire Zone 2101A). Evaluation SMNH-16-063 documents the acceptability of the bracing and hangers on sprinkler systems in Power Block buildings that do not meet the seismic ratings of NFPA 13. Evaluation SMNH-16-064 documents the acceptability of sprinkler locations and spray patterns that do not meet the applicable NFPA 13 code requirements. Calculation SMNH-16-067 documents the functionality of hose station supply and the pressure and application density of the suppression system in the Intake Structure. Evaluation SMNH-16-069 documents the adequacy of the application density for the sprinkler system (1Z43130W03) protecting the Unit 1 Oil Conditioner Room (Fire Area 1023). Evaluation SMNH-16-071 documents the adequacy of the application density for the sprinkler system (2Z43130W23) protecting the Unit 2 Oil Conditioner Room (Fire Area 2023). Evaluation SMNH-16-074 documents the adequacy of the sprinkler system in the Unit 1 HPCI Pump Room.	Calculation SMNH-16-026, NFPA 13 Code Compliance Review, Ver. 1 / All Calculation SMNH-16-046, Engineering Evaluation of Sprinkler Systems in Fire Zones 0702B, 1023, 2101K, 2205Z, and 2301J, Ver. 1 / All Calculation SMNH-16-049, Hydraulic Calculation and Evaluation of Sprinkler System Application Density and Spacing - Unit 2 Turbine Building Area Under Main Condenser, Ver. 1 / All Calculation SMNH-16-063, Engineering Evaluation of Sprinkler System Restraint Configurations, Ver. 1 / All Calculation SMNH-16-064, Engineering Evaluation of Sprinkler Location and Spray Pattern Configurations, Ver. 1 / All Calculation SMNH-16-067, Hydraulic Calculation and Evaluation of Hose Stations and Water Spray / Sprinkler System - Intake Structure, Ver. 1 / All Calculation SMNH-16-069, Hydraulic Calculation and Evaluation of Sprinkler System Application Density - Unit 1 Oil Conditioner Room Sprinkler System, Ver. 1 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
			Evaluation SMNH-16-098 documents the adequacy of the application density for the pre-action sprinkler system (1U43112W02) protecting the Unit 1 Area Under Main Condenser (Fire Zone 1101A). Evaluation SMNH-16-099 documents the adequacy of the wet pipe sprinkler system (1U43130W01) protecting the Unit 1 East Cableway (Fire Area 1104). Evaluation SMNH-16-100 documents the adequacy of the application density for the sprinkler system (1U43130W04) protecting the Unit 1 Main Condenser Area (Fire Zone 1101K). Evaluation SMNH-16-106 documents the adequacy of the application density for the sprinkler system (2U43130W04) protecting the Unit 2 Main Condenser Area (Fire Zone 2101K).	Calculation SMNH-16-071, Hydraulic Calculation and Evaluation of Sprinkler System Application Density - Unit 2 Oil Conditioner Room Sprinkler System, Ver. 1 / All Calculation SMNH-16-074, Hydraulic Calculation and Evaluation of Sprinkler System Application Density Unit 1 HPCI Pump Room Sprinkler System, Ver. 1 / All Calculation SMNH-16-098, Hydraulic Calculation and Evaluation of Sprinkler System Application Density - Unit 1 Turbine Building Area Under Main Condenser, Ver. 1 / All Calculation SMNH-16-099, Hydraulic Calculation and Evaluation of Sprinkler System Application Density - Unit 1 Turbine Building East Cableway Sprinkler System, Ver. 1 / All Calculation SMNH-16-100, Hydraulic Calculation and Evaluation of Sprinkler System Application Density - Unit 1 Turbine Building Main Condenser Area, Ver. 1 / All Calculation SMNH-16-106, Hydraulic Calculation and Evaluation of Sprinkler System Application Density - Unit 2 Turbine Building Main Condenser Area, Ver. 1 / All NFPA 13, Standard for Installation of Sprinkler Systems, 1983 Edition / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
		Complies, with Required Action	Bypass lines or pressure relief trim kits will be installed to prevent pressure buildup in wet pipe sprinkler systems subject to excessive static pressures. (See Attachment S, Table S-2, Item 3.) Implementation items are identified below.	Calculation SMNH-16-026, NFPA 13 Code Compliance Review, Ver. 1 / All NFPA 13, Standard for Installation of Sprinkler Systems, 1983 Edition / All

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

IMP-7 Sprinkler system testing procedures will be revised to ensure inspectors' test connections are appropriately sized during system testing.

IMP-8 Update plant documentation to perform periodic internal sprinkler piping obstruction testing and monitoring.

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.9.1(2)	NFPA 15, Standard for Water Spray Fixed Systems for Fire Protection	Complies with Use of EEEE's	Water spray systems are installed in accordance with NFPA 15 as identified in Calculation SMNH-16-028, NFPA 15 Code Compliance Review. Calculation SMNH-16-062 evaluates the separation distances between the transformers and the Turbine building and documents that the current configuration is adequate for the hazards. Calculation SMNH-16-067 documents the functionality of hose station supply and the pressure and application density of the suppression system in the Intake Structure. Evaluation SMNH-16-107 documents the adequacy of the application density for the water spray system (2X43129W03) protecting the 500 kV Auto Transformers (Fire Area 0801).	Calculation SMNH-16-028, NFPA 15 Code Compliance Review, Ver. 1 / All Calculation SMNH-16-062, Engineering Evaluation of NFPA 805 Power Block Building Separation, Ver. 1 / All Calculation SMNH-16-067, Hydraulic Calculation and Evaluation of Hose Stations and Water Spray / Sprinkler System - Intake Structure, Ver. 1 / All Calculation SMNH-16-107, Hydraulic Calculation and Evaluation of Water Spray System - 500 kV Auto Transformers, Ver. 1 / All NFPA 15, Standard for Water Spray Fixed Systems for Fire Protection, 1982 Edition / All NFPA 15, Standard for Water Spray Fixed Systems for Fire Protection, 2001 Edition / All NFPA 15, Standard for Water Spray Fixed Systems for Fire Protection, 2007 Edition / All NFPA 15, Standard for Water Spray Fixed Systems for Fire Protection, 2012 Edition / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.9.1(3)	NFPA 750, Standard on Water Mist Fire Protection Systems	N/A	There are no water mist fire protection systems required to meet the performance-based or deterministic requirements of Chapter 4.	N/A
3.9.1(4)	NFPA 16, Standard for the Installation of Foam-Water Sprinkler and Foam-Water Spray Systems	N/A	There are no foam-water sprinkler or foam-water spray systems required to meet the performance-based or deterministic requirements of Chapter 4.	N/A
3.9.2	Each system shall be equipped with a water flow alarm.	Complies	No Additional Clarification	A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All Procedure 42SV-FPX-001-0, Surveillance of Low Voltage Switchyard Fixed Water Spray Systems, Ver. 4.7 / Section 5.0 Procedure 52SV-FPX-016-1, Sprinkler System Surveillance-Safety Related Areas, Ver. 1.8 / Section 7
3.9.3	All alarms from fire suppression systems shall annunciate in the control room or other suitable constantly attended location.	Complies	No Additional Clarification	A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.9.4	Diesel-driven fire pumps shall be protected by automatic sprinklers.	Complies	No Additional Clarification	Drawing H-40279, Fire Prot. Pumphouse Sprinkler System, Rev. 1 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.9.5	Each system shall be equipped with an OS&Y gate valve or other approved shutoff valve.	Complies	No Additional Clarification	Drawing H-11304 Sh. 1, Fire Protection Piping E & W Cableways Sprinkler Sys P&ID, Rev. 17 / All Drawing H-11304 Sh. 10, Fire Protection - P&ID Sprinkler System 1FH-73 Fire Protection Pumphouse, Rev. 8 / All Drawing H-11304 Sh. 2, Fire Protection P&ID - Reactor Feed Pump & R.F.P. Oil Cond., Rev. 16 / All Drawing H-11304 Sh. 3, Cond Below El. 130'-0", Ver. 20.0 / All Drawing H-11304 Sh. 4, Fire Prot P&ID Sprinkler System West End of Cond A&B below El. 164'-0", Rev. 11 / All Drawing H-11304 Sh. 5, Fire Protection - P&ID HPCI & RCIC Pump & Turbine Room, Rev. 15 / All Drawing H-11304 Sh. 6, Fire Protection P&ID - Fuel Oil @ S.U. Boiler & Lube Oil Storage Tanks, Rev. 13 / All Drawing H-11304 Sh. 7, Fire Protection P&ID E & W Recirc. Pump, ASD & Dry Waste Storage, Ver. 18.0 / All Drawing H-11322, Fire Protection Piping & Instrument Diagram - Intake Structure, Rev. 12.0 / All Drawing H-11323, Units 1 & 2 Fire Protection Piping-P&ID Cable Spreading Room, Rev. 12 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
				Drawing H-11324, Fire Protection-Piping-P&ID HVAC Room - Unit No. 1, Rev. 17 / All
				Drawing H-11325, Fire Protection P&ID - RPS Rooms & Corridors, Rev. 13 / All
				Drawing H-11326, Fire Protection-Piping-P&ID HVAC Room, Rev. 8 / All
				Drawing H-21016, Fire Protection Piping & Instrument Diagram - Turbine Buidling, Rev. 19.0 / All
				Drawing H-21017, Fire Protection Piping & Instrument Diagram - Reactor & Radwaste Buildings, Ver. 18.0 / All
				Drawing H-21189, Turbine Bldg. Fire Protection for Turbine Bearings P&ID, Rev. 5 / All
				Drawing H-21196, Reactor Building Fire Protection Sys. P&ID Sh. 1 of 2, Ver. 7.0 / All
				Drawing H-21197, Reactor Building Fire Protection Sys. P&ID Sh. 2 of 2, Ver. 9.0 / All
				Drawing H-21198, Turbine Building Fire Protection Sys. P&ID Sh. 1 of 5, Ver. 13.0 / All
				Drawing H-21199, Turbine Building Fire Protection System P&ID Sheet 2 of 5, Ver. 9.0 / All
				Drawing H-21200, Turbine Building Fire Protection System P&ID Sht 3 of 5, Ver. 7.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
				Drawing H-26372, Fire Protection Piping Radwaste Bldg Drywaste Storage Area Sprinkler Sys P&ID, Rev. 6 / All
				Drawing H-26377, Fire Protection P&ID - East Cableway - El. 130'-0", Ver. 10.0 / All
				Drawing H-40390, Fire Prot - P&ID - Reactor Bldg El. 87'-0" - East Torus Water Curtain, Rev. 1 / All
				Drawing H-40391, Fire Prot. - P&ID - Reactor Bldg - FL EL. 87'-0" - West Torus Water Curtain, Rev. 2 / All
				Drawing H-40392, Fire Prot - P&ID - Reactor Bldg FL EL. 130'-0" East 130 Water Curtain, Ver. 6.0 / All
				Drawing H-40393, Fire Protection - P&ID Reactor Bldg. FL. EL. 130'-0" Northwest 130 Water Curtain, Rev. 4.0 / All
				Drawing H-40394, Fire Protection - P&ID Reactor Bldg. FL. EL 130'-0" Southwest 103 Water Curtain, Ver. 4.0 / All
				Drawing H-40395, Fire Protection - P&ID Reactor Bldg. - Floor EL. 158'-0" East 158 Water Curtain, Ver. 3.0 / All
				Drawing H-40396, Fire Protection - P&ID Reactor Bldg. FL. EL. 185'-0" East 185 Water Curtain, Rev. 3 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
				Drawing H-40397, Fire Prot - P&ID - Reactor Bldg FL. EL. 185'-0" - West 185 Water Curtain, Rev. 1 / All
				Drawing H-40406, Fire Protection P&ID - Control Bldg. North Corridor FL. EL. 130'-0", Ver. 5.0 / All
				Drawing H-50041, Fire Prot. - P&ID - Reactor Bldg. FL. EL. 87'-0" East Torus Water Curtain, Ver. 3.0 / All
				Drawing H-50042, Fire Protection - P&ID Reactor Building FL. EL. 87'-0" West Torus Water Curtain, Ver. 4.0 / All
				Drawing H-50043, Fire Prot. - P&ID - Reactor Bldg. FL. EL. 130'-0" - East 130 Water Curtain, Ver. 3.0 / All
				Drawing H-50044, Fire Prot. - P&ID - Reactor Bldg. FL. EL. 130'-0" - NW 130 Water Curtain, Ver. 3.0 / All
				Drawing H-50045, Fire Prot. - P&ID - Reactor Bldg FL. EL. 158'-0" - East 158 Wtr Curtain, Rev. 3 / All
				Drawing H-50046, Fire Protection - P&ID - Reactor Bldg. Floor El. 185'-0" East 185 Water Curtain, Rev. 1 / All
				Drawing H-50047, Fire Protection - P&ID Reactor Bldg. - Floor El. 185'-0" West 185 Water Curtain, Ver. 3.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
				Drawing H-50051, Fire Protection - P&ID Control Building - South Corridor Floor Elevation 130'-0", Ver. 6.0 / All Drawing H-50276, Fire Protection-P&ID-Reactor Bldg. Floor EL. 185'-0" East 185 Water Curtain, Rev. 1 / All
3.9.6	All valves controlling water-based fire suppression systems required to meet the performance or deterministic requirements of Chapter 4 shall be supervised as described in 3.5.14.	Complies	No Additional Clarification	Procedure 42SV-FPX-035-1, Fire Protection Valve Cycling Surveillance, Ver. 5.9 / All Procedure 42SV-FPX-035-2, Fire Protection Valve Cycling Surveillance, Ver. 3.13 / All
3.10 Gaseous Fire Suppression Systems.	N/A	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.10.1	If an automatic total flooding and local application gaseous fire suppression system is required to meet the performance or deterministic requirements of Chapter 4, then the system shall be designed and installed in accordance with the following applicable NFPA codes:	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	N/A
3.10.1(1)	NFPA 12, Standard on Carbon Dioxide Extinguishing Systems	Complies with Use of EEEE's	The carbon dioxide systems are installed in accordance with NFPA 12 as identified in Calculation SMNH-16-025, NFPA 12 Code Compliance Review.	Calculation SMNH-16-025, NFPA 12 Code Compliance Review, Ver. 1 / All NFPA 12, Standard on Carbon Dioxide Extinguishing Systems, 1973 Edition / All
3.10.1(2)	NFPA 12A, Standard on Halon 1301 Fire Extinguishing Systems	N/A	There are no Halon 1301 fire extinguishing systems required to meet the performance-based or deterministic requirements of Chapter 4.	Fire Hazards Analysis, Rev. 36 / All
3.10.1(3)	NFPA 2001, Standard on Clean Agent Fire Extinguishing Systems	N/A	There are no clean agent fire extinguishing systems required to meet the performance-based or deterministic requirements of Chapter 4.	Fire Hazards Analysis, Rev. 36 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.10.2	Operation of gaseous fire suppression systems shall annunciate and alarm in the control room or other constantly attended location identified.	Complies	No Additional Clarification	A-42162, Unit No. 1 / 2 Fire Protection Detection/Annunciation Multiplex Database, Rev. 10 / All
3.10.3	Ventilation system design shall take into account prevention from over-pressurization during agent injection, adequate sealing to prevent loss of agent, and confinement of radioactive contaminants.	Complies	No Additional Clarification	Drawing H-12618, Arch. - Diesel Generator Bldg. - Heating & Ventilation - Plans and Details, Ver. 5.0 / All Drawing H-12619, Architectural - Generator Building Heating & Ventilating General Arrangement & Parts Numbers, Ver. 12.0 / All Drawing H-16055, Control Bldg. HVAC System Plan at El. 147'-0", Rev. 15 / All Drawing SX-11788, FP-LP CO2-Diagram Diesel Generating Building System, Ver. 1.0 / All Fire Hazards Analysis, Rev. 36 / Section 4.11

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.10.4	In any area required to be protected by both primary and backup gaseous fire suppression systems, a single active failure or a crack in any pipe in the fire suppression system shall not impair both the primary and backup fire suppression capability.	N/A	HNP does not have any areas required to be protected by both primary and backup gaseous fire suppression systems.	N/A
3.10.5	Provisions for locally disarming automatic gaseous suppression systems shall be secured and under strict administrative control.	Complies	No Additional Clarification	Procedure 52SV-FPX-010-0, Low Pressure CO2 System Surveillance, Ver. 7.3 / All Procedure NMP-AD-003, Equipment Clearance and Tagging, Ver. 22.1 / All
3.10.6	Total flooding carbon dioxide systems shall not be used in normally occupied areas.	Complies	No Additional Clarification	Drawing SX-11786 Sheet 1, FP-LP CO2 Diagram Turbine and Control Building, Rev. H / All Drawing SX-11788, FP-LP CO2-Diagram Diesel Generating Building System, Ver. 1.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.10.7	Automatic total flooding carbon dioxide systems shall be equipped with an audible pre-discharge alarm and discharge delay sufficient to permit egress of personnel. The carbon dioxide system shall be provided with an odorizer.	Complies	No Additional Clarification	Drawing H-14191, Elementary Diagram Fire Protection System Diesel Bldg, Ver. 15.0 / All Drawing H-14193, Elementary Diagram Fire Protection System Computer Room, Cable Spreading Room, Ver. 17.0 / All Drawing H-23783, Elementary Diagram-Fire Protection System 2X43B Diesel Building, Ver. 4.0 / All Drawing H-41508, Turbine & Control Building - Fire Protection Piping - CO2 System P&ID, Rev. 8 / All Drawing H-41509, Diesel Generator Building CO2 System-P&ID, Rev. 5 / All
3.10.8	Positive mechanical means shall be provided to lock out total flooding carbon dioxide systems during work in the protected space.	Complies	No Additional Clarification	Procedure 52SV-FPX-010-0, Low Pressure CO2 System Surveillance, Ver. 7.3 / Section 7 Vendor Manual SX11785, FP-LP CO2-Operating & SVCE Manual-CO2 System, Ver. 1.0 / Section III

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.10.9	The possibility of secondary thermal shock (cooling) damage shall be considered during the design of any gaseous fire suppression system, but particularly with carbon dioxide.	Complies by Previous NRC Approval	In the response to Appendix A to Branch Technical Position APCSB 9.5-1 for Units No. 1 and 2, item IV.C.5 Georgia Power stated: "Total CO2 flooding systems are being utilized in the Cable Spreading Room, and Computer Room, and Diesel Generator Room which minimize the cooling effects on equipment in these rooms as opposed to sudden cooling by a direct CO2 impingement by nozzles directed at specific components." The NRC Safety Evaluation Report dated October 4, 1978 stated: "Low pressure carbon dioxide flooding systems have been provided for the following areas: (a) Emergency diesel generator rooms; (b) Cable spreading room; and (c) Computer room. Also, manual CO2 hose stations have been provided in the electrical switchgear areas. The CO2 suppression systems are designed according to NFPA Standard No. 12, Carbon Dioxide Extinguishing Systems. We have reviewed the design criteria and basis for these fire suppression systems. We conclude that these systems satisfy the provisions of Appendix A to Branch Technical Position 9.5.1 and are provided in accordance with the applicable portions of the National Fire Protection Associate Code and are, therefore acceptable." The basis for approval has been reviewed. There have been no plant modifications or other changes that would invalidate the basis for approval.	Letter from Ippolito (NRC) to Whitmer (GPC) dated October 4, 1978 / Section II.B HNP Response to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" dated October 27, 1976 / Section IV.C.5

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.10.10	Particular attention shall be given to corrosive characteristics of agent decomposition products on safety systems.	Complies	No Additional Clarification	NFPA Fire Protection Handbook, 20th Edition, 20th ed / Section 17, Chapter 1 Fire Hazards Analysis, Rev. 36 / Section 4.0
3.11 Passive Fire Protection Features.	This section shall be used to determine the design and installation requirements for passive protection features. Passive fire protection features include wall, ceiling, and floor assemblies, fire doors, fire dampers, and through fire barrier penetration seals. Passive fire protection features also include electrical raceway fire barrier systems (ERFBS) that are provided to protect cables and electrical components and equipment from the effects of fire.	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.11.1 Building Separation.	Each major building within the power block shall be separated from the others by barriers having a designated fire resistance rating of 3 hours or by open space of at least 50 ft (15.2 m) or space that meets the requirements of NFPA 80A, Recommended Practice for Protection of Buildings from Exterior Fire Exposures. Exception: Where a performance-based analysis determines the adequacy of building separation, the requirements of 3.11.1 shall not apply.	Complies with Use of EEEE's	Except as identified in SMNH-16-062, each major building within the power block is separated from the others by either 3-hour fire-rated barriers or at least 50 ft of open space. Calculation SMNH-16-062 documents the acceptability of configurations where buildings are not separated by 3-hour fire-rated barriers or 50 feet of open space.	Calculation SMNH-16-062, Engineering Evaluation of NFPA 805 Power Block Building Separation, Ver. 1 / All Fire Hazards Analysis, Rev. 36 / Section 8.0
3.11.2 Fire Barriers.	Fire barriers required by Chapter 4 shall include a specific fire-resistance rating. Fire barriers shall be designed and installed to meet the specific fire resistance rating using assemblies qualified by fire tests. The qualification fire tests shall be in accordance with NFPA 251, Standard Methods of Tests of Fire Endurance of Building Construction and Materials, or ASTM E 119, Standard Test Methods for Fire Tests of Building Construction and Materials.	Complies	Except as identified below, HNP complies with no additional clarification.	Fire Hazards Analysis, Rev. 36 / Sections 8.0, 11.2, and 12.1 DCR94-44, Addition of Promat H Board, Rev. 0 / All Procedure 52GM-FPX-002-0, Promat H System Installation and Repair, Ver. 2.0 / All Report S55054, Technical Evaluation of Promat H One and Three Hour Fire Barriers, Rev. A / Material Suitability

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
		Complies with Use of EEEE's	Calculation SMNH-16-051 documents the acceptability of the depth of embedded raceways in concrete barriers in the Unit 1 and Unit 2 Control Building and Diesel Generator Building. Calculation SMNH-16-059 documents the acceptability of fire area boundaries within the Unit 1 and Unit 2 Reactor Buildings that are not adequately fire-rated. Calculation SMNH-16-060 documents the acceptability of fire area boundaries within the Unit 1 and Unit 2 Turbine Buildings that are not adequately fire-rated. Calculation SMNH-16-061 documents the acceptability of fire area boundaries within the Control Building that are not adequately fire-rated. Calculation SMNH-16-062 documents the acceptability of fire area boundaries between buildings that are not adequately fire-rated. Calculation SMNH-16-088 documents the acceptability of separation between the Service Building fan room and the remainder of the Service Building, and between the Service Building tunnel and the remainder of the Service Building.	Calculation SMNH-16-051, Engineering Evaluation of Embedded Raceways, Ver. 1 / All Calculation SMNH-16-059, Engineering Evaluation of Fire Area Boundaries within the Unit 1 and Unit 2 Reactor Buildings, Ver. 1 / All Calculation SMNH-16-060, Engineering Evaluation of Fire Area Boundaries within the Unit 1 and Unit 2 Turbine Buildings, Ver. 1 / All Calculation SMNH-16-061, Engineering Evaluation of Fire Area Boundaries within the Control Building, Ver. 1 / All Calculation SMNH-16-062, Engineering Evaluation of NFPA 805 Power Block Building Separation, Ver. 1 / All Calculation SMNH-16-088, Engineering Evaluation of Separation of the Service Building Fan Room and Service Building Fan Room and Service Building Tunnel from the Remainder of the Service Building, Ver. 1 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.11.3 Fire Barrier Penetrations.	Penetrations in fire barriers shall be provided with listed fire-rated door assemblies or listed rated fire dampers having a fire resistance rating consistent with the designated fire resistance rating of the barrier as determined by the performance requirements established by Chapter 4. (See 3.11.3.4 for penetration seals for through penetration fire stops.) Passive fire protection devices such as doors and dampers shall conform with the following NFPA standards, as applicable: Exception: Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, a performance-based analysis shall be required to assess the adequacy of fire barrier forming the fire boundary to determine if the barrier will withstand the fire effects of the hazards in the area. Openings in fire barriers shall be permitted to be protected by other means as acceptable to the AHJ.	Complies	Except as identified below, HNP complies with no additional clarification.	Drawing B-19628, Unit 1 Reactor Building Penetration Seals Type, Number and As-Built Location / All (Series) Drawing B-19629, Unit 1 Turbine Building Penetration Seals Type, Number and As-Built Location / All (Series) Drawing B-19630, Unit 1 Radwaste Building Penetration Seals Type, Number and As-Built Location / All (Series) Drawing B-19631, Control Building Penetration Seals Type, Number and As-Built Location / All (Series) Drawing B-19632, Diesel Generator Building Penetration Seals Type, Number and As-Built Location / All (Series) Drawing B-23275, Unit 2 Reactor Building Penetration Seals Type, Number and As-Built Location / All (Series) Drawing B-23276, Unit 2 Turbine Building Penetration Seals Type, Number and As-Built Location / All (Series) Drawing B-23277, Unit 2 Radwaste Building Penetration Seals Type, Number and As-Built Location / All (Series) Form ENG-0149, Fire Damper Surveillance Results Units 1 & 2, Ver. 2.0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
				Letter October 23, 1978 E. I. Hatch Nuclear Plant Unit 1 & 2 Bechtel Job 6511-001/020 HVAC - Fire Dampers / All Fire Hazards Analysis, Rev. 36 / Section 8.0 Procedure 51GM-FPX-002-0, Installation and Repair of Sliding Fire Doors, Ver. 3.0 / All Specification SS-2115-2, Heating, Ventilating, and Air Conditioning of Edwin I. Hatch Nuclear Plant Unit 2, Rev. 0 / Appendix E Specification SS-6915-2, Heating, Ventilating, and Air Conditioning of Edwin I. Hatch Nuclear Plant Unit 1, Rev. 1 / Section 9.13

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
		Complies with Use of EEEE's	Calculation SMNH-16-059 documents the acceptability of fire area boundaries within the Unit 1 and Unit 2 Reactor Buildings, including HVAC openings lacking dampers.	Calculation SMNH-16-059, Engineering Evaluation of Fire Area Boundaries within the Unit 1 and Unit 2 Reactor Buildings, Ver. 1 / All
			Calculation SMNH-16-060 documents the acceptability of fire area boundaries within the Unit 1 and Unit 2 Turbine Buildings, including HVAC openings lacking dampers.	Calculation SMNH-16-060, Engineering Evaluation of Fire Area Boundaries within the Unit 1 and Unit 2 Turbine Buildings, Ver. 1 / All
			Calculation SMNH-16-061 documents the acceptability of fire area boundaries within the Control Building, including HVAC openings lacking dampers.	Calculation SMNH-16-061, Engineering Evaluation of Fire Area Boundaries within the Control Building, Ver. 1 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.11.3(1) Fire Barrier Penetrations.	NFPA 80, Standard for Fire Doors and Fire Windows	Complies with Use of EEEE's	Fire doors are in accordance with NFPA 80, Standard for Fire Doors and Fire Windows, as identified in Calculation SMNH-16-038, NFPA 80 Code Compliance Review. Evaluation SMNH-16-042 documents the acceptability of various fire door configurations that do not meet the applicable NFPA 805 and NFPA 80 code requirements.	Calculation SMNH-16-038, NFPA 80 Code Compliance Review, Ver. 1 / All Calculation SMNH-16-042, Engineering Evaluation of Fire Door Assemblies, Ver. 1 / All Drawing H-12652, Unit 1 Architectural Special Door Schedule and Details, Rev. 4 / All Drawing H-12676, Unit 1 Architectural Turbine Building Door Schedule, Ver. 13.0 / All Drawing H-15871, Unit 1 Architectural Door Schedule, Ver. 26.0 / All Drawing H-25973, Unit 2 Architectural Door Schedule, Rev. 27 / All Drawing H-25974, Unit 2 Architectural Turbine Building Door Schedule, Ver. 17.0 / All NFPA 80, Standard for Fire Doors and Windows, 1975 Edition / All Procedure 51GM-FPX-002-0, Installation and Repair of Sliding Fire Doors, Ver. 3.0 / All Procedure 51GM-FPX-003-0, Installation and Repair of Rolling Fire Doors, Ver. 1.3 / All Procedure 51GM-FPX-004-0, Installation and Repair of Rolling Fire Doors, Ver. 1.4 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
		Complies, with Required Action	Fusible links will be relocated and/or installed on sliding fire doors in compliance with the requirements of NFPA 80. (See Attachment S, Table S-2, Item 1.)	None
3.11.3(2) Fire Barrier Penetrations.	NFPA 90A, Standard for the Installation of Air-Conditioning and Ventilating Systems	Complies with Use of EEEE's	Fire dampers are in accordance with NFPA 90A, Standard for the Installation of Air-Conditioning and Ventilating Systems, as identified in Calculation SMNH-16-039, NFPA 90A Code Compliance Review. Calculation SMNH-16-059 documents the acceptability of fire area boundaries within the Unit 1 and Unit 2 Reactor Buildings, including HVAC openings lacking dampers. Calculation SMNH-16-060 documents the acceptability of fire area boundaries within the Unit 1 and Unit 2 Turbine Buildings, including HVAC openings lacking dampers. Calculation SMNH-16-061 documents the acceptability of fire area boundaries within the Control Building, including HVAC openings lacking dampers.	Calculation SMNH-16-039, NFPA 90A Code Compliance Review, Ver. 1 / All Calculation SMNH-16-059, Engineering Evaluation of Fire Area Boundaries within the Unit 1 and Unit 2 Reactor Buildings, Ver. 1 / All Calculation SMNH-16-060, Engineering Evaluation of Fire Area Boundaries within the Unit 1 and Unit 2 Turbine Buildings, Ver. 1 / All Calculation SMNH-16-061, Engineering Evaluation of Fire Area Boundaries within the Control Building, Ver. 1 / All NFPA 90A, Standard for the Installation of Air Conditioning and Ventilating Systems, 1976 Edition / All Procedure 52GM-FPX-001-0, HVAC Fire Damper Installation and Repair, Ver. 2.4 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.11.3(3) Fire Barrier Penetrations.	NFPA 101, Life Safety Code	Complies with Clarification	The requirements of NFPA 101 applicable to fire doors and fire dampers are bound by NFPA 80 and NFPA 90A. NFPA 101 Section 8.2.3.2.1 refers to NFPA 80 and NFPA 101 Section 9.2.1 refers to NFPA 90A.	Calculation SMNH-16-038, NFPA 80 Code Compliance Review, Ver. 1 / All Calculation SMNH-16-039, NFPA 90A Code Compliance Review, Ver. 1 / All NFPA 101, Life Safety Code, 2000 Edition / Sections 8.2.3.2.1 & 9.2.1
3.11.4 Through Penetration Fire stops.	Through penetration fire stops for penetrations such as pipes, conduits, bus ducts, cables, wires, pneumatic tubes and ducts, and similar building service equipment that pass through fire barriers shall be protected as follows.	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	N/A

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.11.4(a) Through Penetration Fire stops.	The annular space between the penetrating item and the through opening in the fire barrier shall be filled with a qualified fire-resistive penetration seal assembly capable of maintaining the fire resistance of the fire barrier. The assembly shall be qualified by tests in accordance with a fire test protocol acceptable to the AHJ or be protected by a listed fire-rated device for the specified fire-resistive period.	Complies with Use of EEEE's	Penetration seal assemblies are designed, qualified, and installed in accordance with the requirements in the referenced procedures. Fire tests that provide the design basis for seal attributes are noted in the referenced documents. Calculation SMNH 98-023 documents the acceptability of configurations with missing or non-rated penetration seals.	Calculation SMNH-98-023, Fire Protection Penetration Seal Deviation Analysis, Ver. 6 / All Procedure 42FP-FPX-001-0, Installation of Nelson Electric MCT Frames, Ver. 1.4 / All Procedure 42FP-FPX-003-0, Installation of Nelson Electric Fire Stops & Seals, Ver. 3.5 / All Procedure 42FP-FPX-014-0, Installation and Repair of Silicone Foam Seals, Ver. 6.0 / All Procedure 42FP-FPX-017-0, Installation & Repair of Flexible Boot Fire Seals, Ver. 0.5 / All Report S52429, E.I. Hatch Nuclear Plant Fire Rated Penetration Seal Qualification Data, Rev. 0 / All Report S52438, Fire Rated Penetration Seal Qualification Data-SWRI Project No.01-8821-028A, Rev. 0 / All Report S52439, E.I. Hatch Nuclear Plant Fire Rated Penetration Seal Qualification Data Nelson Firestop Products, Rev. 0 / All Report S52440, E.I. Hatch Nuclear Plant Fire Rated Penetration Seal Qualification Data Nelson Firestop Products, Rev. 0 / All Report S52441, Fire Rated Penetration Seal Qualification Data-SWRI Project No.01-8821-028D, Rev. 0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
				Report S52477, E.I. Hatch Nuclear Plan Fire Rated Penetration Seal Qualification Data Misc. Silicone Based Products, Rev. 0 / All
				Report S52478, Fire Rated Penetration Seal Qualification Data Chemtrol Fireshield Products, Rev. 0 / All
				Report S52479, Fire Rated Penetration Seal Qualification Data-Factory Mutual Projects WP-454 & WP-455, Rev. 0 / All
				Report S52480, Fire Rated Penetration Seal Qualification Data-Chemtrol Design FC-225, Rev. 0 / All
				Report S52481, Fire Rated Penetration Seal Qualification Data-UL Project 83NK19792, Rev. 0 / All
				Report S52483, Fire Rated Penetration Seal Qualification Data-Various Sealing Products & Configurations, Rev. 0 / All
				Report S80782, Design FC225 Guide Specification for Fire-Rated Penetration Seals-Fire-Stops-Seismic Gap Seals, Rev. 0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
		Complies, with Required Action	An annular gap seal will be installed for the cable tray penetrant in the floor/ceiling boundary between Fire Area 2101 and the telecommunications room in the Service Building. (See Attachment S, Table S-2, Item 2.)	None
3.11.4(b) Through Penetration Fire stops.	Conduits shall be provided with an internal fire seal that has an equivalent fire-resistive rating to that of the fire barrier through opening fire stop and shall be permitted to be installed on either side of the barrier in a location that is as close to the barrier as possible. Exception: Openings inside conduit 4 in. (10.2 cm) or less in diameter shall be sealed at the fire barrier with a fire-rated internal seal unless the conduit extends greater than 5 ft (1.5 m) on each side of the fire barrier. In this case the conduit opening shall be provided with noncombustible material to prevent the passage of smoke and hot gases. The fill depth of the material packed to a depth of 2 in. (5.1 cm) shall constitute an acceptable smoke and hot gas seal in this application.	Complies	No Additional Clarification	Procedure 42FP-FPX-003-0, Installation of Nelson Electric Fire Stops & Seals, Ver. 3.5 / All Report S55134, Topical Fire Test Report - Internal Conduit Seals - 3 Hour Test, Rev. 0 / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
3.11.5 Electrical Raceway Fire Barrier Systems (ERFBS)	ERFBS required by Chapter 4 shall be capable of resisting the fire effects of the hazards in the area. ERFBS shall be tested in accordance with and shall meet the acceptance criteria of NRC Generic Letter 86-10, Supplement 1, "Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Safe Shutdown Trains Within the Same Fire Area." The ERFBS needs to adequately address the design requirements and limitations of supports and intervening items and their impact on the fire barrier system rating. The fire barrier system's ability to maintain the required nuclear safety circuits free of fire damage for a specific thermal exposure, barrier design, raceway size and type, cable size, fill, and type shall be demonstrated. Exception No. 1: When the temperatures inside the fire barrier system exceed the maximum temperature allowed by the acceptance criteria of Generic Letter 86-10, "Fire Endurance Acceptance Test Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Training Within the Same Fire Area," Supplement 1, functionality of the cable at these elevated temperatures shall be demonstrated. Qualification demonstration of these cables	Complies	No Additional Clarification	DCR94-44, Addition of Promat H Board, Rev. 0 / All Specification S55054, Technical Evaluation of Promat H One and Three Hour Fire Barriers, Rev. A / All

NFPA 805 Ch. 3 Ref.	Requirements/Guidance	Compliance Statement	Compliance Basis	Reference Document
	shall be performed in accordance with the electrical testing requirements of Generic Letter 86-10, Supplement 1, Attachment 1, "Attachment Methods for Demonstrating Functionality of Cables Protected by Raceway Fire Barrier Systems During and After Fire Endurance Test Exposure." Exception No. 2: ERFBS systems employed prior to the issuance of Generic Letter 86-10, Supplement 1, are acceptable providing that the system successfully met the limiting end point temperature requirements as specified by the AHJ at the time of acceptance.			

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

117 Pages Attached

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 Section 3 Guidance
3.0	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. For a complete understanding of the deterministic requirements, work this section in combination with the information in Appendix C, High/Low Pressure Interfaces, Appendix D, Alternative and Dedicated Shutdown Requirements, Appendix E, Acceptance Criteria for Operator Manual Actions and repairs, and Appendix H, Hot Shutdown versus Important to Safe Shutdown Components. To resolve the industry issue related to MSOs, refer to Section 4, Appendix B, Appendix F and Appendix G. The plant specific analysis approved by NRC is reflected in the plant's licensing basis. The methodology described in this section is an acceptable method of performing a post-fire safe shutdown analysis. This methodology is depicted 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. The requirements of Appendix R Section III.G.2 do not apply to the circuits for fire detection and suppression systems, communications systems and 8-hour emergency lighting systems. Additional information is provided in Appendix B to this document related to the circuit failure criteria to be applied in assessing impacts to post-fire safe shutdown, including MSOs. The criteria in Appendix B developed for MSOs has also been included in Section 3.5.1.1 for assessing the potential affects of fire-induced impacts to individual components on the required safe shutdown path for a particular III.G.1 and 2 fire area. Section 4 provides the Resolution Methodology for determining the Plant Specific List of MSOs to be evaluated. Section 5 provides a focused-scope Fire PRA risk methodology for assessing, on an individual basis, the risk significance of any MSOs determined to be impacted within a common plant fire area. The appropriate use of these tools for mitigating the effects of fire-induced circuit failures for this section and for the MSOs addressed in Section 4 and Appendix G are discussed in Appendix H.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

A deterministic methodology was used to assess conformance with the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805 for the Edwin I. Hatch Nuclear Plant (HNP).

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

The HNP NFPA 805 Nuclear Safety Capability Assessment (NSCA) deterministic methodology has been reviewed in detail against the guidance, criteria, and assumptions contained within NEI 00-01, Chapter 3, as documented in the subsequent sections of this table (i.e. Table B-2 from NEI 04-02).

The results of this review conclude that the HNP NSCA has been performed consistent with (i.e., aligns with) the deterministic methodology guidance, criteria, and assumptions from Chapter 3 of NEI 00-01.

Unless otherwise noted, all the statements are applicable to HNP as a whole (i.e., Unit 1 and Unit 2). References to the following appendices and related terms such as "required for hot shutdown" and "important to SSD" are not applicable to NFPA 805 and will not be considered in subsequent sections of this review:

- Appendix F, "Supplemental Selection guidance (Discretionary)."
- Appendix H, "Required for hot shutdown versus important to SSD."

References to 10CFR50 Appendix R such as "III.G.1", "III.G.2", "III.G.3", "green box", "orange box" and the 72-hour coping period are not applicable to NFPA 805 and will not be considered in subsequent sections of this review.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.1	Safe Shutdown Systems and Path Development	This section discusses the identification of systems 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. 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. This section provides some guidance on classifying components as either required or important to SSD circuit components. It also provides some guidance on the tools available for mitigating the effects of fire-induced circuit failures to each of these classes of equipment. For a more detailed discussion of the topic of required and important to SSD components refer to Appendix H. The goal of post-fire safe shutdown is to assure that a 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 required 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 required for 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 (as defined in NRC Information Notice 84-09) - Support systems -- Electrical power and control systems -- Component Cooling systems -- Component Lubrication 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. The components required to perform these functions are classified as required for hot shutdown components. These components are necessary and sufficient to perform the required safe shutdown functions assuming that fire-induced impacts to other plant equipment/cables do not occur. Since fire-induced impacts to other plant equipment/cables can occur in the fire condition, these impacts must also be addressed. The components not necessary to complete the required safe shutdown functions, but which could be impacted by the fire and cause a subsequent impact to the required safe shutdown components are classified as either required for hot shutdown or important to SSD components. Depending on the classification of the components, the tools available for mitigating the affects of fire-induced damage vary. The available tools are generally discussed in this section and in detail in Appendix H. The classification of a component or its power or control circuits may vary from fire area to fire area. Therefore, the required safe shutdown path for any given fire area is comprised of required for hot shutdown components and important to SSD components. The distinction and classification for each required safe shutdown path for each fire area should be discernible in the post-fire safe shutdown analysis. Generic Letter 81-12 specifies consideration of associated circuits of concern with the potential for spurious equipment operation and/or loss of power source, and the common enclosure failures. As described above, spurious operations/actuations can affect the

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

accomplishment of the required 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. To address the issue of multiple spurious operations, Section 4 provides a Resolution Methodology for developing a Plant Specific List of MSOs for evaluation. Appendix B provides the circuit failure criteria applicable to the evaluation of the Plant Specific list of MSOs.

Common power source and common enclosure concerns could also affect the safe shutdown path and must be addressed.

In addition to the tools described for components classified as required for hot shutdown, fire-induced impacts to cables and components classified as important to SSD may be mitigated using some additional tools. For important to SSD component failures, operator manual actions, fire modeling and/or a focused-scope fire PRA may be used to mitigate the impact. (If the use of a Focused-Scope Fire PRAs is not permitted in the Plants Current License Basis, then, a License Amendment Request (LAR) will be necessary to use the Focused-Scope Fire PRA).

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Hatch Nuclear Plant systems / functions / components required to achieve and maintain safe and stable plant conditions post-fire per the Nuclear Safety Performance Criteria of NFPA 805 are identified in HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 8.6. These systems / functions / components are used to meet the performance goals listed above, which include Reactivity Control, Pressure Control, Inventory Control, Decay Heat Removal, Process Monitoring and Support Systems. Components that are not necessary to complete required safe shutdown functions, but which could be impacted by the fire and cause a subsequent impact to a required safe shutdown component are included as required SSD components.

The identification and analysis of these systems / functions / components includes addressing associated circuit issues for spurious operations, high/low pressure interfaces, common power supplies and common enclosures. Spurious operations and high/low pressure interface concerns are discussed in PI-02-001, Sections 6.3 and Appendix A. Common power supply and common enclosure concerns have been addressed through coordination calculations.

A computer database and analysis tool, ARCPlus, is utilized to demonstrate that the Nuclear Safety Performance Criteria of NFPA 805 is met for each fire area of the plant.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

Calculation SENH-17-002, Circuit Analysis for Fire Safety Analysis – Fire Data Manager Output, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1 Criteria/Assumptions	The following criteria and assumptions should be considered, as applicable, when identifying systems available and necessary to perform the required safe shutdown functions and combining these systems into safe shutdown paths. This list provides recognized examples of criteria/assumptions but should not be considered an all-inclusive list. The final set of criteria/assumptions should be based on regulatory requirements and the performance criteria for post-fire safe shutdown for each plant.

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This is an introductory paragraph and contains no specific requirements.

Reference Documents

Not Applicable

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.1 Criteria/Assumptions	[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
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The HNP safe shutdown systems used to achieve and maintain a safe and stable condition are consistent with the information provided in the GE Report referenced above.

For each safe shutdown system, plant P&IDs, one-line diagrams and system operating procedures were used to identify the safe shutdown flow paths and operational characteristics that must be established to accomplish the desired safe shutdown functions. Any components whose spurious operation, including flow diversions, could impair safe shutdown system operability were also identified. This analysis has been documented in HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.0.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.2 Criteria/Assumptions	[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 10 CFR 50 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
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Hatch Nuclear Plant complies with the BWROG position detailed in GE Report GE-NE-T43-00002-00-03-R01, which has been accepted by the NRC for the use of SRVs and low pressure systems (LPCI/CS) to achieve and maintain a safe and stable plant condition. HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.2.2 details these systems.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.3 Criteria/Assumptions	[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

Not Applicable

Comments

HNP Units 1 and 2 are GE Type 4 BWRs; PWR guidance is not applicable.

Alignment Statement

Not Required

Alignment Basis

Not Applicable

Reference Documents

Not Applicable

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.4 Criteria/Assumptions	The classification of shutdown capability as alternative/dedicated shutdown is made independent of the selection of systems used for shutdown. Alternative/dedicated 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 operator manual actions to the extent allowed by the regulations and the licensing basis of the plant (see Appendix E), repairs (cold shutdown only), exemptions, deviations, GL 86-10 fire hazards analyses or fire protection design change evaluations permitted by GL 86-10, as appropriate. These may also be used in conjunction with alternative/dedicated shutdown capability. A discussion of time zero for the fire condition, as it relates to operator manual actions and repairs, is contained in Appendix E.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Unlike 10 CFR 50 Appendix R, NFPA 805 makes no distinction for alternative / dedicated shutdown. HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 11.4.1 documents the remote shutdown methodology, which is only utilized for a fire in the Control Complex (Fire Area 0024).

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.5 Criteria/Assumptions	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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.1 explicitly states the assumption listed in Section 3.1.1.5 of NEI 00-01.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.6 Criteria/Assumptions	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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.1 explicitly states the assumption listed in Section 3.1.1.6 of NEI 00-01.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.7 Criteria/Assumptions	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	None

Alignment Statement

Aligns with Intent

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.1 explicitly states the criteria listed in Section 3.1.1.7 of NEI 00-01.

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.3.5.5 describes how the NSCA models offsite power, as well as onsite power from the emergency diesel generators. Offsite power is only credited in fire areas where it can be demonstrated to be free of fire damage.

NFPA 805 does not identify "alternate shutdown" areas, and does not require that cold shutdown be completed within 72 hours. The NSCA demonstrates that the plant can be placed in a safe and stable condition in all fire areas. The potential fire effects on systems and components required to maintain cold shutdown will be addressed in the NPO analysis.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.8 Criteria/Assumptions	Post-fire safe shutdown systems and components are not required to be safety-related.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.1 explicitly states the criteria listed in Section 3.1.1.8 of NEI 00-01.

The HNP NSCA model includes non-safety related plant systems / components.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.9 Criteria/Assumptions	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	None

Alignment Statement

Aligns with Intent

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report documents that the plant can achieve and maintain a safe and stable condition in all fire areas. Per Section 1.3.1 of NFPA 805, given a fire, a plant is not required to transition to cold shutdown with 72 hours, but instead provides reasonable assurance to achieve and maintain the fuel in safe and stable condition. The HNP NFPA 805 NSPC analysis has defined the safe and stable conditions as being able to achieve and maintain hot shutdown until such time as the plant can either transition to cold shutdown, or can safely return to power operation.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.10 Criteria/Assumptions	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 (See Appendix E); automatic initiation of systems selected for safe shutdown is not required but may be included as an option, if the additional cables and equipment are also included in the analysis. Spurious actuation of automatic systems (Safety Injection, Auxiliary Feedwater, High Pressure Coolant Injection, Reactor Core Isolation Cooling, etc.) due to fire damage, however, should be evaluated.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.1 explicitly states the criteria listed in Section 3.1.1.10 of NEI 00-01.

Automatic main system initiation (i.e. ATTS initiation signals) are not credited unless they are specifically modeled and analyzed to be available. However, fire induced automatic initiation signals are evaluated for the possibility of spurious component operation and their subsequent adverse impact on safe shutdown.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.11 Criteria/Assumptions	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
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.1 explicitly states the criteria listed in Section 3.1.1.11 of NEI 00-01.

The potential effects of a fire on Hatch Units 1 and 2 are analyzed simultaneously within the NSCA analysis for each Fire Area. This ensures that if a single fire impacts multiple units, the analyst will be required to understand and resolve the impacts of the failures on each unit caused by the fire.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 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 hot 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

Applicable

Comments

None

Alignment Statement

Not Required

Alignment Basis

This is an introductory paragraph and contains no specific requirements.

Reference Documents

Not Applicable

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

NEI 00-01 Ref	NEI 00-01 Section 3 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. Each licensee should have an operator manual action to either vent the instrument air header or to remove RPS power in their post-fire safe shutdown procedures. The presence of this action precludes the need to perform circuit analysis for the reactivity control function and is an acceptable way to accomplish this function. If this action is a "time critical" action, the timing must be justified.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Reactivity control capabilities required to achieve and maintain a safe and stable plant condition post-fire are identified in HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.2.1.

Reactivity Control is established and maintained by a manual reactor scram, or automatic reactor scram where free of fire impacts, and insertion of control rods. Subcritical conditions are achieved for all reactor operating modes and maintaining subcritical conditions after a reactor scram is a passive function.

Plant procedures provide contingency operator actions in the event a required reactor scram does not occur.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

Procedure 34AB-X43-001-2, Fire Procedure, Ver. 18.1

Procedure 34AB-X43-001-1, Fire Procedure, Ver. 15.1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.1 Reactivity Control	[PWR] Makeup/Charging There must be a method for ensuring that adequate shutdown margin is maintained from initial reactor SCRAM to cold shutdown conditions, by controlling Reactor Coolant System temperature and ensuring borated water is utilized for RCS makeup/charging.

Applicability

Not Applicable

Comments

HNP Units 1 and 2 are GE Type 4 BWRs; PWR guidance is not applicable.

Alignment Statement

Not Required

Alignment Basis

Not Applicable

Reference Documents

Not Applicable

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

NEI 00-01 Ref	NEI 00-01 Section 3 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) Initial pressure control may be provided by the SRVs mechanically cycling at their setpoints (electrically cycling for EMRVs). Mechanically-actuated SRVs require no electrical analysis to perform their overpressure protection function. The SRVs may also be 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 (ADS) is not a required function. Automatic initiation of the ADS may be credited, if available. If automatic ADS is not available and use of ADS is desired, an alternative means of initiation of ADS separate from the automatic initiation logic for accomplishing the pressure control function should be provided.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Reactor Pressure Vessel (RPV) pressure control capabilities required to achieve and maintain a safe and stable plant condition post-fire are identified in HNP Calculation SENH-15-009, Nuclear.Safety Capability Assessment Report, Section 9.2.2.

Overpressure protection is provided by the SRVs in the self-activated spring lift mode. The SRVs are also credited as a means of depressurization so that low pressure injection systems can be used to provided inventory to the reactor vessel. When used to depressurize the RPV to allow for low pressure injection, the SRVs are operated manually from the Main Control Room, or remote shutdown panel.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.2 Pressure Control Systems	[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

Not Applicable

Comments

HNP Units 1 and 2 are GE Type 4 BWRs; PWR guidance is not applicable.

Alignment Statement

Not Required

Alignment Basis

Not Applicable

Reference Documents

Not Applicable

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

NEI 00-01 Ref	NEI 00-01 Section 3 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. Spurious actuation of automatic systems, however, should be evaluated (High Pressure Coolant Injection, High Pressure Core Spray, Reactor Core Isolation Cooling, etc.).

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Reactor Pressure Vessel (RPV) inventory control capabilities required to achieve and maintain a safe and stable plant condition post-fire are identified in HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.2.2.

RPV Inventory is maintained using high pressure systems, High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC), or low pressure coolant injection (LPCI) using RHR or the Core Spray (CS) system. For low pressure injection, the SRVs are credited for pressure reduction.

Makeup water will be provided from the Condensate Storage Tank (CST) and subsequently transitioned to the suppression pool. Adequate CST water capacity is available to maintain RPV level until conditions are met to transfer the RCIC, HPCI, RHR or CS pump suction to the suppression pool. Inventory required for RPV makeup when using low pressure injection is recirculated from the suppression pool.

Automatic system initiation is not credited unless it is specifically modeled and analyzed to be available. Fire induced automatic initiation signals are evaluated for the possibility of spurious component operation and their subsequent adverse impact on safe shutdown.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.3 Inventory Control	[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. Spurious actuation of automatic systems, however, should be evaluated (Safety Injection, High Pressure Injection, Auxiliary Feedwater, Emergency Feedwater, etc.).

Applicability

Not Applicable

Comments

HNP Units 1 and 2 are GE Type 4 BWRs; PWR guidance is not applicable.

Alignment Statement

Not Required

Alignment Basis

Not Applicable

Reference Documents

Not Applicable

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

NEI 00-01 Ref	NEI 00-01 Section 3 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. (This is not a hot shutdown requirement.) This does not restrict the use of other systems.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns with Intent

Alignment Basis

Reactor Pressure Vessel (RPV) decay heat removal capabilities required to achieve and maintain a safe and stable plant condition post-fire are identified in HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.2.3.

Decay heat is removed initially by natural circulation within the reactor pressure vessel and self-activated spring lift mode of the SRVs. Long-term decay heat removal is accomplished using Alternate Shutdown Cooling. This involves using the LPCI or CS pump to circulate water between the reactor and the suppression pool through manually operated SRVs. The Suppression Pool Cooling mode (SPC) and the Residual Heat Removal Service Water system (RHRSW) are used to cool the suppression pool to remove the stored energy from the pool to the ultimate heat sink.

Net Positive Suction Head (NPSH) requirements of safe shutdown systems taking suction from the suppression pool are discussed in HNP Calculation SENH-15-009, Section 6.3.2. NPSH requirements for Unit 2 systems have been dispositioned by calculation and not included in the NSCA model. NPSH requirements for Unit 1 systems have either been dispositioned by calculation or explicitly modeled and evaluated, depending on the system.

As pointed out in Section 1.3.1 of NFPA 805, given a fire, a plant is not required to transition to cold shutdown within 72 hours, but instead provide reasonable assurance to achieve and maintain the fuel in a safe and stable condition. For HNP, the required end state of safe and stable under NFPA 805 will be met when the plant is in a stable hot shutdown configuration.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.4 Decay Heat Removal	[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 is not a hot shutdown requirement.) This does not restrict the use of other systems.

Applicability

Not Applicable

Comments

HNP Units 1 and 2 are GE Type 4 BWRs; PWR guidance is not applicable.

Alignment Statement

Not Required

Alignment Basis

Not Applicable

Reference Documents

Not Applicable

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.5 Process Monitoring	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 (10 CFR 50 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/dedicated shutdown (III.G.3). The use of this same list for III.G.2 redundant Post-Fire Safe Shutdown is acceptable, but the analyst needs to review the specific license basis for the plant under evaluation. 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 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	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Process monitoring instrumentation required to achieve and maintain a safe and stable plant condition post-fire is identified in HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.2.5. The instrumentation required by the NSCA model is consistent with minimum process monitoring instrumentation expectations identified in USNRC Information Notice (IN) 84-09, and as previously approved by the USNRC in the 10 CFR 50 Appendix R licensing basis for HNP.

- Reactor coolant level and pressure: These instruments are modeled in support of the Inventory and Pressure Performance Goals.
- Suppression pool level and temperature: These instruments are modeled in support of the Decay Heat Removal Performance Goal
- Diagnostic instrumentation for safe shutdown systems: These instruments are modeled in support of their appropriate system.
- Level indication for tanks needed for safe shutdown: These instruments are included in the system logics for which the tank is required.
- Isolation condenser level: There is no isolation condenser at HNP.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.5 Process Monitoring	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 (10 CFR 50 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/dedicated shutdown (III.G.3). The use of this same list for III.G.2 redundant Post-Fire Safe Shutdown is acceptable, but the analyst needs to review the specific license basis for the plant under evaluation. In general, process monitoring instruments similar to those listed below are needed to successfully use existing operating procedures (including Abnormal Operating Procedures). 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	Comments
Not Applicable	HNP Units 1 and 2 are GE Type 4 BWRs; PWR guidance is not applicable.

Alignment Statement

Not Required

Alignment Basis

Not Applicable

Reference Documents

Not Applicable

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.6 Support Systems	[Blank Heading - No specific guidance]

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This is an introductory header and contains no specific requirements.

Reference Documents

Not Applicable

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

NEI 00-01 Ref	NEI 00-01 Section 3 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 any beneficial effects of 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 may 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 125VDC distribution system can be powered from sources feed 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	None

Alignment Statement

Aligns

Alignment Basis

Electrical distribution systems required to achieve and maintain a safe and stable plant condition post-fire are identifed in HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.2.4.

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.3.5.5 documents that offsite power has been analyzed and is credited where available. If offsite power is not available, at least one emergency diesel generator is credited to supply the required electrical power.

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.3.5.5 documents that there is a 250VDC Distribution system that supplies motive DC

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

power to safe shutdown systems. The 125 VDC Distribution system supplies control DC power to safe shutdown systems. Also, the 120VAC system provides instrument AC power to safe shutdown systems.

HNP Calculation SENH-15-009, Section 9.3.5.5.4, identifies that the station batteries can supply any required control power during the interim time period required for the diesel generators to become operational.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.6.2 Cooling Systems	Various cooling water systems are 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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Cooling water systems required to achieve and maintain a safe and stable plant condition post-fire are identified in HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Sections 9.3.5.1 and 9.3.5.2.

The HNP NSCA model requires the Residual Heat Removal Service Water system (RHRSW) and Plant Service Water system (PSW) to provide the vital support function of cooling water for the other mechanical systems / functions / components of the NSCA model.

The RHRSW system is credited to provide cooling water to the RHR heat exchanger for the Decay Heat Removal Performance Goal. The PSW system is credited to provide cooling water to the following:
- Emergency Diesel Generators
- Seal Cooling for RHR Pumps
- HVAC Systems

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.6.3 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, however, are not required to support post-fire safe shutdown in all cases. The need for HVAC system operation is based on plant specific configurations and plant specific heat loads. Typical potential 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 could be required or useful in supporting post-fire safe shutdown. Transient temperature response analyses are often utilized to demonstrate that specific HVAC systems would or would not be required. If HVAC systems are credited, the potential for adverse fire effects to the HVAC system must also be considered, including: - Dampers closing due to direct fire exposure or due to hot gases flowing through ventilation ducts from the fire area to an area not directly affected by the fire. Where provided, smoke dampers should consider similar effects from smoke. - Recirculation or migration of toxic conditions (e.g., smoke from the fire, suppressants such as Carbon Dioxide). In certain situations, adequate time exists to open doors to provide adequate cooling to allow continued equipment operation. Therefore, the list of required safe shutdown components as it relates to HVAC Systems may be determined based on transient temperature analysis. Should this analysis demonstrate that adequate time exists to open doors to provide the necessary cooling, this is an acceptable approach to achieving HVAC Cooling. The temperature analysis must demonstrate the adequacy of the cooling effect from opening the door within the specified time. Only those components whose operation is required to provide HVAC Cooling for required safe shutdown components in a time frame that cannot be justified for operator manual actions are considered themselves to be required safe shutdown components. This latter set of HVAC Cooling Components are required to meet the criteria for required safe shutdown components with regard to the available mitigating tools.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Heating Ventilation and Air Conditioning systems (HVAC) required to maintain a safe and stable plant condition post-fire are identifed in HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report. Section 6.2.4 identifies HVAC systems that are not included in the NSCA model. Section 6.3.5 describes systems included in the NSCA model.

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

The HNP NSCA model requires HVAC to provide the vital support function of air cooling for plant equipment operability and to maintain Main Control Room habitability for plant operation personnel. HVAC systems have been evaluated for the following:

- Main Control Room
- Diesel Generator Room(s)
- 4kV Switchgear Room(s)
- Intake Structure
- RHR/CS Pump Rooms
- RCIC Pump Room
- HPCI Pump Room
- Control Building
- Alternate Shutdown Panels

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 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:

Applicability

Applicable

Comments

None

Alignment Statement

Not Required

Alignment Basis

This is an introductory section and contains no specific guidance.

Reference Documents

Not Applicable

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

NEI 00-01 Ref		NEI 00-01 Section 3 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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 3.0 identifies the following types of documents and databases as sources of design input utilized for the development of the NSCA model:
- HNP FSAR
- Piping and Instrumentation Diagrams (P&IDs)
- Single Line Diagrams
- Electrical Three-line Diagrams
- Schematic and Wiring Diagrams
- Instrument Loop Diagrams
- System Functional Descriptions
- Normal, Emergency and Abnormal Operating Procedures
- GE Calculations and Reports

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 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 other equipment whose mal-operation or spurious operation could impact the required safe shutdown function. The components in this latter set are classified as either required for hot shutdown or as important to SSD as explained in Appendix H.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The available combinations of safe shutdown systems required to meet the safe shutdown functions have been identified and depicted in the CAFTA fault tree developed for use with the ARCPlus safe shutdown compliance software. Components required to achieve and maintain the nuclear safety functions and components whose fire-induced failure could prevent the operation or result in the mal-operation of those components needed to meet the nuclear safety criteria are included. These systems have been documented in HNP Calculation SENH-15-009, Section 6.3.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.1.3.3	Define Combination 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, paths may be defined on a divisional basis since the availability of electrical power and other support systems must be demonstrated for each path. During the equipment selection phase, identify any additional support systems and list them for the appropriate path.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.3 identifies the overall process utilized to model the combinations of plant systems that satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.1 also identifies the overall process utilized to logically relate individual systems in support of each performance goal. Success paths for each performance goal are specified by the CAFTA fault tree logic, which represents the minimum system combinations required to achieve a specific performance goal.

Certain support systems / functions, such as electrical power and cooling water, are modeled to directly support specific components and systems rather than to directly support performance goals. These relationships are illustrated by the CAFTA fault tree logic in HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section . 9.3.5.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 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 for an example of a table illustrating how to document the various combinations of systems for selected shutdown paths.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns with Intent

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.3 identifies the overall process utilized to model the combinations of plant systems that satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.2 also identifies that each performance goal may have multiple success paths representing a different combination of systems / functions.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 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 functions. 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 mal-operation of the safe shutdown systems. For each path it will be important to understand which components are classified as required safe shutdown components and which are classified as important to safe shutdow components. When evaluating the fire-induced impact to each affected cable/component in each fire area, this classification dictates the tools available for mitigation the affects.

Applicability

Applicable

Comments

None

Alignment Statement

Not Required

Alignment Basis

This is an introductory paragraph and contains no specific guidance or requirements.

Reference Documents

Not Applicable

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

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.2.1	Criteria/Assumptions	Consider the following criteria and assumptions when identifying equipment necessary to perform the required safe shutdown functions:

Applicability

Applicable

Comments

None

Alignment Statement

Not Required

Alignment Basis

This is an introductory paragraph and contains no specific requirements.

Reference Documents

Not Applicable

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

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.2.1.1	Criteria/Assumptions	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.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns with Intent

Alignment Basis

There is no explicit distinction made in the HNP NSCA between primary and secondary equipment; however, a smiliar approach is maintained through the system-to-equipment logic success paths and the component-to-component logic success paths in the CAFTA NSCA fault tree model. Secondary components such as those described above are included with the primary component, and any required or associated circuits are also assigned to the primary component.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.2.1.2 Criteria/Assumptions	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. For example, post-fire coefficients of friction for rising stem valves cannot be readily determined. Handwheel sizes and rim pulls are based on well lubricated stems. Any post-fire operation of a rising stem valve should be well justified using an engineering evaluation.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.2 explicitly states the assumption listed in Section 3.2.1.2 of NEI 00-01. Fire damage to manual valves and piping will not prevent the component from maintaining its related system's pressure boundary. Mechanical components susceptible to fire damage (brazed or soldered instrument lines, instrument tubing for credited instruments, etc.) are identified and evaluated on a fire area basis in the form of a standalone evaluation. NFPA 805 credited recovery actions, including those that require manual operation of valves are subject to an assessment of feasibility with due consideration to the location and effects of the fire.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

Calculation SENH-16-007, Recovery Action Feasibility Assessment, Ver. 1

Calculation SENH-15-008, Instrument Sense Line Analysis for Appendix R & NSCA SSD Models, Ver. 2

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.2.1.3 Criteria/Assumptions	Assume that all components, including manual valves, are in their normal position as shown on P&IDs or in the plant operating procedures, that there are no LCOs in effect, that the Unit is operating at 100% power and that no equipment has been taken out of service for maintenance.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.2 explicitly states the assumption listed in Section 3.2.1.3 of NEI 00-01. The normal operating position of a component is considered to be the position of the component during normal reactor power operation. This is based on system P&IDs and equipment lineups per HNP procedures.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.2.1.4 Criteria/Assumptions	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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.2 explicitly states the assumption listed in Section 3.2.1.4 of NEI 00-01. The fire is not assumed to affect the ability of a check valve to perform its boundary isolation function.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.2.1.5 Criteria/Assumptions	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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.2 explicitly states the assumption listed in Section 3.2.1.5 of NEI 00-01. Impact to instrument cables is assumed to fail the instrument in the least desirable state (i.e. the instrument could fail high, low, or in some intermediate condition, whichever is worse). In addition, the analysis includes instruments which provide permissive or controlling signals that can cause spurious operation. These instruments are modeled in direct support of the affected component using component logics, cable logics, or a combination of both as identified in SENH-15-009, Section 6.2.5.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 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:

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This is an introductory paragraph and contains no specific requirements.

Reference Documents

Not Applicable

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

NEI 00-01 Ref		NEI 00-01 Section 3 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. When developing the SSEL, 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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.3 identifies the overall process utilized to model the combination of plant components for each plant system that is identified as being required to satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

A review of P&IDs, electrical drawings, instrument loop diagrams, etc. was performed to identify the NSCA systems, and to identify and develop the NSCA logics. Secondary components are included with the primary component that would be affected by fire damage to the secondary component.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 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. Additionally, refer to Section 4 for the Resolution Methodology for determining the Plant Specific List of MSOs requiring evaluation. Criteria for making the determination as to which of these components are to be classified as required for hot shutdown or as important to SSD is contained in Appendix H. If additional systems are identified which are necessary for the operation of the safe shutdown system under review, include these as required for hot shutdown systems. 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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.3 identifies the overall process utilized to model the combinations of plant components, including MSOs, for each plant system that is identified as being required to satisfy each of the Nuclear Safety Performance Criteria from Section 1.5.1 of NFPA 805.

A review of P&IDs, electrical drawings, instrument loop diagrams, etc. was performed to identify the NSCA systems, and to identify and develop the NSCA logic.

- Mechanical and electrical system components such as pumps, valves, fans, circuit breakers, transformers, diesel generators, motor control centers, batteries, battery chargers, distribution panels, instrumentation, dampers, etc. which have an active function in achieving safe shutdown are included in the NSCA.

- Mechanical and electrical system passive components such as pumps, valves, fans, circuit breaker, instrumentation and dampers, etc. are included in the NSCA if they maintain a system boundary or if the spurious operation(s) of the passive component(s) has an adverse impact on NSCA capabilities.

- Mechanical system passive components such as tanks and heat exchangers which have no spurious failure mode are included in the NSCA for completeness.

- Control panels and discrete electrical components such as relays, starters, fuses, and other nonelectrically operated circuit breakers, instrument power supplies, etc. are not explicitly identified or included in the NFPA 805 NSPC Equipment List. These secondary, or subcomponents, are represented in the NSCA by virtue of the circuit analysis and cables that interconnect them to the primary component.

- Manual valves that are repositioned for credited NFPA 805 Recovery Actions are in the NFPA 805 NSPC Equipment List.

- Components whose spurious actuation could adversely affect the system's ability to perform its safe shutdown function was also included in the NFPA 805 NSPC Equipment List.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 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. Criteria for making the determination as to which of these components are to be classified as required for hot shutdown or as important to SSD is contained in Appendix H. 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 post-fire safe shutdown and it documents various equipment-related attributes used in the analysis. 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	None

Alignment Statement

Aligns with Intent

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.3 identifies the overally process utilized to model the combinations of plant components for each plant system that is identified as being required to satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

A Safe Shutdown Equipment List (SSEL) has been developed, and the required information to support a safe shutdown analysis using the ARCPlus software has been included. The SSEL is found in HNP caluculation SENH-15-009, Attachment A. Specific safe shutdown "paths" are not identified; rather, each components is tied to at least one basic event in the safe shutdown fault tree. The fault tree depicts the various components and system interrelationships that must be met to achieve safe shutdown.

Instrument sense lines are discussed in HNP Calculation SENH-15-009, Section 6.1.7. This section documents the instrument tubing sensing line information for HNP, which supports the achievement of the NSPC of NFPA 805 with respect to instrument tubing sensing lines per the requirements/guidance of NFPA 805, 2001 Edition, Section B.2.1.(e) and NEI 00-01, Revision 2, Sections 3.2.1.2, 3.2.2.3, and 3.4.1.9. The HNP instrument sense line analysis is documented within Calculation SENH-15-008.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

SENH-15-008, Instrument Sense Line Analysis for Appendix R & NSCA SSD Models, Ver. 2

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

NEI 00-01 Ref		NEI 00-01 Section 3 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. Criteria for making the determination as to which of these components are to be classified as required for hot shutdown or as important to SSD is contained in Appendix H.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.3 identifies the overall process utilized to model the combinations of plant components for each plant system that is identified as being required to satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

The NFPA 805 NSPC Equipment List is designed for use with ARCPlus. The Equipment List contains SSE ID, the Basic Event, the Component Type, the Normal Position(s), the Required Position(s), the Failure Position(s) and Location Information (Fire Area, etc.). HNP Calculation SENH-15-009, Attachment A, documents this equipment list.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

NEI 00-01 Ref		NEI 00-01 Section 3 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

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.3 identifies the overall process utilized to model the combinations of plant components for each plant system that is identified as being required to satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

Interlocks and supporting components have been identified and are modeled in the CAFTA NSCA fault tree model. This includes a component or system's need for electrical power, HVAC and/or cooling water. These relationships are maintained by the CAFTA NSCA fault tree model.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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 Section 3 Guidance
3.3	Safe Shutdown Cable Selection and Location	This section provides industry guidance on one acceptable approach 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 mal-operation 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. The classification of a cable as either an important to SSD circuit cable or a required safe shutdown cable is also derived from the classification applied to the component that it supports. This classification can vary from one fire area to another depending on the approach used to accomplish post-fire safe shutdown in the area. Refer to Appendix H for the criteria to be used for classifying required and important to SSD components.

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This is introductory information and contains no specific guidance.

Reference Documents

Not Applicable

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 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 components, systems and functions. Consider the following criteria when selecting cables that impact safe shutdown equipment:

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This is introductory information and contains no specific guidance.

Reference Documents

Not Applicable

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.1.1 Criteria/Assumptions	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. Review 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.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.3 explicitly states the criteria listed in Section 3.3.1.1.1 of NEI 00-01.

All cables including those from interlocks, instruments, and power supplies that could potentially adversely impact the desired operation of safe shutown equipment are listed. This includes cables external to the component control circuit if any cable fault could adversely impact the required state of the component unless the cable (s) are included with another SSE. See PI-02-001, Section 7.1 and 8.2.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.1.2 Criteria/Assumptions	In cases where the failure (including spurious operations) of a single cable could impact more than one piece of safe shutdown equipment, associate the cable with each piece of safe shutdown equipment.

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.3 explicitly states the criteria listed in Section 3.3.1.1.2 of NEI 00-01.

All cables, including those from interlocks, instruments, and power supplies, that could potentially adversely impact the desired operation of a safe shutdown equipment are listed. Circuit analysis is done on a component level; where a cable may affect several SSEs, that cable is identified as required for all applicable SSEs. Cable may be required directly or via component to component relationships. See PI-02-001 Sections 7 and 8.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.1.2.1 Criteria/Assumptions	Electrical devices such as relays, switches and signal resistor units are considered to be acceptable isolation devices. In the case of instrument loops and electrical metering circuits, 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. Refer to Section 3.5 for the types of faults that should be considered when evaluating the acceptability of the isolation device being credited.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.3 explicitly states the criteria listed in Section 3.3.1.1.2.1 of NEI 00-01.

Circuit isolation devices include relays, switches, isolators, and other isolation devices as noted in PI-02-001. The use of switches and relays as isolation devices is done cognizant of what may cause them to change state. Cables separated from the circuit of concern by suitable isolation devices are not required.

In some cases during circuit analysis, normally-open relay or switch contacts have been used to form the "boundary" of a circuit. When this approach was used, the analyst considered whether the switch or relay could change state as part of the anticipated shutdown process and sequence of events.

Instrument Loops:
- For cases in which the output of an instrument module (signal comparator, summing amplifier, etc.) is not required to support the function being analyzed, but the signal of interest supplies the module along with one or more other signals, input-to-input isolation between the two signals has been assumed.

- For cases in which the output of an instrument module (signal comparator, summing amplifier, etc.) is required to support the function being analyzed, all inputs to the module have been addressed in the circuit analysis, since any false input signal could affect the output signal.

- The following instrument loop modules were considered to provide adequate isolation:
  - Instrument loop isolators
  - Outputs from isolating power supplies
  - Signal Comparator
  - Bi-stable contacts (dry contacts)
  - Summing amplifiers
  - Controller units

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1
Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.1.3 Criteria/Assumptions	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. To be properly screened out, however, the circuits associated with these devices 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. Refer to Section 3.5 for the types of faults that should be considered when evaluating the acceptability of the isolation device being credited.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.3 explicitly states the criteria listed in Section 3.3.1.1.3 of NEI 00-01.

Cables listed on the main scheme drawing are screened out as not required if they will not affect the safe shutdown function of the component. Cables can be screened out because they are isolated, because a fault will not cause undesired affects on the circuit, or because they are part of an associated circuit such as a motor space heaters, testing, annunciator, and/or computer inputs.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.1.4 Criteria/Assumptions	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. The set of cables described above are classified as required safe shutdown cables. Evaluate the power cables for breaker coordination concerns. The non-safe shutdown cables off of the safe shutdown buses are classified as required for hot shutdown or as important to SSD based on the criteria contained in Appendix H.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.3 explicitly states the criteria listed in Section 3.3.1.1.4 of NEI 00-01.

Power Supply Incorporation:
Power supply cable selection typically ends at the closest electrical isolation device for the component identified in the SSEL. For instance, power supply cables to an MCC will not be listed for a motor operated valve, only the power supply cable from the MCC to the valve will be listed for the valve. The MCC would be identified as a safe shutdown component in the SSEL and a separate circuit analysis performed for the MCC. Required power supplies are verified to ensure that they are on the SSEL, if not, they are added. If a power supply is required to be added to SSEL then a circuit analysis for the new power supply is required.

Electrical Coordination:
Breaker coordination is ensured by reviewing the time current curves from the plant's coordination study to ensure coordination. Coordination assures that the protective device nearest the fault operates prior to operation of upstream devices. The means of assuring circuit protection and coordination is provided in a series of calculations. These calculations demonstrate that the Class 1E and non-Class 1E power supplies credited for safe shutdown compliance do have adequate coordination.

All modifications are reviewed to ensure that the existing satisfactory circuit coordination is not compromised by future design changes.

Per LAR Table S-2, Item #6 and #7, coordination concerns will be addressed by modifications.

Multiple High Impedance Faults:
NEI 00-01 Appendix B.1 identifies Multiple High Impedance Faults (MHIF) as a unique common power supply concern and gives a resolution framework. Per LAR Table S-3, IMP-9, MHIF analysis will be documented.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1
Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Calculation SENH-88-011, Coordination Study - Cooling TWR MTRS & SWGR, Rev. 0
Calculation SENH-89-005, Coordination Study for 600V And 208V Switchgear Busses, Ver. 11
Calculation SENH-89-008, Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-010, Hatch U1 Micro Versa Trip Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-026, Coordination Study for 600V, 480V, & 120V Switchgears, Ver. 11
Calculation SENH-90-024, 125/250VDC Switchgear 1A & B / Breaker Coordination Utilizing EC Trip Devices, Ver. 6
Calculation SENH-91-011, Coordination Study for 250 VDC Switch Gear 2R22-S016 & S017, Ver. 7
Calculation SENH-94-013, Coordination Study for Non-Appendix R Breakers and Fuses in Response to REA HT-93753., Ver. 4
Calculation SETH-85-082, Appendix R Protection Device Coordination Study of 600/208/120V AC Circuits, Ver. 20
Calculation SETH-85-196, Appendix R Protective Device Coordination Study of 250/125V DC Circuits, Ver. 6
Calculation SENH-15-005, Protection Coordination Study for Alternate 4 KV Power Supply to Service Building, Ver. 1

Implementation Items

IMP-9 Update electrical coordination calculations to document the current analysis of alignment with NEI 00-01 Appendix B.1 regarding Multiple High Impedance Faults (MHIF).

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.1.4.1 Criteria/Assumptions	The automatic initiation logics for the credited post-fire safe shutdown systems are generally not required to support safe shutdown. Typically, each system can be controlled manually by operator actuation in the main control room or emergency control station. The emergency control station includes those plant locations where control devices, such as switches, are installed for the purpose of operating the equipment. If operator actions to manually manipulate equipment at locations outside the MCR or the emergency control station are necessary, those actions must conform to the regulatory requirements on operator manual actions (See Appendix E). If not protected from the effects of fire, the fire-induced failure of automatic initiation logic circuits should be considered for their potential to adversely affect any post-fire safe shutdown system function.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.3 explicitly states the criteria listed in Section 3.3.1.1.4.1 of NEI 00-01.

Instruments which do not provide a credited control function, but whose spurious operation could adversely affect safe shutdown are considered to be required safe shutdown components. Examples include instrumentation involved in the initiation of the ESFAS automatic control logics. The population of cables that are involved with the automatic initiation logics are identified through circuit analysis, and the instrumentation is depicted on the Safe Shutdown fault trees. Any manual action to recover components due to actuation signals are evaluated for feasibility.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.1.5 Criteria/Assumptions	Cabling for the electrical distribution system is a concern for those breakers that feed 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 and classified as a required safe shutdown cable. 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. Additionally, the non-safe shutdown circuits off of each of the required safe shutdown components in the electrical distribution system can impact safe shutdown if not properly coordinated. These cables are classified as required for hot shutdown based on the criteria contained in Appendix H.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.3 explicitly states the criteria listed listed in Section 3.3.1.1.5 of NEI 00-01.

Breaker coordination is ensured by reviewing the time current curves from the plant's coordination study to ensure coordination. Coordination assures that the protective device nearest the fault operates prior to operation of upstream devices. The means of assuring circuit protection and coordination is provided in a series of calculations. These calculations demonstrate that the Class 1E and non-Class 1E power supplies credited for safe shutdown compliance do have adequate coordination.

All modifications are reviewed to ensure that the existing satisfactory circuit coordination is not compromised by future design changes.

Per LAR Table S-2, Item #6 and #7, coordination concerns will be addressed by modifications.

Multiple High Impedance Faults:
NEI 00-01 Appendix B.1 identifies Multiple High Impedance Faults (MHIF) as a unique common power supply concern and gives a resolution framework. Per LAR Table S-3, IMP-9, MHIF analysis will be documented.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1
Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1
Calculation SENH-15-005, Protection Coordination Study for Alternate 4 KV Power Supply to Service Building, Ver. 1
Calculation SENH-88-011, Coordination Study - Cooling TWR MTRS & SWGR, Rev. 0
Calculation SENH-89-005, Coordination Study for 600V And 208V Switchgear Busses, Ver. 11

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Calculation SENH-89-008, Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-010, Hatch U1 Micro Versa Trip Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-026, Coordination Study for 600V, 480V, & 120V Switchgears, Ver. 11
Calculation SENH-90-024, 125/250VDC Switchgear 1A & B / Breaker Coordination Utilizing EC Trip Devices, Ver. 6
Calculation SENH-91-011, Coordination Study for 250 VDC Switch Gear 2R22-S016 & S017, Ver. 7
Calculation SENH-94-013, Coordination Study for Non-Appendix R Breakers and Fuses in Response to REA HT-93753., Ver. 4
Calculation SETH-85-082, Appendix R Protection Device Coordination Study of 600/208/120V AC Circuits, Ver. 20
Calculation SETH-85-196, Appendix R Protective Device Coordination Study of 250/125V DC Circuits, Ver. 6

Implementation Items

IMP-9 Update electrical coordination calculations to document the current analysis of alignment with NEI 00-01 Appendix B.1 regarding Multiple High Impedance Faults (MHIF).

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.1.6 Criteria/Assumptions	Exclusion analysis may be used to demonstrate a lack of potential for any impacts to post-fire safe shutdown from a component or group of components regardless of the cable routing. For these cases, rigorous cable searching and cable to component associations may not be required.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.3 explicitly states the criteria listed in Section 3.3.1.1.6 of NEI 00-01.

Exclusion analysis was not utilized in the development of the NSCA model and component selection activities.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.3.2	Associated Circuit of Concern Cables	Appendix R, through the guidance provided in NRC Generic Letter 81-12, requires that separation features be provided for associated non-safety circuits that could prevent operation or cause mal-operation 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 7.1.5 and further clarified in a NRC memorandum dated March 22, 1982 from R. Mattson to D. Eisenhut, Reference 7.1.6. They are as follows: - Spurious actuations (4) - Common power source - Common enclosure. Each of these cables is classified as an associated circuit of concern cable. (4) As explained in NRC RIS 2005-30 and in Appendix H, components whose spurious operations could directly prevent the required safe shutdown path in any fire area from performing its required hot shutdown function are classified as required for hot shutdown components. Components whose spurious operation could affect important to safe shutdown components might be associated circuits of concern for spurious actuation.

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This is introductory information and contains no specific guidance.

Reference Documents

Not Applicable

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.3.2	Associated Circuit of Concern Cables - Cable Whose Failure May Cause Spurious Actuations	Cables Whose Failure May Cause Spurious Operations Safe shutdown system spurious operation concerns can result from fire damage to a cable whose failure could cause the spurious operation/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	None

Alignment Statement

Aligns

Alignment Basis

PI-02-001 Rev. 1 describes the circuit identification and analysis for each NSCA component to ensure that the identified functional requirements are met. These functional requirements include: (1) normal position (2) required position (3) failed position.

Cables are identified by the circuit analysis as being required if failure alone (or their failure in combination with another cable) could adversely affect the desired position(s) / function(s) for the NSCA component. This analysis is performed based on consideration of the effects of open circuits, short circuits, and/or grounds; as well as interlocked circuits.

Multiple simultaneous circuit failures are postulated in the circuit identification and analysis (affecting multiple cables, and affecting multiple conductors within cables). No limit is prescribed to the number or type circuit failures that are postulated to occur except as modified by the following:

- Spurious operation, when resulting only from properly sequenced three-phase to three-phase external hot shorts is only postulated in the circuit identification and analysis for high/low pressure interface valves.

- Spurious operation, when only resulting from positive to positive (+ to +) and negative to negative (- to -) external DC hot shorts in ungrounded DC circuits is only postulated in the circuit identification and analysis for high/low pressure interface valves.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.3.2	Associated Circuit of Concern Cables - Common Power Source Cable	Common Power Source Cables The concern for the common power source associated circuits of concern 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	None

Alignment Statement

Aligns

Alignment Basis

PI-02-001 Rev. 1 states that the circuit analysis was performed under the assumption that circuit overcurrent protective devices (e.g. relays, circuit breakers, fuses) are assumed to operate in accordance with their designed time-current tripping characteristics.

Breaker coordination is ensured by reviewing the time current curves from the plant's coordination study to ensure coordination. Coordination assures that the protective device nearest the fault operates prior to operation of upstream devices. The means of assuring circuit protection and coordination is provided in a series of calculations. These calculations demonstrate that the Class 1E and non-Class 1E power supplies credited for safe shutdown compliance do have adequate coordination.

All modifications are reviewed to ensure that the existing satisfactory circuit coordination is not compromised by future design changes.

Per LAR Table S-2, Item #6 and #7, coordination concerns will be addressed by modifications.

Multiple High Impedance Faults:
NEI 00-01 Appendix B.1 identifies Multiple High Impedance Faults (MHIF) as a unique common power supply concern and gives a resolution framework. Per LAR Table S-3, IMP-9, MHIF analysis will be documented.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1
Calculation SENH-15-005, Protection Coordination Study for Alternate 4 KV Power Supply to Service Building, Ver. 1
Calculation SENH-88-011, Coordination Study - Cooling TWR MTRS & SWGR, Rev. 0
Calculation SENH-89-005, Coordination Study for 600V And 208V Switchgear Busses, Ver. 11
Calculation SENH-89-008, Overcurrent Coordination Study, Ver. 7

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Calculation SENH-89-010, Hatch U1 Micro Versa Trip Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-026, Coordination Study for 600V, 480V, & 120V Switchgears, Ver. 11
Calculation SENH-90-024, 125/250VDC Switchgear 1A & B / Breaker Coordination Utilizing EC Trip Devices, Ver. 6
Calculation SENH-91-011, Coordination Study for 250 VDC Switch Gear 2R22-S016 & S017, Ver. 7
Calculation SENH-94-013, Coordination Study for Non-Appendix R Breakers and Fuses in Response to REA HT-93753., Ver. 4
Calculation SETH-85-082, Appendix R Protection Device Coordination Study of 600/208/120V AC Circuits, Ver. 20
Calculation SETH-85-196, Appendix R Protective Device Coordination Study of 250/125V DC Circuits, Ver. 6

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.3.2	Associated Circuit of Concern Cables - Common Enclosure Cables	Common Enclosure Cables The concern with common enclosure associated circuits of concern 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	None

Alignment Statement

Aligns

Alignment Basis

PI-02-001 Rev. 1 states that the circuit analysis was performed under the assumption that circuit overcurrent protective devices (e.g. relays, circuit breakers, fuses) are assumed to operate in accordance with their design time-current tripping characteristics.

The electrical circuit design for HNP provides proper circuit protection in the form of circuit breakers, fuses and other devices that are designed to isolate cable faults before the cable ignition temperature is reached. Adequate electrical circuit protection and cable sizing were included as part of the original plant electrical design and are maintained as part of the design change process. Fire rated barrier and penetration seal designs used at HNP preclude the propagation of fire from one fire area to the next to alleviate fire propagation concerns.

DC Control Power is included in the analysis when it is required for proper breaker operation and fault clearing capabilities.

Per LAR Table S-3, IMP-10, Current Transformers that are credible secondary fire concerns will be identified and resolved.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1
Calculation SENH-16-001, Current Transformer Analysis Report, Ver. 1
Calculation SENH-15-005, Protection Coordination Study for Alternate 4 KV Power Supply to Service Building, Ver. 1
Calculation SENH-88-011, Coordination Study - Cooling TWR MTRS & SWGR, Rev. 0
Calculation SENH-89-005, Coordination Study for 600V And 208V Switchgear Busses, Ver. 11
Calculation SENH-89-008, Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-010, Hatch U1 Micro Versa Trip Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-026, Coordination Study for 600V, 480V, & 120V Switchgears, Ver. 11
Calculation SENH-90-024, 125/250VDC Switchgear 1A & B / Breaker Coordination Utilizing EC Trip Devices, Ver. 6

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Calculation SENH-91-011, Coordination Study for 250 VDC Switch Gear 2R22-S016 & S017, Ver. 7
Calculation SENH-94-013, Coordination Study for Non-Appendix R Breakers and Fuses in Response to REA HT-93753., Ver. 4
Calculation SETH-85-082, Appendix R Protection Device Coordination Study of 600/208/120V AC Circuits, Ver. 20
Calculation SETH-85-196, Appendix R Protective Device Coordination Study of 250/125V DC Circuits, Ver. 6

Implementation Items

IMP-9 Update electrical coordination calculations to document the current analysis of alignment with NEI 00-01 Appendix B.1 regarding Multiple High Impedance Faults (MHIF).

IMP-10 Current Transformer analysis (SENH-16-001) will be updated to provide additional evaluation based on the latest industry guidance provided in NUREG/CR-7150, Vol. 3 dated November 2017. Should a credible CT secondary fire scenario be identified, additional analysis or modifications will be completed prior to implementation of NFPA 805 to ensure no detrimental effect on the NSCA, ΔCDF or ΔLERF.

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 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:

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This is introductory information and contains no specific guidance.

Reference Documents

Not Applicable

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.3.3.1	Identify Circuits Necessary 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	None

Alignment Statement

Aligns

Alignment Basis

All cables that support or could adversely affect the ability to achieve and maintain post fire safe shutdown during all plant operating modes were identified using the methodology defined in PI-02-001 Rev. 1. During the cable selection process, power supplies were identified and added to the Equipment List.

The required drawings to perform and verify the cable selection and circuit analysis include Elementary Diagrams, Block Diagrams, One-Line Diagrams, Control Wiring Diagrams, and others were reviewed as required.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 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 in an undesirable manner that impacts post-fire safe shutdown.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

PI-02-001 Rev. 1, Setions 7.1, 7.2 and 8.2, describe the circuit identification and analysis for each NSCA component to ensure that the identified functional requirements are met. These functional requirements include: (1) normal position (2) required position (3) failed position.

Cables are identified by the circuit analysis as being required if failure alone (or their failure in combination with another cable) could adversely affect the desired position(s) / function(s) for the NSCA component. This analysis is performed based on consideration of the effects of open circuits, short circuits, and/or grounds; as well as interlocked circuits. Cables, power supplies and additional equipment controlling interlocks which can adversely affect the NSCA component are identified as required, and either assigned directly to the NSCA component as cables or required through component to component relationships and added to the SSEL.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 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 mal-operation of each piece of safe shutdown equipment. Cables are classified as either required for hot shutdown or important to SSD based on the classification of the component to which they are associated and the function of that component in supporting post-fire safe shutdown in each particular fire area. Refer to Appendix H for additional guidance. 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. Criteria for making the determination as to which cables are to be classified as required for hot shutdown or as important to SSD is contained in Appendix H.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

As identified in SENH-15-009, Nuclear Safety Capability Assessment Report, Section 10.0, all cables that support or could adversely affect the ability to achieve and maintain post fire safe shutdown have been identified using the methodology defined by PI-02-001 Rev. 1, Section 8.2. This methodology identifies the steps of performing a circuit analysis and entering the required fields of the analysis into FDM.

Fire Data Manager (FDM) is a relational database that lists the cables potentially affecting each piece of equipment. Along with cable power supplies, FDM also includes associated drawing numbers, their revision and any interlocks that are investigated to determine their impact on the operation of the equipment.

Plant Data Management System (PDMS) is a relational database that maintains the equipment, cable and route location information.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 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 required for hot shutdown and important to 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. Additionally, when assessing fire-induced damage to circuits that could potentially result in MSOs, the circuit failure criteria in Appendix B should be used. 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	None

Alignment Statement

Aligns

Alignment Basis

The following general circuit failure modes and their effect on circuit behavior/component response have been considered in the circuit analysis evaluation

Cable Failure Modes:
- Shorts-to-Ground
- Hot Shorts
- Open Circuits
- Line-to-Line Faults (3 phase)

Circuit/ Component Effects:
- Spurious Operation
- Loss of Power
- Loss of Control
- Erroneous or Failed Indication

Manual initiation was credited for this analysis. Automatic operation of specific components was credited where appropriate (such as minimum flow valves). In general, automatic initiation (i.e., SIAS initiation signals) was not credited in this analysis. However, fire induced automatic initiation signals were evaluated for the possibility of spurious component operation and their subsequent adverse impact on safe shutdown.

The required cable selection for spurious operation components identify the minimum population of cables that could cause the component to spuriously operate. This criterion conservatively assumes other cables of the appropriate polarity and potential are routed in the same raceway with the selected cable(s).

For multiple conductor cables, all potential fault consequences due to any combination of hot shorts (inter or intra), shorts to ground, or open circuits were considered

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.5.1	Criteria/Assumptions	Apply the following criteria/assumptions when performing fire-induced circuit failure evaluations. Refer to the assessment of the NEI/EPRI and CAROLFIRE Cable Test Results in Appendix B to this document for the basis for these criteria and for further elaboration on the application of the criteria.

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This is introductory guidance information and contains no specific requirements.

Reference Documents

Not Applicable

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.1.1 Circuit Failure Criteria	Circuit Failure Criteria: The criteria provided below addresses the effects of multiple fire-induced circuit failures impacting circuits for components classified as either "required for hot shutdown" or "important to safe shutdown". Consider the following circuit failure types on each conductor of each unprotected cable. Criteria differences, however, do apply depending on whether the component is classified as required for hot shutdown or important to safe shutdown. - 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. – A hot short in the control circuitry for an MOV can bypass the MOV protective devices, i.e. torque and limit switches. This is the condition described in NRC Information Notice 92-18. In this condition, the potential exists to damage the MOV motor and/or valve. Damage to the MOV could result in an inability to operate the MOV either remotely, using separate controls with separate control power, or manually using the MOV hand wheel. This condition could be a concern in two instances: (1) For fires requiring Control Room evacuation and remote operation from the Remote Shutdown Panel, the Auxiliary Control Panel or Auxiliary Shutdown Panel; (2) For fires where the selected means of addressing the effects of fire induced damage is the use of an operator manual action. In each case, analysis must be performed to demonstrate that the MOV can be subsequently operated electrically or manually, as required by the safe shutdown analysis. - 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: Open circuits as a result of conductor melting have not occurred in any of the recent cable fire testing and they are not considered to be a viable form of cable failure.] - 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. A short-to-ground may also result in a change of state for normally energized equipment.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Hot shorts, open circuits and shorts-to-ground are all considered for both intra-cable sources and inter-cable sources. See alignment basis for previous NEI 00-01, Section 3.5.

The potential exists for hot shorts to damage MOV motors and/or valves as a result of bypassed protective devices (i.e. torque and limit switches), as described in NRC Information Notice 92-18. MOVs that are susceptible to IN 92-18 failures are identified in calculations SMNH-16-085, Evaluation of MOVs for Operator Manual Action (OMA) for IN 92-18 Type Hot Short Events, and SMNH-16-084, Evaluation of MOV Pressure Boundary Integrity for IN 92-18 Type Hot Short Events. IN 92-18 failures that challenge the

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Nuclear Safety Performance Criteria were identified as VFDRs during the NSCA analysis.

Per LAR Table S-2, Item #5, IN 92-18 concerns will be addressed by modifications.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1
Calculation SMNH-16-085, Evaluation of MOVs for Operator Manual Action (OMA) for IN 92-18 Type Hot Short Events, Ver. 1
Calculation SMNH-16-084, Evaluation of MOV Pressure Boundary Integrity for IN 92-18 Type Hot Short Events, Ver. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.1.1 Circuit Failure Criteria	Circuits for "required for hot shutdown" components: Because Appendix R Section III.G.1 requires that the hot shutdown capability remain "free of fire damage", there is no limit on the number of concurrent/simultaneous fire-induced circuit failures that must be considered for circuits for components "required for hot shutdown: located within the same fire area. For components classified as "required for hot shutdown", there is no limit on the duration of the hot short. It must be assumed to exist until an action is taken to mitigate its effects. Circuits required for the operation of or that can cause the mal-operation of "required for hot shutdown" components that are impacted by a fire are considered to render the component unavailable for performing its hot shutdown function unless these circuits are properly protected as described in the next sentence. The required circuits for any "required for hot shutdown" component, if located within the same fire area where they are credited for achieving hot shutdown, must be protected in accordance with one of the requirements of Appendix R Section III.G.2 or plant specific license conditions.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The circuit analysis instructions in PI-02-001 Rev. 1 do not differentiate between "required for safe shutdown" and "important to safe shutdown" components. Less conservative analytical techniques for addressing "important to safe shutdown" were not employed in the baseline circuit analysis.

For all components:
- Shorts-to-ground, short-circuits, line-to-line faults, and open circuits in any and all combinations are considered.

- Dynamic aspects of the cable faults are not considered. Each cable fault is evaluated for the possible equipment response it could elicit. Timing aspects and the ultimate circuit/equipment state are not factored into the criteria. For example, if a hot short between two conductors can produce a spurious opening of a solenoid valve, the analysis has identified "fail open" as an equipment response. How long the hot short persists before the fault degrades to a ground fault and terminates the spurious operation is not a factor considered by the analysis.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.5.1.1	Circuit Failure Criteria	Circuits for "important to safe shutdown" components: Circuits for components classified as "important to safe shutdown" are not specifically governed by the requirements of Appendix R Section III.G.1, III.G.2 or III.G.3. To address fire-induced impacts on these circuits, consider the three types of circuit failures identified above to occur individually on each conductor with the potential to impact any "important to safe shutdown" component with the potential to impact components "required for hot shutdown". In addition, consider the following additional circuit failure criteria for circuits for "important to safe shutdown" components located within the same fire area with the potential to impact components "required for hot shutdown": - As explained in Figure 3.5.2-3, multiple shorts-to-ground are to be evaluated for their impact on ungrounded circuits. - As explained in Figure 3.5.2-5, for ungrounded DC circuits, a single hot short from the same source is assumed to occur unless it can be demonstrated that the occurrence of a same source short is not possible in the affected fire area. If this approach is used, a means to configuration control this condition must be developed and maintained. - For the double DC break solenoid circuit design discussed in the NRC Memo from Gary Holahan, Deputy Director Division of Systems Technology, dated December 4, 1990 and filed under ML062300013, the effect of two hot shorts of the proper polarity in the same multi-conductor cable should be analyzed for non-high low pressure interface components. [Reference Figure B.3.3 (f) of NFPA 805-2001.] - Multiple spurious operations resulting from a fire-induced circuit failure affecting a single conductor must be included in the post-fire safe shutdown analysis. - Multiple fire-induced circuit failures affecting multiple conductors within the same multi-conductor cable with the potential to cause a spurious operation of an "important to safe shutdown" component must be assumed to exist concurrently. - Multiple fire-induced circuit failures affecting separate conductors in separate cables with the potential to cause a spurious operation of an "important to safe shutdown" component must be assumed to exist concurrently when the effect of the fire-induced circuit failure is sealed-in or latched. - Conversely, multiple fire-induced circuit failures affecting separate conductors in separate cables with the potential to cause a spurious operation of an "important to safe shutdown" component need not be assumed to exist concurrently when the effect of the fire-induced circuit failure is not sealed-in or latched. This criterion applies to consideration of concurrent hot shorts in secondary circuits and to their effect on a components primary control circuit. It is not to be applied to concurrent single hot shorts in primary control circuit for separate components in an MSO combination. - For components classified as "important to safe shutdown", the duration of a hot short may be limited to 20 minutes. (If the effect of the spurious actuation involves a "sealing in" or "latching" mechanism, that is addressed separately from the duration of the spurious actuation, as discussed above.) - For any impacted circuits for "important to safe shutdown" components that are located within the same fire area, protection in accordance with the requirements of Appendix R Section III.G.2 or plant specific license conditions may be used. In addition, consideration may be given to the use of fire modeling or operator manual actions, as an alternative to the requirements of Appendix R Section III.G.2. (Other resolution options may also be acceptable, if accepted by the Authority Having Jurisdiction.)

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Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The circuit analysis instructions in PI-02-001 Rev. 1 do not differentiate between "required for safe shutdown" and "important to safe shutdown" components. Less conservative analytical techniques for addressing "important to safe shutdown" were not employed in the baseline circuit analysis (e.g. How long the hot short persists before the fault degrades to a ground fault and terminates the spurious operation is not a factor considered by the analysis)

Multiple ground faults were considered. An existing but unspecified ground fault from the same power source was used as an analytical technique to analyze ungrounded circuits.

PI-02-001 Rev. 1 discusses the three cases for inter/intra-cable proper polarity hot-shorts. Consideration of proper polarity hot-shorts is in alignment with the guidance of NEI 00-01.

Hot shorts, open circuits and shorts-to-ground are all considered for both intra-cable sources and inter-cable sources and for any and all combinations. For the analysis of hot shorts, circuit analysis was performed using the hot probe method, utilizing multiple hot probes for components susceptible to multiple hot shorts.

Circuit analysis combined with the cable/equipment/system relationships in the NSCA model are used to evaluate multiple spurious operations resulting from fire induced failure of a single conductor, concurrent multiple fire-induced circuit failures affecting multiple conductors within the same multi-conductor cable, as well as concurrent multiple fire-induced circuit failures affecting separate conductors in separate cables.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.1.2 Spurious Operation Criteria	Spurious Operation Criteria: The following criteria address the effect of multiple spurious operations of components classified as either "required for hot shutdown" or "important to safe shutdown" on post-fire safe shutdown. These criteria are to be applied to the population of components whose spurious operation has been determined to be possible based on an application of the circuit failure criteria described above when assessing impacts to post-fire safe shutdown capability in any fire area. - The set of concurrent combinations of spurious operations provided through the MSO Process outlined in Section 4 and the list of MSO contained in Appendix G must be included in the analysis of MSOs. - MSOs do not need to be combined, except as explained in Section 4.4.3.4 of this document. - Section 4.4.3.4 states that the expert panel should review the plant specific list of MSOs to determine whether any of the individual MSOs should be combined due to the combined MSO resulting in a condition significantly worse than either MSO individually. - In this review, consideration of key aspects of the MSOs should be factored in, such as the overall number of spurious operations in the combined MSOs, the circuit attributes in Appendix B, and other physical attributes of the scenarios. – Specifically, if the combined MSOs involve more than a total of four components or if the MSO scenario requires consideration of sequentially selected cable faults of a prescribed type, at a prescribed time, in a prescribed sequence in order for the postulated MSO combination to occur, then this is considered to be beyond the required design basis for MSOs.
Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The criteria and resolution methodology outlined in endorsed portions of NEI-00-01, Revision 2 are used as a basis for resolving potential fire-induced MSOs, which includes Section 4. The HNP MSO analysis is based on the BWR Generic MSO List in Appendix G to Revision 3 of NEI 00-01. HNP Calculation SENH-15-009, Section 6.1.8, documents this analysis.

The number of circuit failures considered by the MSO analysis is consistent with the guidance in NEI-00-01 Revision 2 (as accepted by RG 1.189), provided the affected circuits/equipment are located within a single fire area under consideration. Thus, dependence on defense-in-depth attributes as described in RG 1.189 is not initially credited. Where a specific assumption is applied to a scenario or credit is taken for a limited number of circuit failures, the basis for the assumption or defense-in-depth attributes are captured in the scenario assessment.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

Calculation SENH-12-002, Expert Panel for Addressing Multiple Spurious Operations, Ver. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.1.3 Criteria/Assumptions	Assume that circuit contacts are initially positioned (i.e., open or closed) consistent with the normal mode/position of the "required for hot shutdown" or "important to safe shutdown" equipment as shown on the schematic drawings. The analyst must consider the position of the "required for hot shutdown" and "important to 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 "required for hot shutdown" and "important to safe shutdown equipment".

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

- Equipment was assumed to be in its "Normal" position immediately prior to the postulated fire event. When the normal and/or initial position of a component is indeterminate, the circuit analysis was based on the worst case starting conditions.

- In conducting circuit analysis, interposing contacts (aux contacts) were assumed in the state that corresponds to the "Initial" position of the component, which in most cases was the same as the "Normal" position for the component as described above.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 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 required for hot shutdown and important to safe 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 necessary to achieve and maintain 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 hot shutdown be free of fire damage and that these circuits be designed for the fire-induced effects of a hot short, short-to-ground, or an open circuit. With respect to the electrical distribution system, the issue of breaker coordination must also be addressed. Criteria for making the determination as to which breakers are to be classified as required for hot shutdown is contained in Appendix H. This section will discuss specific examples of each of the following types of circuit failures: - Open circuit - Short-to-ground - Hot short Also, refer to Appendix B for the circuit failure criteria to be applied in assessing the impact of the Plant Specific List of MSOs on post-fire safe shutdown.

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This is introductory guidance information and contains no specific requirements.

Reference Documents

Not Applicable

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 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 required for hot shutdown and important to 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. - 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 for hot shutdown and important to 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, possibly resulting in the occurrence of an additional fire in the location of the CT itself. Figure 3.5.2-1 shows an open circuit on a grounded control circuit. 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	None

Alignment Statement

Aligns

Alignment Basis

Open Circuit in any and all combinations has been considered during circuit analysis.

Current Transformers:
Per LAR Table S-3, IMP-10, Current Transformers that are credible secondary fire concerns will be identified and resolved.

Adequate fire rated barrier and penetration seal designs were established as part of the original Appendix R Post-Fire Safe Shutdown Analysis. These same barriers are being credited for the transition to NFPA 805 and will preclude the propagation of fire from one fire area to the next.

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

Implementation Items

IMP-10 Current Transformer analysis (SENH-16-001) will be updated to provide additional evaluation based on the latest industry guidance provided in NUREG/CR-7150, Vol. 3 dated November 2017. Should a credible CT secondary fire scenario be identified, additional analysis or modifications will be completed prior to implementation of NFPA 805 to ensure no detrimental effect on the NSCA, ΔCDF or ΔLERF.

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.5.2.2	Circuit Failures Due to a Short-to-Ground	This section provides guidance for addressing the effects of a short-to-ground on circuits for required for hot shutdown and important to 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. There is no limit to the number of shorts-to-ground that could be caused by the fire. 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. 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. 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 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. 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.

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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. If multiple grounds can occur in a single fire area, they should be assumed to occur simultaneously unless justification to the contrary is provided.

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. If multiple grounds can occur in a single fire area, they should be assumed to occur simultaneously unless justification to the contrary is provided. Note that a simultaneous short-to-ground at locations No. 1 and No. 2 could result in a spurious close/stop. This condition is identical to that portrayed in Figure 3.5.2-5 should a hot short occur on the ungrounded circuit shown in Figure 3.5.2-5 at location No. 1.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Shorts-to-ground in any and all combinations are considered during circuit analysis.

For ungrounded (AC and DC) circuits an unspecified ground fault from the same power source will be used as an analytical technique. This process accounts for potential ground faults due to the result of a fire.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.5.2.3	Circuit Failures Due to a Hot Short	This section provides guidance for analyzing the effects of a hot short on circuits for required for required for hot shutdown and important to 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 operation of equipment. The spuriously operated device (e.g., relay) may be interlocked with another circuit that causes the spurious operation of other equipment. This type of hot short is called an intra-cable hot short (also known as 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 and a de-energized conductor may also cause a spurious operation of equipment. This is called an inter-cable hot short (also known as cable-to-cable hot short/external hot short). - A hot short in the control circuitry for an MOV can bypass the MOV protective devices, i.e. torque and limit switches. This is the condition described in NRC Information Notice 92-18. In this condition, MOV motor damage can occur. Damage to the MOV motor could result in an inability to operate the MOV either remotely, using separate controls with separate control power, or manually using the MOV hand wheel. This condition could be a concern in two instances: (1) For fires requiring Control Room evacuation and remote operation from the Remote Shutdown Panel; (2) For fires where the selected means of addressing the effects of fire induced damage is the use of an operator manual action. In this latter case, analysis must be performed to demonstrate that the MOV thrust at motor failure does not exceed the capacity of the MOV hand wheel. For either case, analysis must demonstrate the MOV thrust at motor failure does not damage the MOV pressure boundary. 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 the impact of these individual cable faults on the operation of the equipment controlled by this circuit. 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:

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A hot short at this location would energize the open relay and result in the undesired opening of a motor-operated valve.

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 the impact of these cable faults on the operation of the equipment controlled by this circuit.

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	None

Alignment Statement

Aligns

Alignment Basis

All grounded and ungrounded circuits consider any and all hot shorts, as well as coincident multiple hot shorts.

Proper polarity three-phase hot shorts are considered only for high/low pressure interface.

The potential exists for hot shorts to damage MOV motors and/or valves as a result of bypassed protective devices (i.e. torque and limit switches), as described in NRC Information Notice 92-18. MOVs that are susceptible to IN 92-18 failures are identified in calculations SMNH-16-085, Evaluation of MOVs for Operator Manual Action (OMA) for IN 92-18 Type Hot Short Events, and SMNH-16-084, Evaluation of MOV Pressure Boundary Integrity for IN 92-18 Type Hot Short Events. IN 92-18 failures that challenge the Nuclear Safety Performance Criteria were identified as VFDRs during the NSCA analysis.

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Per LAR Table S-2, Item #5, IN 92-18 concerns will be addressed by modifications.

PI-02-001 Rev. 1 discusses the three cases for inter/intra-cable proper polarity hot-shorts. Consideration of proper polarity hot-shorts is in alignment with the guidance of NEI 00-01.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1
Calculation SMNH-16-085, Evaluation of MOVs for Operator Manual Action (OMA) for IN 92-18 Type Hot Short Events, Ver. 1
Calculation SMNH-16-084, Evaluation of MOV Pressure Boundary Integrity for IN 92-18 Type Hot Short Events, Ver. 1

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location

Physical location of equipment and cables shall be identified.

NEI 00-01 Ref	NEI 00-01 Section 3 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	None

Alignment Statement

Aligns

Alignment Basis

The Plant Data Management System (PDMS) serves as the master equipment list and cable & raceway system for Plant Hatch. This is where cable routing information and cable endpoint data are maintained.

Fire Data Manager (FDM) maintains the Safe Shutdown Equipment Functional State Circuit Analyses. A "snapshot" of PDMS cable routing data is stored in FDM and associated with the "required" safe shutdown cables.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location

NEI 00-01 Ref		NEI 00-01 Section 3 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. For raceway and cable endpoints in multiple fire areas, each fire area where the raceway or cable endpoint exists must be included. 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

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

PDMS relates equipment and cable raceways with fire zone and fire area locations. These locations define all cable route points as required to perform the nuclear safety capability analysis.

Together, PDMS and FDM provide the Safe Shutdown Equipment Functional State Circuit Analyses and associated location information for "required" safe shutdown cables.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.5.2.4	Circuit Failures Due to Inadequate Circuit Coordination	The evaluation of 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 hot 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 circuits' common power source breaker coordination issue, load breaker 4 must be fully coordinated with the feeder breaker to the Train A bus. 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 circuits of a common power source and evaluating circuit coordination cases 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 example, for breakers that have internal breaker tripping devices and do not require control power to trip the breaker, assure that the time-current characteristic curve for any affected load breaker is to the left of the time-current characteristic curve for the bus feeder breaker and that the available short circuit current for each affected breaker is to the right of the time-current characteristic curve for the bus feeder breaker or that the bus feeder breaker has a longer time delay in the breaker instantaneous range than the load breaker. For breakers requiring control power for the breaker to trip, the availability of the required control power must be demonstrated in addition to the proper alignment of the time-current characteristic curves described above. The requirement for the availability of control power would apply to load breakers fed from each safe shutdown bus where a fire-induced circuit failure brings into questions the availability of coordination for a required for hot shutdown component. - 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 circuit of concern are routed and the power source is required for hot shutdown. Prepare a list of the following information for each fire area: - Cables of concern.

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location

- 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 circuit of concern cables routed in an area of the same path as required by the power source. Evaluate adequate separation and other mitigation measures based upon the criteria in Appendix R, NRC staff guidance, and plant licensing bases.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

PI-02-001 Rev. 1 states that the circuit analysis was performed under the assumption that circuit overcurrent protective devices (e.g. relays, circuit breakers, fuses) are assumed to operate in accordance with their designed time-current tripping characteristics.

Power Supply Incorporation:
Power supply cable selection typically ends at the closest electrical isolation device for the component identified in the SSEL. For instance, power supply cables to an MCC will not be listed for a motor operated valve, only the power supply cable from the MCC to the valve will be listed for the valve. The MCC would be identified as a safe shutdown component in the SSEL and a separate circuit analysis performed for the MCC. Required power supplies are verified to ensure that they are on the SSEL, if not, they are added. If a power supply is required to be added to SSEL then a circuit analysis for the new power supply is required.

Electrical Coordination:
Breaker coordination is ensured by reviewing the time current curves from the plant's coordination study to ensure coordination. Coordination assures that the protective device nearest the fault operates prior to operation of upstream devices. The means of assuring circuit protection and coordination is provided in a series of calculations. These calculations demonstrate that the Class 1E and non-Class 1E power supplies credited for safe shutdown compliance do have adequate coordination.

All modifications are reviewed to ensure that the existing satisfactory circuit coordination is not compromised by future design changes.

Plant circuit breakers that require an external source of control power to perform their protective trip function have been identified and analyzed for common power supply concerns. This assessment has demonstrated that all load breakers on the credited switchgear will remain functional to isolate potentially fire affected (non-credited) loads.

Per LAR Table S-2, Item #6 and #7, coordination concerns will be addressed by modifications.

Multiple High Impedance Faults:
NEI 00-01 Appendix B.1 identifies Multiple High Impedance Faults (MHIF) as a unique common power supply concern and gives a resolution framework. Per LAR Table S-3, IMP-10, MHIF analysis will be documented.

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1
Calculation SENH-15-005, Protection Coordination Study for Alternate 4 KV Power Supply to Service Building, Ver. 1
Calculation SENH-88-011, Coordination Study - Cooling TWR MTRS & SWGR, Rev. 0
Calculation SENH-89-005, Coordination Study for 600V And 208V Switchgear Busses, Ver. 11
Calculation SENH-89-008, Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-010, Hatch U1 Micro Versa Trip Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-026, Coordination Study for 600V, 480V, & 120V Switchgears, Ver. 11
Calculation SENH-90-024, 125/250VDC Switchgear 1A & B / Breaker Coordination Utilizing EC Trip Devices, Ver. 6
Calculation SENH-91-011, Coordination Study for 250 VDC Switch Gear 2R22-S016 & S017, Ver. 7
Calculation SENH-94-013, Coordination Study for Non-Appendix R Breakers and Fuses in Response to REA HT-93753., Ver. 4
Calculation SETH-85-082, Appendix R Protection Device Coordination Study of 600/208/120V AC Circuits, Ver. 20
Calculation SETH-85-196, Appendix R Protective Device Coordination Study of 250/125V DC Circuits, Ver. 6

Implementation Items

IMP-9 Update electrical coordination calculations to document the current analysis of alignment with NEI 00-01 Appendix B.1 regarding Multiple High Impedance Faults (MHIF).

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.5.2.5	Circuit Failures Due to Common Enclosure Concerns	The common enclosure 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	None

Alignment Statement

Aligns

Alignment Basis

PI-02-001 Rev. 1 states that the circuit analysis was performed under the assumption that circuit overcurrent protective devices (e.g. relays, circuit breakers, fuses) are assumed to operate in accordance with their design time-current tripping characteristics.

The electrical circuit design for HNP provides proper circuit protection in the form of circuit breakers, fuses and other devices that are designed to isolate cable faults before the cable ignition temperature is reached. Adequate electrical circuit protection and cable sizing were included as part of the original plant electrical design and are maintained as part of the design change process. Adequate fire rated barrier and penetration seal designs were established as part of the original Appendix R Post-Pire Safe Shutdown Analysis. These same barriers are being credited for the transition to NFPA 805 and will preclude the propagation of fire from one fire area to the next.

DC Control Power is included in the analysis when it is required for proper breaker operation and fault clearing capabilities.

Per LAR Table S-3, IMP-10, Current Transformers that are credible secondary fire concerns will be identified and resolved.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1
Calculation SENH-16-001, Current Transformer Analysis Report, Ver. 1
Calculation SENH-15-005, Protection Coordination Study for Alternate 4 KV Power Supply to Service Building, Ver. 1
Calculation SENH-88-011, Coordination Study - Cooling TWR MTRS & SWGR, Rev. 0
Calculation SENH-89-005, Coordination Study for 600V And 208V Switchgear Busses, Ver. 11
Calculation SENH-89-008, Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-010, Hatch U1 Micro Versa Trip Overcurrent Coordination Study, Ver. 7
Calculation SENH-89-026, Coordination Study for 600V, 480V, & 120V Switchgears, Ver. 11

NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location

Calculation SENH-90-024, 125/250VDC Switchgear 1A & B / Breaker Coordination Utilizing EC Trip Devices, Ver. 6
Calculation SENH-91-011, Coordination Study for 250 VDC Switch Gear 2R22-S016 & S017, Ver. 7
Calculation SENH-94-013, Coordination Study for Non-Appendix R Breakers and Fuses in Response to REA HT-93753., Ver. 4
Calculation SETH-85-082, Appendix R Protection Device Coordination Study of 600/208/120V AC Circuits, Ver. 20
Calculation SETH-85-196, Appendix R Protective Device Coordination Study of 250/125V DC Circuits, Ver. 6

Implementation Items

IMP-10 Current Transformer analysis (SENH-16-001) will be updated to provide additional evaluation based on the latest industry guidance provided in NUREG/CR-7150, Vol. 3 dated November 2017. Should a credible CT secondary fire scenario be identified, additional analysis or modifications will be completed prior to implementation of NFPA 805 to ensure no detrimental effect on the NSCA, ΔCDF or ΔLERF.

NFPA 805 Section: 2.4.2.4 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 Section 3 Guidance
3.4	Fire Area Assessment and Compliance Strategies	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. Knowing which components and systems are performing which safe shutdown functions, the required and important to SSD components can be classified. Once these component classifications have been made the tools available for mitigating the affects of fire induced damage can be selected. Refer to Appendix H for additional guidance on classifying components as either required for hot shutdown or important to safe shutdown. For MSOs the Resolution Methodology outlined in Section 4, Section 5, Appendix B and Appendix G should be applied. Components in each MSO are classified as either required for hot shutdown or important to safe shutdown components using the criteria from Appendix H. Similarly, this classification determines the available tools for mitigating the affects of fire-induced damage to the circuits for these components. 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	None

Alignment Statement

Not Required

Alignment Basis

This is introductory guidance information and contains no specific requirements.

Reference Documents

Not Applicable

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 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

Applicable

Comments

None

Alignment Statement

Not Required

Alignment Basis

This is an introductory paragraph and contains no specific requirements.

Reference Documents

Not Applicable

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.1 Criteria/Assumptions	Assume only one fire in any single fire area at a time.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.4 explicitly states the assumption listed in Section 3.4.1.1 of NEI 00-01. The analysis assumes only one fire area in any single fire area at a time.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.2 Criteria/Assumptions	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 postulated in the regulation.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.4 explicitly states the criteria listed in Section 3.4.1.2 of NEI 00-01.

The analysis assumes that a fire may affect all unprotected cables and equipment within a given fire area. The size and intensity of the fire causing this system and equipment damage is not determined. This is conservative and bounds the exposure fire that is postulated in the regulation.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.3 Criteria/Assumptions	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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.4 explicitly states the criteria listed in Section 3.4.1.3 of NEI 00-01.

The analysis of the effects of a fire on nuclear safety capability for each area of the plant is a relatively complex process that involves many pieces of input data that have distinct relationships. PDMS routing information, the fault trees and the ARCPlus™ Database determine the functional states that are impacted by in a fire area. All cable and equipment impacts affecting the required safe shutdown path are addressed. This is accomplished in the analysis by achieving success in the fault tree.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.4 Criteria/Assumptions	Use the criteria from Appendix H to classify each impacted cable/component as either a required or important to SSD cable/component.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.4 identifies the overall process utilized to identify the combinations of plant components for each plant system that is identified as being required to satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

Hatch conservatively treats all plant equipment required to achieve and maintain safe and stable plant conditions as required for hot shutdown.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.5 Criteria/Assumptions	Use operator manual actions where appropriate, for cable/component impacts classified as important to SSD cable/components, to achieve and maintain post-fire safe shutdown conditions in accordance with NRC requirements (refer to Appendix E). For additional criteria to be used when determining whether an operator manual action may be used for a flow diversion off of the primary flow path, refer to Appendix H.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.4 explicitly states the criteria listed in Section 3.4.1.5 of NEI 00-01.

Manual actions are used throughout the safe shutdown analysis. All manual actions are evaluated for their risk significance to support safe shutdown. All risk significant actions have had their feasibility evaluated.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.6 Criteria/Assumptions	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	None

Alignment Statement

Aligns with Intent

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.4 explicitly states the criteria listed in Section 3.4.1.6 of NEI 00-01.

NFPA 805 Nuclear Safety Performance Criteria (NSPC) requires the licensee to demonstrate that the plant can achieve and maintain a safe and stable condition, but does not explicitly require the licensee to demonstrate that cold shutdown can be achieved within 72 hours and maintained indefinitely thereafter. The HNP NFPA 805 NSPC analysis has defined the safe and stable condition as being able to achieve and maintain hot shutdown until such a time as the plant can either transition to cold shutdown, or can safely return to power operation.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.7 Criteria/Assumptions	For the components on the required safe shutdown path classified as required hot shutdown components as defined in Appendix H, 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 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 components impacted circuit(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 non-inerted 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, LARs and licensing change processes to satisfy the requirements mentioned above and to demonstrate equivalency depending upon the plant's license requirements.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.4 explicitly states the criteria listed in Section 3.4.1.7 of NEI 00-01.

This guidance is followed when performing fire area compliance assessments at HNP.

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.8 Criteria/Assumptions	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 operation, is to be addressed in accordance with regulatory requirements and the NPP's current licensing basis. With respect to MSOs, the criteria in Section 4, Appendix B, Appendix G and Appendix H should be used.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.4 explicitly states the criteria listed in Section 3.4.1.8 of NEI 00-01.

In certain special isolated cases the guidance does provide the option ofconsidering the use of other equipment that may not have been previously credited or not be specifically identified, but which can perform the same safe shutdown function as the impacted equipment. This is done on a case by case basis. For example, such as with skid mounted components associated with the emergency diesel generators or control room condensing units, all of the required components may not be shown on the logic. In these cases, the effect of fire damage to these support components is captured by tying the component to a basic event that is depicted on the CAFTA logic.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.9 Criteria/Assumptions	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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 6.1.4 explicitly states the criteria listed in Section 3.4.1.9 of NEI 00-01.

HNP Calculation SENH-15-008, Instrument Sense Line Analysis for Appendix R and NSCA SSD explicitly addressed the criteria listed in Section 3.4.1.9 of NEI 00-01 in the purpose and scope of the calculation.

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 9.1.7 identifes the methodology of how instrument tubing was evaluated in the NSCA.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

SENH-15-008, Instrument Sense Line Analysis for Appendix R & NSCA SSD Models, Ver. 2

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref		NEI 00-01 Section 3 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:

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This is introductory guidance and contains no specific requirements.

Reference Documents

Not Applicable

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref		NEI 00-01 Section 3 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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 11.0 identifies the overall process utilized to perform deterministic fire area assessment and the fire area assessment results for the NSCA components identified as being required to satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

The Plant Data Management System (PDMS) controls the cable routing, fire zone and fire area information. The Fire Data Manager (FDM) control the logic associated with cables to their components. The NFPA 805 compliance software (ARCPlus) relates the information in PDMS to the safe shutdown fault tree by basic events. Reports can be generated by both software packages.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref		NEI 00-01 Section 3 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. Determine which components are required hot shutdown components and which components are important to SSD components using the guidance in Appendix H. Based on an assessment as described above, designate the required safe shutdown path(s) for the fire area. Classify the components on the required safe shutdown path necessary to perform the required safe shutdown functions as required safe shutdown components. Identify all equipment not in the safe shutdown path whose spurious operation or mal-operation could affect the shutdown function. Criteria for classifying these components as required for hot shutdown or as important to SSD is contained in Appendix H. Include the affected 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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 11.0 identifies the overall process utilized to perform deterministic fire area assessment and the fire area assessment results for the NSCA components identified as being required to satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

The ARCPlus software, PDMS and CAFTA safe shutdown fault tree are used by the analyst to identify the least affected path of safe shutdown equipment. In a manner consistent with the guidance above, the best overall safe shutdown strategy is developed.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 11.0 identifies the overall process utilized to perform deterministic fire area assessment and the fire area assessment results for the NSCA components identified as being required to satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

The impact of the fire-induced equipment and cable failures are evaluated for the credited train of safe shutdown systems and for those components on the non-credited trains of safe shutdown equipment that might adversely impact the credited train. PI-02-001, Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Section 7.0 identifies guidance consistent with NEI 00-01 on performing circuit analysis and dispositioning cable failures in the NSCA analysis.

Reference Documents

PI-02-001, SNC Project Instruction Document for Performance of NSCA and Fire PRA Circuit Analysis Using ARCPlus FDM, Rev. 1

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

Calculation SENH-17-002, Circuit Analysis for Fire Safety Analysis – Fire Data Manager Output, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref		NEI 00-01 Section 3 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 1-1): Required for Hot Shutdown Components: - Re-design the circuit or component to eliminate the concern. This option will require a revision to the post-fire safe shutdown analysis. - Re-route the cable of concern. This option will require a revision to the post-fire safe shutdown analysis. - Protect the cable in accordance with III.G.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. - 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, LARs, Generic Letter 86-10 evaluation or fire protection design change evaluations with a licensing change process. Important to Safe Shutdown Components: - Any of the options provided for required for hot shutdown components. - Perform and operator manual action in accordance with Appendix E. - Address using fire modeling or a focused-scope fire PRA using the methods of Section 5 for MSO impacts. [Note: The use of fire modeling will require a review by the Expert Panel and the use of a focused-scope fire PRA will require a LAR.] 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	None

Alignment Statement

Aligns

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report, Section 8.0 identifies the overall process utilized to perform deterministic fire area assessment and the fire area assessment results for the NSCA components identified as being required to satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

This guidance is followed in the NSCA when performing fire area compliance assessments for HNP. Compliance strategies and developed and entered into the ARCPlus software. A set of standardized templates adapted for HNP are used to recover the necessary basic events and gates to demonstrate success in the safe shutdown fault tree. Circuit failures having the potential to adversley impact the compliance strategy were identified as Variances From Determinsitc Requirements (VFDRs) and will be transitioned to the Fire Risk Evaluations.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref		NEI 00-01 Section 3 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	None

Alignment Statement

Aligns

Alignment Basis

HNP Calculation SENH-15-009, Nuclear Safety Capability Assessment Report documents the compliance strategies for both units for a fire in each Fire Area. The ARCPlus software is used to apply compliance strategies to basic events and gates until overall success is achieved in the fault tree. The ARCPlus Compliance Assessment by Scenario report can then be generated, which displays all affected equipment and cables, as well as the compliance strategies used to achieve success in the fault tree.

Reference Documents

Calculation SENH-15-009, Nuclear Safety Capability Assessment (NSCA) Report, Ver. 1

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

Table C-1 746 Pages Attached
Table C-2 125 Pages Attached

Attachment C is redacted in its entirety.

D. NEI 04-02 Non-Power Operational Modes Transition

6 Pages Attached

Attachment D is redacted in its entirety.

Compartment Identification – Control Building

Compartment Selection and Justification Basis

Although the Control Building includes unit-specific and common locations., the support systems are common to the entire building; as such, this compartment will address the RCA areas throughout the Control Building.

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
0001	Working Floor
0002B	Control Building Freight Elevator
0007A	Control Building East Corridor
0007C	HP Cold Lab Storage Area
0007D	Respirator Room
0007E	HP Cold Lab Test Area
0007F	SCBA Room
0014A	RC Lab
0014B	Health Physics Hallway
0014C	Health Physics Area Storage
0014D	HP Reference Area
0014F	Decontamination Room and Shower
0014G	HP Counting Room
0014H	Hot Lab
0014I	HP Foreman's Office
0014K	Working Floor
0014M	Men's Room
1003	Oil Storage Tank Room
1004	Station Battery Room 1A
1005	Station Battery Room 1B
1006	Unit 1 Water Analysis Room
1008	Unit 1 AC Inverter Room
1009	RPS Battery Room South
1010	RPS Battery North Room
2003	Oil Storage Tank Room
2004	Station Battery Room 2A

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
2005	Station Battery Room 2B
2006	Unit 2 Water Analysis Room
2008	Unit 2 AC Inverter Room
2009	RPS Battery North Room
2010	RPS Battery South Room

Related Pre-Fire Plans	
Pre-Fire Plan	Pre-Fire Plan Description
A-43965 Sh. 5A	Control Building Area Elevation 112'-0"
A-43965 Sh. 5B	Control Building General Area Elevation 112'-0"
A-43965 Sh. 6A	Control Building Freight Elevator Elevation 112'-0"
A-43965 Sh. 6B	Control Building Freight Elevator Elevation 112'-0"
A-43965 Sh. 7A	Control Building East Corridor Elevation 112'-0"
A-43965 Sh. 7B	Control Building East Corridor Elevation 112'-0"
A-43965 Sh. 8A	Water Analysis Room Elevation 112'-0"
A-43965 Sh. 8B	Water Analysis Room Elevation 112'-0"
A-43965 Sh. 9A	Cold Lab and Adjacent Rooms Control Bldg. Elevation 112'-0"
A-43965 Sh. 9B	Cold Lab and Adjacent Rooms Control Bldg. Elevation 112'-0"
A-43965 Sh. 10A	Oil Storage Tank Room Control Bldg. Elevation 112'-0"
A-43965 Sh. 10B	Oil Storage Tank Room Control Bldg. Elevation 112'-0"
A-43965 Sh. 11A	W Station Battery Room 1A Control Building Elevation 112'-0"
A-43965 Sh. 11B	W Station Battery Room 1A Control Building Elevation 112'-0"
A-43965 Sh. 12A	E Station Battery Room 1B Control Bldg. Elevation 112'-0"
A-43965 Sh. 12B	E Station Battery Room 1B Control Bldg. Elevation 112'-0"
A-43965 Sh. 13A	AC Inverter Room Control Bldg. Elevation 112'-0"
A-43965 Sh. 13B	AC Inverter Room Control Bldg. Elevation 112'-0"
A-43965 Sh. 14A	RPS Battery Room 1B Control Bldg. Elevation 112'-0"
A-43965 Sh. 14B	RPS Battery Room 1B Control Bldg. Elevation 112'-0"
A-43965 Sh. 15A	RPS Battery Room 1A Control Bldg. Elevation 112'-0"
A-43965 Sh. 15B	RPS Battery Room 1A Control Bldg. Elevation 112'-0"
A-43965 Sh. 16A	Oil Storage Tank Room Control Bldg. Elevation 112'-0"

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43965 Sh. 16B	Oil Storage Tank Room Control Bldg. Elevation 112'-0"
A-43965 Sh. 17A	W Station Battery Room 2A Control Bldg. Elevation 112-0"
A-43965 Sh. 17B	W Station Battery Room 2A Control Bldg. Elevation 112-0"
A-43965 Sh. 18A	E Station Battery Room 2B Control Bldg. Elevation 112'-0"
A-43965 Sh. 18B	E Station Battery Room 2B Control Bldg. Elevation 112'-0"
A-43965 Sh. 19A	Water Analysis Room Control Bldg. Elevation 112'-0"
A-43965 Sh. 19B	Water Analysis Room Control Bldg. Elevation 112'-0"
A-43965 Sh. 20A	AC Inverter Room Control Bldg. Elevation 112'-0"
A-43965 Sh. 20B	AC Inverter Room Control Bldg. Elevation 112'-0"
A-43965 Sh. 21A	RPS Battery Room 2A Control Bldg. Elevation 112'-0"
A-43965 Sh. 21B	RPS Battery Room 2A Control Bldg. Elevation 112'-0"
A-43965 Sh. 22A	RPS Battery Room 2B Control Bldg. Elevation 112'-0"
A-43965 Sh. 22B	RPS Battery Room 2B Control Bldg. Elevation 112'-0"
A-43965 Sh. 23A	Health Physics Area Control Bldg. Elevation 130'-0"
A-43965 Sh. 23B	Health Physics Area Control Bldg. Elevation 130'-0"
A-43965 Sh. 24A	General Area and Corridors Control Bldg. Elevation 130'-0"
A-43965 Sh. 24B	General Area and Corridors Control Bldg. Elevation 130'-0"

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation

The Control Building exhaust system is split between both Unit 1 and Unit 2, with three 50% capacity Unit 1 fans and two 100% capacity Unit 2 fans. These fans exhaust air to the Unit 1 and 2 Reactor Building vent plenums, where it is filtered by two 50% capacity filter trains, each consisting of a bank of prefilters, carbon adsorbers, and HEPA filters, before being released to the environment. Supply air is delivered to areas of the Control Building by two normally operating fans, with a third on standby.

Division I and II battery rooms are each provided with a 100% capacity exhaust fans that are independent of each other. These fans are designed to operate in case of a loss of offsite power when normal ventilation is not available or when required by operations personnel.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation
Liquid wastes from the Control Building are processed by the Unit 2 radwaste system and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train. Liquids from the floor drain sample tank, if below the technical specification limits, are discharged from the plant after dilution with the cooling tower blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Laboratory drains and other drains containing chemicals are received by the chemical waste tank. These wastes are processed by filtration (after being neutralized, if required). If a decision to reuse the water has been made, then the water passes through an ion exchange unit and sent to the condensate storage tank for reuse. However, if the sampling and analysis indicate that the radioactivity concentrations are low enough to meet the discharge criteria, the water in the sample tank is released to the discharge pipe. Hot shower wastes are treated in the Unit 1 radwaste system.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-Fire Plans to Minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the pre-fire plans to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage, ventilation, and monitoring systems described above remain available during non-power operations.

Conclusion
Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Main Control Room Roof

Compartment Selection and Justification Basis

The Main Control Room Roof is considered part of the Control Building but is open to the Turbine Building 164' elevation operating deck due to lack of a roof or ceiling. The liquid support system for the Main Control Room Roof is common to the Control Building, and the gaseous support system is common to the Turbine Building. Since this unique combination of support systems applies only to the Main Control Room Roof, this compartment will address that zone independently.

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
0031	Main Control Room Roof

Related Pre-Fire Plans	
Pre-Fire Plan	Pre-Fire Plan Description
A-43965 Sh. 50A	Control Room Roof Control Bldg. Elevation 180'-0"
A-43965 Sh. 50B	Control Room Roof Control Bldg. Elevation 180'-0"

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation

Due to the Main Control Room Roof being open to the Turbine Building, any gaseous effluents are processed by the Turbine Building ventilation system. Turbine Building exhaust air flows from low-radiation areas to high-radiation areas and is then ducted to filter banks in an effort to minimize the escape of potential airborne radioactivity to the outside atmosphere via the Reactor Building vent plenum. Potentially contaminated areas of the Turbine Building are maintained at a slight negative pressure to ensure the inward leakage of air. Normally, air is supplied by one fan in operation while a second is on standby. The Turbine Building ventilation exhaust system is supplied by two pairs of exhaust fans, one pair per unit. The exhaust fans are normally powered by non-diesel-backed power; however, if offsite power is lost, each fan is capable of being powered by diesel-backed power. The filter banks for the Turbine Building employ HEPA and charcoal filters in order to minimize radioactive particulate and halogen releases. The filter banks are employed with radiation monitors and are backed up by the Reactor Building vent plenum isokinetic probe. Radiation monitors survey the bank performance and provide the Main Control Room with high-level annunciation. Monitored and filtered air is exhausted from the Turbine Building by a duct system to the outside environment via the Reactor Building vent plenum.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation
Liquid wastes from the Control Building are processed by the Unit 2 radwaste system and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train. Liquids from the floor drain sample tank, if below the technical specification limits, are discharged from the plant after dilution with the cooling tower blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Laboratory drains and other drains containing chemicals are received by the chemical waste tank. These wastes are processed by filtration (after being neutralized, if required). If a decision to reuse the water has been made, then the water passes through an ion exchange unit and is sent to the condensate storage tank for reuse. However, if the sampling and analysis indicate that the radioactivity concentrations are low enough to meet the discharge criteria, the water in the sample tank is released to the discharge pipe. Hot shower wastes are treated in the Unit 1 radwaste system.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-Fire Plans to Minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the pre-fire plans to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage, ventilation, and monitoring systems described above remain available during non-power operations.

Conclusion
Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Unit 1 Turbine Building

Compartment Selection and Justification Basis

The Unit 1 Turbine Building includes unit-specific locations. The gaseous support systems for the Turbine Building are common but the liquid support systems are unit specific; as such, this compartment will address the Unit 1 Turbine Building only.

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
0101A	Unit 1 Turbine Building Main Floor Area
0101B	Unit 1 Main Turbine Deck Area
0101C	Unit 1 Reactor Feed Pump Room A
0101D	Unit 1 Reactor Feed Pump Room B
1101A	Area Under Main Condenser
1101C	Condensate Pump Area
1101D	Steam Jet Air Ejector Rooms
1101E	Vacuum Pump Room
1101F	Condensate Polishing Room
1101G	RBCCW
1101H	East Corridor
1101I	West Cableway
1101J	Turbine Building Working Floor
1101K	Main Condenser Area
1101M	East Switchgear Mezzanine
1101N	West Switchgear Mezzanine
1102	Northwest Stairway
1103	Northeast Stairway
1104	East Cableway
1105	East Cableway Foyer

Related Pre-Fire Plans	
Pre-Fire Plan	Pre-Fire Plan Description
A-43965 Sh. 34A	Unit 1 East Cableway Control Bldg. Elevation 130'-0"
A-43965 Sh. 34B	Unit 1 East Cableway Control Bldg. Elevation 130'-0"
A-43965 Sh. 34C	Unit 1 East Cableway Foyer Control Bldg. Elevation 130'-0"

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43965 Sh. 34D	Unit 1 East Cableway Foyer Control Bldg. Elevation 130'-0"
A-43965 Sh. 75A	Condenser Bay Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 75B	Condenser Bay Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 76A	Condenser Pump Area Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 76B	Condenser Pump Area Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 77A	SJAE Room Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 77B	SJAE Room Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 78A	Condenser Polishing Room Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 78B	Condenser Polishing Room Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 79A	RBCCW Room Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 79B	RBCCW Room Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 80A	Unit 1 East Corridor Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 80B	East Corridor Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 81A	West Cableway Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 81B	West Cableway Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 82A	Working Floor Turbine Bldg. Elevation 130'-0"
A-43965 Sh. 82B	Working Floor Turbine Bldg. Elevation 130'-0"
A-43965 Sh. 83A	Condenser Bay Turbine Bldg. Elevation 130'-0"
A-43965 Sh. 83B	Condenser Bay Turbine Bldg. Elevation 130'-0"
A-43965 Sh. 84A	Northeast Switchgear Mezzanine Turbine Bldg. Elevation 147'-0"
A-43965 Sh. 84B	Northeast Switchgear Mezzanine Turbine Bldg. Elevation 147'-0"
A-43965 Sh. 85A	Northwest Switchgear Mezzanine Turbine Bldg. Elevation 147'-0"
A-43965 Sh. 85B	Northwest Switchgear Mezzanine Turbine Bldg. Elevation 147'-0"
A-43965 Sh. 86A	Main Steam Line Area Turbine Bldg. Elevation 147'-0"
A-43965 Sh. 86B	Main Steam Line Area Turbine Bldg. Elevation 147'-0"
A-43965 Sh. 87A	Main Floor Area Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 87B	Main Floor Area Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 88A	Main Turbine Deck Area Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 88B	Main Turbine Deck Area Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 89A	Reactor Feed Pump A Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 89B	Reactor Feed Pump A Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 90A	Reactor Feed Pump B Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 90B	Reactor Feed Pump B Turbine Bldg. Elevation 164'-0"

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation
Turbine Building exhaust air flows from low-radiation areas to high-radiation areas and is then ducted to filter banks in an effort to minimize the escape of potential airborne radioactivity to the outside atmosphere via the Reactor Building vent plenum. Potentially contaminated areas of the Turbine Building are maintained at a slight negative pressure to ensure the inward leakage of air. Normally, air is supplied by one fan in operation while a second is on standby. The Turbine Building ventilation exhaust system is supplied by two pairs of exhaust fans, one pair per unit. The exhaust fans are normally powered by non-diesel-backed power; however, if offsite power is lost, each fan is capable of being powered by diesel-backed power. The filter banks for the Turbine Building employ HEPA and charcoal filters in order to minimize radioactive particulate and halogen releases. The filter banks are employed with radiation monitors and are backed up by the Reactor Building vent plenum isokinetic probe. Radiation monitors survey the bank performance and provide the Main Control Room with high-level annunciation. Monitored and filtered air is exhausted from the Turbine Building by a duct system to the outside environment via the Reactor Building vent plenum.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation
Liquid wastes from the Unit 1 Turbine Building are processed by the Unit 1 radwaste system and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train. Liquids from the floor drain sample tank, if below the technical specification limits, are discharged from the plant after dilution with the cooling tower blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Wastes from the Turbine Building decontamination drains, such as laboratory drains and other drains containing chemicals, are received by the chemical waste tank. These wastes are processed by filtration (after being neutralized, if required). If a decision to reuse the water has been made, then the water passes through an ion exchange unit and is sent to the condensate storage tank for reuse. However, if the sampling and analysis indicate that the radioactivity concentrations are low enough to meet the discharge criteria, the water in the sample tank is released to the discharge pipe. Hot shower wastes are treated by the Unit 1 radwaste system.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-Fire Plans to Minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the pre-fire plans to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage, ventilation, and monitoring systems described above remain available during non-power operations.

Conclusion
Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Unit 2 Turbine Building

Compartment Selection and Justification Basis

The Unit 2 Turbine Building includes unit-specific locations. The gaseous support systems for the Turbine Building are common but the liquid support systems are unit specific; as such, this compartment will address the Unit 2 Turbine Building only.

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
0101I	Unit 2 Main Turbine Deck Area
0101J	Unit 2 Turbine Building Main Floor Area
0101K	Unit 2 Reactor Feed Pump A
0101L	Unit 2 Reactor Feed Pump Room B
2101A	Area Under Main Condenser
2101C	Condensate Pump Area
2101D	Steam Jet Air Ejector Rooms
2101E	Vacuum Pump Room
2101F	Condensate Polishing Room
2101G	Offgas Recombiner
2101H	East Corridor
2101I	West Cableway and Service Building Tunnel
2101J	Turbine Building Working Floor
2101K	Turbine Building 130-ft. Main Condenser Area
2101M	East Switchgear Mezzanine
2101N	West Switchgear Mezzanine
2102	Southwest Stairwell
2103	Southeast Stairway
2104	East Cableway

Related Pre-Fire Plans	
Pre-Fire Plan	Pre-Fire Plan Description
A-43965 Sh. 43A	Unit 2 East Cableway Control Bldg. Elevation 130'-0"
A-43965 Sh. 43B	Unit 2 East Cableway Control Bldg. Elevation 130'-0"
A-43965 Sh. 124A	Condenser Bay Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 124B	Condenser Bay Turbine Bldg. Elevation 112'-0"

A-43965 Sh. 125A	Condenser Pump Area Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 125B	Condenser Pump Area Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 126A	Condensate Polishing Room Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 126B	Condensate Polishing Room Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 127A	East Corridor Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 127B	East Corridor Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 128A	West Cableway Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 128B	West Cableway Turbine Bldg. Elevation 112'-0"
A-43965 Sh. 129A	Working Floor Turbine Bldg. Elevation 130'-0"
A-43965 Sh. 129B	Working Floor Turbine Bldg. Elevation 130'-0"
A-43965 Sh. 130A	Condenser Bay Turbine Bldg. Elevation 130'-0"
A-43965 Sh. 130B	Condenser Bay Turbine Bldg. Elevation 130'-0"
A-43965 Sh. 131A	Southeast Switchgear Mezzanine Turbine Bldg. Elevation 147'-0"
A-43965 Sh. 131B	Southeast Switchgear Mezzanine Turbine Bldg. Elevation 147'-0"
A-43965 Sh. 132A	Southwest Switchgear Mezzanine Turbine Bldg. Elevation 147'-0"
A-43965 Sh. 132B	Southwest Switchgear Mezzanine Turbine Bldg. Elevation 147'-0"
A-43965 Sh. 133A	Main Turbine Deck Area Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 133B	Main Turbine Deck Area Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 134A	Main Floor Area Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 134B	Main Floor Area Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 135A	Reactor Feed Pump A Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 135B	Reactor Feed Pump A Turbine Bldg. Elevation 164'-0"
A-43965 Sh. 136A	Reactor Feed Pump B Turbine Bldg. Elevation 164'-0"
A-43065 Sh. 136B	Reactor Feed Pump B Turbine Bldg. Elevation 164'-0"

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation

Turbine Building exhaust air flows from low-radiation areas to high-radiation areas and is then ducted to filter banks in an effort to minimize the escape of potential airborne radioactivity to the outside atmosphere via the Reactor Building vent plenum. Potentially contaminated areas of the Turbine Building are maintained at a slight negative pressure to ensure the inward leakage of air. Normally, air is supplied by one fan in operation while a second is on standby. The Turbine Building ventilation exhaust system is supplied by two pairs of exhaust fans, one pair per unit. The exhaust fans are normally powered by non-diesel-backed power; however, if offsite power is lost, each fan is capable of being powered by diesel-backed power. The filter banks for the Turbine Building employ HEPA and charcoal filters in order to minimize radioactive particulate and halogen releases. The filter banks are employed with radiation monitors and are backed up by the Reactor Building vent plenum isokinetic probe. Radiation

monitors survey the bank performance and provide the Main Control Room with high-level annunciation. Monitored and filtered air is exhausted from the Turbine Building by a duct system to the outside environment via the Reactor Building vent plenum.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation
Liquid wastes from the Unit 2 Turbine Building are processed by the Unit 2 radwaste system and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train. Liquids from the floor drain sample tank, if below the technical specification limits, are discharged from the plant after dilution with the cooling tower blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Wastes from the Turbine Building decontamination drains, such as laboratory drains and other drains containing chemicals, are received by the chemical waste tank. These wastes are processed by filtration (after being neutralized, if required). If a decision to reuse the water has been made, then the water passes through an ion exchange unit and is sent to the condensate storage tank for reuse. However, if the sampling and analysis indicate that the radioactivity concentrations are low enough to meet the discharge criteria, the water in the sample tank is released to the discharge pipe. Hot shower wastes are treated by the Unit 1 radwaste system.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-Fire Plans to Minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the pre-fire plans to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage, ventilation, and monitoring systems described above remain available during non-power operations.

Conclusion
Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Unit 1 Reactor Building

Compartment Selection and Justification Basis

The Unit 1 Reactor Building (below the refueling floor) includes unit-specific locations. The support systems for the Reactor Building are unit specific; as such, this compartment will address the Unit 1 Reactor Building only.

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
1201	Unit 1 Drywell and Torus
1203A	Reactor Building South
1203B	Reactor Building Southeast Corner Room
1203C	Reactor Building Southwest Corner Room
1203F	Working Floor South
1203I	Stairwell Vestibule
1203K	Working Floor South
1205A	Reactor Building North
1205B	Reactor Building Northeast Corner Room
1205C	Northwest Corner Room
1205F	Working Floor North
1205I	Working Floor North
1205L	Reactor Water Cleanup (RWCU) Heat Exchanger Room
1205M	Cleanup Phase Separators
1205N	HVAC Room
1205Q	Standby Gas Filter Room
1205R	Working Floor North
1205S	Working Floor South
1205T	Filter Demineralizer Room
1205U	Southwest Corridor
1205W	Room South of Spent Fuel Pit
1205X	Stack Monitoring Room
1205Y	Working Floor
1205Z	HPCI Pump Room
1609	Unit 1 Reactor Building 185 ft Roof

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43965 Sh. 51A	SE RHR & Core Spray Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 51B	SE RHR & Core Spray Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 52A	RCIC Pump & Turbine Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 52B	RCIC Pump & Turbine Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 53A	NE RHR & Core Spray Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 53B	NE RHR & Core Spray Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 54A	CRD and DRW Sump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 54B	CRD and DRW Sump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 55A	HPCI Pump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 55B	HPCI Pump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 56A	South Torus Area Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 56B	South Torus Area Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 57A	North Torus Area Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 57B	North Torus Area Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 58A	RB South CRD Area Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 58B	RB South CRD Area Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 59A	RB North CRD Area Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 59B	North CRD Area Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 60A	Working Floor South Reactor Bldg. Elevation 158'-0"
A-43965 Sh. 60B	Working Floor South Reactor Bldg. Elevation 158'-0"
A-43965 Sh. 61A	Working Floor and Pump Room Reactor Building Elevation 158'-0"
A-43965 Sh. 61B	Working Floor and Pump Room Reactor Bldg. Elevation 158'-0"
A-43965 Sh. 62A	RWCU Heat Exchanger Room Reactor Bldg. Elevation 158'-0"
A-43965 Sh. 62B	RWCU Heat Exchanger Room Reactor Bldg. Elevation 158'-0"
A-43965 Sh. 65A	HVAC Room Reactor Bldg. Elevation 164'-0"
A-43965 Sh. 65B	HVAC Room Reactor Bldg. Elevation 164'-0"
A-43965 Sh. 66A	Stand-by Gas Filters & Fan Room Reactor Bldg. Elevation 164'-0"
A-43965 Sh. 66B	Stand-by Gas Filters & Fan Room Reactor Bldg. Elevation 164'-0"
A-43965 Sh. 67A	Working Floor North Reactor Bldg. Elevation 185'-0"
A-43965 Sh. 67B	Working Floor North Reactor Bldg. Elevation 185'-0"
A-43965 Sh. 68A	Working Floor South Reactor Bldg. Elevation 185'-0"
A-43965 Sh. 68B	Working Floor South Reactor Bldg. Elevation 185'-0"
A-43965 Sh. 69A	RWCU Equipment Room Reactor Bldg. Elevation 185'-0"

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43965 Sh. 69B	RWCU Equipment Room Reactor Bldg. Elevation 185'-0"
A-43965 Sh. 70A	Southwest Corridors Reactor Bldg. Elevation 185'-0"
A-43965 Sh. 70B	Southwest Corridors Reactor Bldg. Elevation 185'-0"
A-43965 Sh. 71A	Exhaust Filter Unit Room Reactor Bldg. Elevation 185'-0"
A-43965 Sh. 71B	Exhaust Filter Unit Room Reactor Bldg. Elevation 185'-0"
A-43965 Sh. 72A	Stack Monitoring Room Reactor Bldg. Elevation 203'-0"
A-43965 Sh. 72B	Stack Monitoring Room Reactor Bldg. Elevation 203'-0"
A-43965 Sh. 73A	Working Floor & Air Supply Room Reactor Bldg. Elevation 203'-0"
A-43965 Sh. 73B	Working Floor & Air Supply Room Reactor Bldg. Elevation 203'-0"
A-43965 Sh. 143A	Drywell/Torus Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 143B	Drywell/Torus Reactor Bldg. Elevation 130'-0"

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation

Unit 1 Reactor Building reactor zone areas are ducted to an exhaust system that is split into two subsystems, the accessible and inaccessible area exhaust systems. For each exhaust system, one exhaust fan is normally operating while the second is on standby. Exhaust from the working floor areas, control rod drive areas, and ASD areas is ducted to the accessible area exhaust subsystem and then routed to the Reactor Building vent plenum. Exhaust from the accessible area is continuously monitored by two radiation monitors. If a release of radioactivity is detected by the radiation monitors, the accessible area exhaust fans are automatically de-energized and the exhaust isolation dampers are automatically closed. During this period, all exhaust from the reactor zone is filtered by the inaccessible area exhaust filter trains and is ducted to the outside via the Reactor Building vent plenum.

Exhaust from the fuel pool pump and heat exchanger area, reactor water cleanup system, main steam pipe chase, residual heat removal and core spray pump rooms, high-pressure coolant injection room, and torus chamber room is ducted to the inaccessible area exhaust subsystem. Exhaust from these areas is filtered by two filter trains and is ducted to the environment via the Reactor Building vent plenum. Each filter train consists of a bank of prefilters, a bank of carbon adsorbers to minimize iodine releases, and a bank of HEPA filters to minimize particulate releases. Radiation monitors survey the bank performance with high-level annunciation in the Main Control Room. The radiation monitors are also backed up by the reactor building vent plenum isokinetic probe. Two channels of radiation monitors are installed to monitor the inaccessible area exhaust. If a release of radioactivity is detected in the inaccessible area exhaust, the supply and exhaust fans (in both the accessible and inaccessible areas) shut off, the secondary containment isolation dampers are closed, and the standby gas treatment system is started and annunciated. The standby gas treatment system filters and discharges the air to the main stack.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation
Liquid wastes from the Unit 1 Reactor Building are processed by the Unit 1 radwaste system and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train. Liquids from the floor drain sample tank, if below the technical specification limits, are discharged from the plant after dilution with the cooling tower blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Wastes from the Unit 1 Reactor Building reactor water cleanup flow glass drain, fuel pool filter-demineralizer drain, fuel pool chemical cleaning drain, laboratory drains and other drains containing chemicals are received by the chemical waste tank. These wastes are processed by filtration (after being neutralized, if required). If a decision to reuse the water has been made, then the water passes through an ion exchange unit and is sent to the condensate storage tank for reuse. However, if the sampling and analysis indicate that the radioactivity concentrations are low enough to meet the discharge criteria, the water in the sample tank is released to the discharge pipe. Hot shower wastes are treated by the Unit 1 radwaste system.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-Fire Plans to Minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the pre-fire plans to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke

and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage systems described above remain available during non-power operations.

The ventilation and monitoring systems described above remain available. Certain procedurally-controlled secondary containment alignments could allow openings between the Reactor Building and adjacent buildings, between the Reactor Building reactor zone and the Reactor Building refueling zone, and between the Reactor Building and the atmosphere (e.g., railroad airlock) in accordance with the plant Technical Requirements Manual (TRM). Both units are not open to the atmosphere at the same time, and a unit with its reactor zone open to the atmosphere is not simultaneously open to the refueling zone. The release of radioactive materials due to smoke migration is not expected as the reactor buildings are maintained under negative pressure and HVAC systems remain in operation. Operations notification and approval is required prior to breaching or realigning secondary containment. Fire brigade members will be aware of the opening and are trained to contain and/or monitor gaseous effluents prior to safe removal, and to remain in contact with Radiation Protection during this process.

Conclusion
Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Unit 2 Reactor Building

Compartment Selection and Justification Basis

The Unit 2 Reactor Building (below the refueling floor) includes unit-specific locations. The support systems for the Reactor Building are unit specific; as such, this compartment will address the Unit 2 Reactor Building for only.

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
2201	Unit 2 Drywell and Torus
2203A	Reactor Building North
2203B	Reactor Building Northeast Corner Room
2203C	Northwest Corner Room
2203F	Working Floor North
2203I	Stair Vestibule
2203K	Working Floor North
2205A	Reactor Building South
2205B	Southeast Corner Room
2205C	Southwest Corner Room
2205F	Working Floor South
2205H	Main Steam Chase
2205I	Working Floor South
2205L	RWCU Heat Exchanger Room
2205M	Cleanup Phase Separator Room
2205N	Chiller Room
2205Q	Standby Gas Filter Room
2205R	Working Floor South
2205S	Working Floor North
2205T	Heating and Ventilation Room
2205U	Northwest Corridor
2205V	Exhaust Filter and Demineralizer Room
2205W	Area North of Spent Fuel Pit
2205X	Stack Monitoring Room
2205Y	Working Floor
2205Z	HPCI Pump Room

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43965 Sh. 99A	NE RHR & Core Spray Pump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 99B	NE RHR & Core Spray Pump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 100A	RCIC Pump & Turbine Room Reactor Building El. Below 130'-0"
A-43965 Sh. 100B	RCIC Pump & Turbine Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 101A	SE RHR & Core Spray Pump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 101B	SE RHR & Core Spray Pump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 102A	CRD Pump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 102B	CRD Pump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 103A	HPCI Pump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 103B	HPCI Pump Room Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 104A	North Torus Area Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 104B	North Torus Area Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 105A	South Torus Area Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 105B	South Torus Plan Reactor Bldg. El. Below 130'-0"
A-43965 Sh. 106A	North CRD Area Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 106B	North CRD Area Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 107A	South CRD Area Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 107B	South CRD Area Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 108A	Main Steam Chase Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 108B	Main Steam Chase Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 109A	Working Floor North Reactor Bldg. Elevation 158'-0"
A-43965 Sh. 109B	Working Floor North Reactor Bldg. Elevation 158'-0"
A-43965 Sh. 110A	Working Floor & Pump Room Reactor Bldg. Elevation 158'-0"
A-43965 Sh. 110B	Working Floor & Pump Room Reactor Building Elevation 158'-0"
A-43965 Sh. 111A	RWCU HX Room Reactor Building Elevation 158'-0"
A-43965 Sh. 111B	RWCU Heat Exchanger Room Reactor Building Elevation 158'-0"
A-43965 Sh. 112A	Chiller Room Reactor Building Elevation 164'-0"
A-43965 Sh. 112B	Chiller Room Reactor Building Elevation 164'-0"
A-43965 Sh. 115A	Standby Gas Filter Room Reactor Building Elevation 185'-0"
A-43965 Sh. 115B	Standby Gas Filter Room Reactor Building Elevation 185'-0"
A-43965 Sh. 116A	Working Floor South Reactor Building Elevation 185'-0"
A-43965 Sh. 116B	Working Floor South Reactor Building Elevation 185'-0"
A-43965 Sh. 117A	Working Floor North Reactor Building Elevation 185'-0"

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43965 Sh. 117B	Working Floor North Reactor Building Elevation 185'-0"
A-43965 Sh. 118A	HVAC Room Reactor Building Elevation 185'-0"
A-43965 Sh. 118B	HVAC Room Reactor Building Elevation 185-0"
A-43965 Sh. 119A	Northwest Corridor Reactor Building Elevation 185'-0"
A-43965 Sh. 119B	Northwest Corridor Reactor Building Elevation 185'-0"
A-43965 Sh. 120A	Exhaust Filter & Demin Room Reactor Building Elevation 203'-0"
A-43965 Sh. 120B	Exhaust Filter & Demin Room Reactor Building Elevation 203'-0"
A-43965 Sh. 121A	Stack Monitoring Room Reactor Building Elevation 203'-0"
A-43965 Sh. 121B	Stack Monitoring Room Reactor Building Elevation 203'-0"
A-43965 Sh. 122A	Working Floor Reactor Building Elevation 203'-0"
A-43965 Sh. 122B	Working Floor Reactor Building Elevation 203'-0"
A-43965 Sh. 144A	Drywell/Torus Reactor Bldg. Elevation 130'-0"
A-43965 Sh. 144B	Drywell/Torus Reactor Bldg. Elevation 130'-0"

Smoke and By-Products of Combustion – Airborne Effluent Evaluation
Exhaust air in the Unit 2 Reactor Building reactor zone is collected via ducting in the inaccessible areas and routed to the exhaust system filter train. The filter train consists of a prefilter, carbon adsorber and HEPA filter. The reactor zone exhaust ventilation system is monitored by duct-mounted radiation monitors. Once the exhaust passes through the one filter train, it is routed to the Reactor Building vent plenum and then released to the environment. In the case of high radiation in the air stream, the normal supply and exhaust systems will shut down and the standby gas treatment system operation will be initiated.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation
Liquid wastes from the Unit 2 Reactor Building are processed by the Unit 2 radwaste system and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train. Liquids from the floor drain sample tank, if below the technical specification limits, are discharged from the plant after dilution with the cooling tower blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Wastes from the Unit 2 Reactor Building reactor water cleanup flow glass drain, fuel pool filter-demineralizer drain, fuel pool chemical cleaning drain, laboratory drains and other drains containing chemicals are received by the chemical waste tank. These wastes are processed by filtration (after being neutralized, if required). If a decision to reuse the water has been made, then the water passes through an ion exchange unit and is sent to the condensate storage tank for reuse. However, if the sampling and analysis indicate that the radioactivity concentrations are low enough to meet the discharge criteria, the water in the sample tank is released to the discharge pipe. Hot shower wastes are treated by the Unit 1 radwaste system.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-fire Plans to minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the pre-fire plans to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage systems described above remain available during non-power operations.

The ventilation and monitoring systems described above remain available. Certain procedurally-controlled secondary containment alignments could allow openings between the Reactor Building and adjacent buildings, between the Reactor Building reactor zone and the Reactor Building refueling zone, and between the Reactor Building and the atmosphere (e.g., railroad airlock) in accordance with the plant Technical Requirements Manual

(TRM). Both units are not open to the atmosphere at the same time, and a unit with its reactor zone open to the atmosphere is not simultaneously open to the refueling zone. The release of radioactive materials due to smoke migration is not expected as the reactor buildings are maintained under negative pressure and HVAC systems remain in operation. Operations notification and approval is required prior to breaching or realigning secondary containment. Fire brigade members will be aware of the opening and are trained to contain and/or monitor gaseous effluents prior to safe removal, and to remain in contact with Radiation Protection during this process.

Conclusion

Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Waste Gas Treatment Building

Compartment Selection and Justification Basis

The Waste Gas Treatment Building is a standalone building with its own gaseous and liquid support system configurations; as such, this compartment will address the Waste Gas Treatment Building only.

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
0601A	Waste Gas Treatment Working Floors
0601B	Unit 1 Waste Gas Charcoal Adsorber
0601C	Unit 2 Waste Gas Charcoal Adsorber

Related Pre-Fire Plans	
Pre-Fire Plan	Pre-Fire Plan Description
A-43966 Sh. 39A	Waste Gas Treatment Bldg. Elevation 106'-0"
A-43966 Sh. 39B	Waste Gas Treatment Bldg. Elevation 106'-0"
A-43966 Sh. 40A	Waste Gas Treatment Bldg. Elevation 124'-0"
A-43966 Sh. 40B	Waste Gas Treatment Bldg. Elevation 124'-0"

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation

Air from the building spaces and carbon adsorber vaults is expelled to the main stack by two 100% capacity exhaust fans serving the waste gas treatment areas for both units. One exhaust fan is normally operating while the second is on standby. Waste Gas Treatment Building air is monitored by the main stack radiation monitor and released through the main stack.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation

Liquid wastes from the Waste Gas Treatment Building are processed by the Unit 2 radwaste system and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train. Liquids from the floor drain sample tank, if below the technical specification limits, are discharged from the plant after dilution with the cooling tower

blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-Fire Plans to Minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the pre-fire plans to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage, ventilation, and monitoring systems described above remain available during non-power operations.

Conclusion
Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Unit 1 Radwaste and Radwaste Addition Buildings

Compartment Selection and Justification Basis
The Unit 1 Radwaste and Radwaste Addition Buildings include unit-specific locations. The support systems for the Unit 1 Radwaste and Radwaste Addition Buildings are unit specific; as such, this compartment will address the Unit 1 Radwaste and Radwaste Addition Buildings only.

Related Fire Areas/Zones	
Fire Area/Zone	**Fire Area/Zone Description**
1301A	West Condensate Phase Separator Room
1301B	East Condensate Phase Separator Room
1301C	Waste Sludge and Spent Resin Tank and Pump Rooms
1301D	Chemical Waste and Floor Drain Collection Tank Rooms
1301E	Waste Collector Tank Room
1301F	Waste Sludge and Sample Tank Room
1301G	Working Floor
1301H	Unit 1 Radwaste Control Room
1301I	Chemical Treatment Room
1301J	Working Floor
1301K	Radwaste Exhaust Filter Room
1301L	Hopper B Room
1301M	Southeast Radwaste Building
1301N	Hopper A Room
1301P	Working Floor
1301Q	Ventilation Room
1301R	Centrifuge Room – A
1301S	Centrifuge Room – B
1302A	Concentrated Radwaste Pump Room
1302B	Chemical Waste Neutralizer Tank Room
1302C	Chemical Waste Sample Tank Room
1302D	Concentrator Tank Room
1302E	Chemical Waste Neutralizer Pump Room and Dumbwaiter Hall
1302F	Working Floor
1302G	Radwaste Concentrate Tank Room
1302H	Floor Drain Sample Tank Room
1302I	HVAC Room

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
1302J	Working Floor
1302K	Floor Drain Demineralizer Room
1302L	Solidification Area

Related Pre-Fire Plans	
Pre-Fire Plan	Pre-Fire Plan Description
A-43965 Sh. 91A	Radwaste Building Elevation 108'-0"
A-43965 Sh. 91B	Radwaste Building Elevation 108'-0"
A-43965 Sh. 92A	Radwaste Bldg. Control Room Elevation 132'-4"
A-43965 Sh. 92B	Radwaste Bldg. Control Room Elevation 132'-4"
A-43965 Sh. 93A	Radwaste Building Elevation 132'-4"
A-43965 Sh. 93B	Radwaste Building Elevation 132'-4"
A-43965 Sh. 94A	Radwaste Building Elevation 144'-4"
A-43965 Sh. 94B	Radwaste Building Elevation 144'-4"
A-43965 Sh. 95A	Radwaste Building Elevation 156'-4"
A-43965 Sh. 95B	Radwaste Building Elevation 156'-4"
A-43965 Sh. 96A	Radwaste Building Addition Elevation 108'-0"
A-43965 Sh. 96B	Radwaste Building Addition Elevation 108'-0"
A-43965 Sh. 97A	Radwaste Building Addition Elevation 132'-4"
A-43965 Sh. 97B	Radwaste Building Addition Elevation 132'-4"
A-43965 Sh. 98A	Radwaste Building Addition Elevation 150'-4"
A-43965 Sh. 98B	Radwaste Building Addition Elevation 150'-4"

Smoke and By-Products of Combustion – Airborne Effluent Evaluation

Exhaust from different areas of the Unit 1 Radwaste and Radwaste Addition Buildings is filtered by two 50% capacity filter trains and is ducted to the outside environment via the Reactor Building vent plenum. Each system is normally supplied with fresh air by one fan in operation while a second is on standby. Each filter train consists of a bank of prefilters, a bank of carbon adsorbers, and a bank of HEPA filters to minimize the potential for radioactive particulate and radioactive iodine releases. Radiation monitors survey the bank performance and provide the Main Control Room with high-level annunciation. These radiation monitors are backed up by the Reactor Building vent plenum isokinetic probe.

The radwaste control room is ventilated by two HVAC systems; one is considered the primary, while the other is considered the backup. Both systems consist of an air handling unit with a fan and prefilters and both are tied to a common supply and return ductwork and are ducted to the outside environment via the Reactor Building vent plenum.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation
Liquid wastes from the Unit 1 Radwaste and Radwaste Addition Buildings are processed by the Unit 1 radwaste system and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train. Liquids from the floor drain sample tank, if below the technical specification limits, are discharged from the plant after dilution with the cooling tower blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-fire Plans to minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the pre-fire plans to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage, ventilation and monitoring systems described above remain available during non-power operations.

Conclusion
Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Unit 2 Radwaste Building

Compartment Selection and Justification Basis

The Unit 2 Radwaste Building includes unit-specific locations. The support systems for the Unit 2 Radwaste Building are unit specific; as such, this compartment will address the Unit 2 Radwaste Building only.

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
2301A	Condensate Sludge Mix Pump Room
2301B	Condensate Phase Separator
2301C	Oil Skimmer Room
2301D	Floor Drain and Chemical Waste Tank Room
2301E	Decontamination Solution Concentration Pump Tank Room
2301F	Chemical Waste Neutralizer Tank and Pump Room
2301G	Spent Resin Pump and Tank Room
2301H	Waste Sludge Phase Separator Room
2301I	Waste and Surge Tank Room
2301J	Dry Waste Storage Area
2301K	HVAC Room
2301L	Solidification Area
2301M	Conveyor Room
2301N	Floor Drain and Waste Collector Filter Holding Pump Area
2301P	Steam Generator Room
2301Q	Decontamination Solution Concentrate Room
2301R	Working Floor
2301S	Hopper A Room
2301T	Hopper B Room
2301U	Working Floor
2301V	Unit 2 Radwaste Control Room
2301W	Centrifuge Room A
2301X	Centrifuge Room B
2301Y	Chemical Treatment Area
2301Z	Radwaste Building Supply Ventilation Room

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43965 Sh. 137A	Radwaste Building Elevation 103'-0"
A-43965 Sh. 137B	Radwaste Building Elevation 103'-0"
A-43965 Sh. 138A	Radwaste Building Elevation 132'-0"
A-43965 Sh. 138B	Radwaste Building Elevation 132'-0"
A-43965 Sh. 139A	HVAC Room Elevation 132'-0"
A-43965 Sh. 139B	HVAC Room Elevation 132'-0"
A-43965 Sh. 140A	Radwaste Building Elevation 148'-0"
A-43965 Sh. 140B	Radwaste Building Elevation 148'-0"
A-43965 Sh. 141A	Radwaste Building Elevation 164'-0"
A-43965 Sh. 141B	Radwaste Building Elevations 164'-0" & 178'-0"
A-43965 Sh. 142A	Radwaste Control Room Elevation 164'-0"
A-43965 Sh. 142B	Radwaste Control Room Elevation 164'-0"

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation
Exhaust air from the Unit 2 Radwaste Building is ducted into the Reactor Building vent plenum. Normally, air is supplied by one fan in operation while a second is on standby. The exhaust system consists of two filter trains, each train consists of prefilters, charcoal adsorbers, and HEPA filters. Radiation monitors are provided in the exhaust duct of the radwaste building to detect high radiation. If high radiation is detected an alarm will notify the Main Control Room. The radwaste control room HVAC system is normally used to provide temperature control but still ducts to the environment via the Reactor Building vent plenum.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation
Liquid wastes from the Unit 2 Radwaste Building are processed by the Unit 2 radwaste system and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train. Liquids from the floor drain sample tank, if below the technical specification limits, are discharged from the plant after dilution with the cooling tower blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-Fire Plans to minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the pre-fire plans to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage, ventilation, and monitoring systems described above remain available during non-power operations.

Conclusion
Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Offgas Recombiner Building

Compartment Selection and Justification Basis

The Offgas Recombiner Building is a standalone building with its own gaseous and liquid support system configurations; as such, this compartment will address the Offgas Recombiner Building only.

Related Fire Areas	
Fire Area/Zone	Fire Area/Zone Description
1608A	Offgas Recombiner Working Floors
1608B	Offgas Preheater No.1
1608C	Offgas Condenser
1608D	Offgas Preheater No. 2

Related Pre-Fire Plans	
Pre-Fire Plan	Pre-Fire Plan Description
A-43966 Sh. 37A	Off Gas Recombiner Building Working Floors
A-43966 Sh. 37B	Off Gas Recombiner Building Working Floors
A-43966 Sh. 38A	Off Gas Recombiner Building Preheaters and Condenser
A-43966 Sh. 38B	Off Gas Recombiner Building Preheaters and Condenser

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation

Air from within the Off-Gas Recombiner Building is processed by a filter train consisting of a pre-filter and HEPA filter. Radiation monitors are provided in the exhaust ducts of the Off-Gas Recombiner Building ventilation system and remote switches are provided to shut down the exhaust fan of the ventilation system in case of excess radiation. Filtered air is exhausted to the atmosphere directly from the Off-Gas Recombiner Building.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation

Liquid wastes from the Off-Gas Recombiner Building are processed by the Unit 2 radwaste system and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train.

Liquids from the floor drain sample tank, if below the technical specification limits, are discharged from the plant after dilution with the cooling tower blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-Fire Plans to Minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage, ventilation, and monitoring systems described above remain available during non-power operations.

Conclusion
Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Unit 1 and 2 Refueling Floor

Compartment Selection and Justification Basis

The Unit 1 and 2 Refueling Floor includes unit-specific locations that are not separated by a physical barrier. Although the support systems for the Unit 1 and 2 Refueling Floor are unit specific, there is no physical separation between the two units within this compartment; as such, this compartment will address the Unit 1 and Unit 2 Refueling Floors together.

Related Fire Areas/Zones	
Fire Area/Zone	**Fire Area/Zone Description**
0201A	Refueling Floor
0201B	Refueling Floor

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43965 Sh. 74A	Refueling Floor Reactor Bldg. Elevation 228'-0"
A-43965 Sh. 74B	Refueling Floor Reactor Bldg. Elevation 228'-0"
A-43965 Sh. 123A	Refueling Floor Reactor Building Elevation 228'-0"
A-43965 Sh. 123B	Refueling Floor Reactor Building Elevation 228'-0"

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation

Exhaust air in the Unit 1 and 2 Reactor Building refueling zone may be drawn from either the floor area or by a sweep action from the fuel pool surface via ducting in an effort to reduce the potential for airborne radioactivity during work in the fuel pool. Normally, air is supplied by one fan in operation while a second is on standby. All exhaust air from the area is discharged to filter trains that consist of a prefilter, carbon adsorber and HEPA filter. The exhaust air is then ducted to the Reactor Building vent plenum to be released to the environment. The refueling zone exhaust system is equipped with duct-mounted radiation monitors for monitoring the exhaust air stream. If the radiation monitors detect a high-radiation level, the supply and exhaust fans in the refueling zone HVAC system will automatically be shut down, isolation dampers will close, and an alarm will be annunciated in the Main Control Room. A high-radiation condition in either refueling zone initiates simultaneous stopping of the opposite refueling and reactor zone HVAC systems, since both units refueling zone communicates with one another.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor gaseous effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated gaseous effluents within the RCA to the extent possible.

Fire Suppressant Runoff – Liquid Effluent Evaluation
Liquid wastes from the Unit 1 and 2 Reactor Buildings are processed by the Unit 1 and 2 radwaste systems, respectively, and are collected in the waste collector tank or floor drain collector tank and are then processed by filtration and ion exchange. The waste is sampled in the waste sample tank or floor drain sample tank respectively. Liquids from the waste sample tank, if satisfactory for reuse, are transferred to the condensate storage tank as makeup water. If liquids do not meet specifications for reuse, they are returned to the system for additional processing by the waste-filter demineralizer train. Liquids from the floor drain sample tank, if below the technical specifications limits, are discharged from the plant after dilution with the cooling tower blowdown. Occasionally, water that does not meet the specifications for reuse may be diluted and discharged from the plant based on capacities or unexpected occurrences.

Wastes from the Unit 1 and 2 Reactor Building reactor water cleanup flow glass drain, fuel pool filter-demineralizer drain, fuel pool chemical cleaning drain, laboratory drains and other drains containing chemicals are received by the chemical waste tank. These wastes are processed by filtration (after being neutralized, if required). If a decision to reuse the water has been made, then the water passes through an ion exchange unit and is sent to the condensate storage tank for reuse. However, if the sampling and analysis indicate that the radioactivity concentrations are low enough to meet the discharge criteria, the water in the sample tank is released to the discharge pipe. Hot shower wastes are treated by the Unit 1 radwaste system.

Provisions are in place to commence direct communication between the Fire Brigade and Radiation Protection and to contain and/or monitor liquid effluents prior to safe removal. Fire Brigade and Radiation Protection are trained to contain contaminated liquid effluents within the RCA to the extent possible.

Administrative Controls – Pre-fire Plans to minimize the Risk of Radioactive Release
Pre-fire plans will identify potentially contaminated areas, provide instructions for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Pre-fire plans will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised pre-fire plans will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the pre-fire plans to address radioactive release requirements of NFPA 805.

Fire Brigade Training to Minimize Radioactive Release
Training materials reinforce the use of pre-fire plans. Training materials will identify potentially contaminated areas, provide instruction for communication with Radiation Protection and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training materials will describe the presence and potential use of monitored ventilation and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials will meet NFPA 805 radioactive release performance criteria.

Transition Report Attachment S, Table S-3, implementation item IMP-4 is tracking revisions to the Fire Brigade training materials to address radioactive release requirements of NFPA 805.

Non-Power Operations
The drainage and monitoring systems described above remain available during non-power operations.

The ventilation and monitoring systems described above remain available. Certain procedurally-controlled secondary containment alignments could allow openings between the Reactor Building and adjacent buildings, between the Reactor Building reactor zone and the Reactor Building refueling zone, and between the Reactor Building and the atmosphere (e.g., railroad airlock) in accordance with the plant Technical Requirements Manual (TRM). Both units are not open to the atmosphere at the same time, and a unit with its reactor zone open to the atmosphere is not simultaneously open to the refueling zone. The release of radioactive materials due to smoke migration is not expected as the reactor buildings are maintained under negative pressure and HVAC systems remain in operation. Operations notification and approval is required prior to breaching or realigning secondary containment. Fire brigade members will be aware of the opening and are trained to contain and/or monitor gaseous effluents prior to safe removal, and to remain in contact with Radiation Protection during this process.

Conclusion
Based on the availability of engineered controls for both smoke and fire suppression water runoff and use of revised pre-fire plans and training materials, the approach used at HNP will be considered acceptable to meet NFPA 805 radioactive release performance criteria.

Compartment Identification – Bounded Areas

Compartment Selection and Justification Basis

The plant locations in this compartment were grouped based on their inclusion within Vendor Document S77684, which demonstrates that the potential release of contaminated effluents resulting from a fire involving radioactive contents in these areas is below 10 CFR 20 limits and satisfies the acceptance criteria of FAQ 09-0056.

Related Fire Areas/Zones	
Fire Area/Zone	**Fire Area/Zone Description**
0602	Off Gas Stack (Main Stack)
0603	Low Level Radwaste Facility
0806	Technical Support Center
1603	Unit 1 Condensate Storage Tank/Pump
1604A	Unit 1 Reactor Building Railroad Airlock - el 130 ft
1805A	Instrument Calibration Room
1805B	Respirator Decontamination Room
2603	Unit 2 Condensate Storage Tank/Pump
2604	Hot Machine Shop and Unit 2 Nitrogen Storage Tank
2607	Unit 2 Radwaste Dilution Valve Pit
N/A	ALARA Office Area
N/A	Break Room
N/A	Dosimetry Area
N/A	Simulator Building First Floor
N/A	Laundry Storage Building
N/A	Outage & Modification Building First Floor
N/A	Retrofit Building Elevation 130'-0"
N/A	Sealand Storage Facility
N/A	Warehouse No. 3
N/A	Warehouse No. 6

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43966 Sh. 28A	Break Room Service Building
A-43966 Sh. 28B	Break Room Service Building
A-43966 Sh. 30A	HP Checkpoint Area Service Building

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43966 Sh. 30B	HP Checkpoint Area Service Building
A-43966 Sh. 31A	ALARA Office Area Service Building
A-43966 Sh. 31B	ALARA Office Area Service Building
A-43966 Sh. 36A	T.S.C./Mechanical Room Service Building Annex
A-43966 Sh. 36B	T.S.C./Mechanical Room Service Building Annex
A-43966 Sh. 41A	Railroad Airlock El. 130'-0"
A-43966 Sh. 41B	Railroad Airlock El. 130'-0"
A-43966 Sh. 42A	Hot Machine Shop and Nitrogen Storage Tank
A-43966 Sh. 42B	Hot Machine Shop and Nitrogen Storage Tank
A-43966 Sh. 43A	Condensate Storage Tank
A-43966 Sh. 43B	Condensate Storage Tank
A-43966 Sh. 44A	Condensate Storage Tank
A-43966 Sh. 44B	Condensate Storage Tank
A-43966 Sh. 45A	Main Stack
A-43966 Sh. 45B	Main Stack
A-43966 Sh. 59A	Radwaste Dilution Wtr Valve Pit
A-43966 Sh. 59B	Radwaste Dilution Wtr Valve Pit
A-43966 Sh. 60A	Retrofit Building Elevation 130'-0"
A-43966 Sh. 60B	Retrofit Building Elevation 130'-0"
A-43966 Sh. 64A	Decontamination and Calibration Building
A-43966 Sh. 64B	Decontamination and Calibration Building
A-43966 Sh. 69A	Laundry Storage Building
A-43966 Sh. 69B	Laundry Storage Building
A-43966 Sh. 71A	Warehouse No. 6
A-43966 Sh. 71B	Warehouse No. 6
A-43966 Sh. 85A	Simulator Building First Floor
A-43966 Sh. 85B	Simulator Building First Floor
A-43966 Sh. 91A	Inprocessing Building First Floor
A-43966 Sh. 91B	Inprocessing Building First Floor
A-43966 Sh. 99A	Warehouse Number 3
A-43966 Sh. 99B	Warehouse Number 3
A-43966 Sh. 105A	Low Level Radwaste Building
A-43966 Sh. 105B	Low Level Radwaste Building

Related Pre-Fire Plans	
Pre-Fire Plan	Pre-Fire Plan Description
A-43966 Sh. 129A	Sealand Storage Shelter
A-43966 Sh. 129B	Sealand Storage Shelter

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation
The potential release of contaminated gaseous effluents resulting from a fire involving radioactive contents in these areas is bounded by Vendor Document S77684. This calculation demonstrates that releases are below 10 CFR 20 limits and satisfies the acceptance criteria of FAQ 09-0056.

Fire Suppressant Runoff – Liquid Effluent Evaluation
The potential release of contaminated liquid effluents resulting from a fire involving radioactive contents in these areas is bounded by Vendor Document S77684. This calculation demonstrates that releases are below 10 CFR 20 limits and satisfies the acceptance criteria of FAQ 09-0056.

Administrative Controls – Pre-Fire Plans to Minimize the Risk of Radioactive Release
Vendor Document S77684 concludes that a fire in any one of these areas will not jeopardize the ability to meet the NFPA 805 radioactive release performance criteria. Therefore, inclusion of administrative controls to minimize the risk of radioactive release in the pre-fire plans for these areas is not required.

Fire Brigade Training to Minimize Radioactive Release
Vendor Document S77684 concludes that a fire in any one of these areas will not jeopardize the ability to meet the NFPA 805 radioactive release performance criteria. Therefore, fire brigade training to minimize the risk of radioactive release in these areas is not required.

Non-Power Operations
The conclusions of Vendor Document S77684 apply to all modes of plant operation, including non-power operations.

Conclusion
The potential release of contaminated effluents resulting from a fire involving radioactive contents in these areas is bounded by Vendor Document S77684. This calculation demonstrates that releases are below 10 CFR 20 limits and satisfies the acceptance criteria of FAQ 09-0056.

Compartment Identification – Screened-Out Areas

Compartment Selection and Justification Basis
The following areas have been screened from this review as a radiological release due to firefighting activities in these areas is not postulated.

Related Fire Areas/Zones	
Fire Area/Zone	**Fire Area/Zone Description**
0002A	Control Building Stairwell
0014E	HP Receiving Area
0014J	HP Office
0014L	HVAC Room
0014N	Ladies' Rest Room
0024A	Cable Spread Room
0024B	Computer Room
0024C	Main Control Room
0024D	Main Control Room Entryway
0025	CO2 Tank Room
0028	LPCI Inverter Room
0040	Vertical Cable Chase
0101E	East Main Control Room Entry Way
0101F	Main Control Room Break Area
0101G	Chart Storage Room and Hallway
0101H	Shift Clerk/Operations Supervisor's Office
0401	Diesel Building Hallway
0501	Intake Structure
0604	Division I, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (East of DG Building)
0605	Division II, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (East of DG Building)
0606	Division I, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (West of DG Building)
0607	Division II, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (West of DG Building)
0608	Diesel Generator 1B, Unit 1 and 2 Pullboxes and Associated Ductbanks (West of DG Building)
0609	Outside Cable Buses West of DG Building (Excluding Duct Runs in the Existing Switchyard FAs – 1606)
0610	Division I, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (North of DG Building)
0611	Miscellaneous Underground Yard Pullboxes (Not covered by any other fire areas)
0612	Division II, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (North of DG Building)
0702A	Fire Pump House – Water Pump Room
0702B	Fire Pump House – West Fire Pump Room

Related Fire Areas/Zones	
Fire Area/Zone	**Fire Area/Zone Description**
0703	Fire Pump House – Central Fire Pump Room
0704	East Fire Pump Room
0801	Unit 1 and Unit 2 Main 500-kV Switchyard
0802	Unit 1 and Unit 2 500-kV Switchyard
0803	Main Meteorological Tower
0804	Backup Meteorological Tower
0805	Chlorine Building
0808	Central Alarm Station (CAS) Building
0809	Hydrogen and Oxygen Storage Facility
0811	Service Building Fan Room
1013	RPS MG Set Room
1015	Unit 1 Annunciator Room
1016	West 600-V Switchgear Room
1017	East 600-V Switchgear Room
1018	West DC Switchgear Room
1019	Transformer Room
1020	East DC Switchgear Room
1023	Oil Conditioner Room
1401	Day Tank Room 1C
1402	Battery Room 1C
1403	Diesel Generator 1C
1404	Switchgear Room 1G
1405	Day Tank Room 1B
1406	Battery Room 1B
1407	Diesel Generator Room 1B
1408	Switchgear Room 1F
1409	Day Tank Room 1A
1410	Battery Room 1A
1411	Diesel Generator Room 1A
1412	Switchgear Room 1E
1601	Unit 1 Service Water Valve Pit 1A
1602	Unit 1 Service Water Valve Pit 1B
1604B	Unit 1 Nitrogen Storage Tank

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
1605	Unit 1 Circulating Water Pump Pit
1606	Unit 1 Transformer Yard
1610	Diesel Fuel Oil Storage Tank 1A
1611	Diesel Fuel Oil Storage Tank 1B
1612	Diesel Fuel Oil Storage Tank 1C
1801	Unit 1 "A" Cooling Tower Switchgear Building
1802	Unit 1 "B" Cooling Tower Switchgear Building
1803	Unit 1 "C" Cooling Tower Switchgear Building
1804	Unit 1 Turbine Building Back Entrance (Frisker Building)
1806	Unit 1 HVAC Chiller Building
1807	Unit 1 Helper Cooling Tower
2013	RPS MG Set Room
2014	Unit 2 Switchgear Access Hallway
2015	Unit 2 Annunciator Room
2016	West 600V Switchgear Room
2017	East 600-V Switchgear Room
2018	West DC Switchgear Room
2019	Transformer Room
2020	East DC Switchgear Room
2021	Unit 2 Switchgear Hallway Enclosure
2023	Oil Conditioner Room
2401	Day Tank Room 2A
2402	Battery Room 2A
2403	Diesel Generator Room 2A
2404	Switchgear Room 2E
2405	Day Tank Room 2C
2406	Battery Room 2C
2407	Diesel Generator Room 2C
2408	Switchgear Room 2F
2409	Switchgear Room 2G
2601	Unit 2 Service Water Valve Pit 2A
2602	Unit 2 Service Water Valve Pit 2B
2605	Unit 2 Circulating Water Pump Pit

Related Fire Areas/Zones	
Fire Area/Zone	**Fire Area/Zone Description**
2606	Unit 2 Transformer Yard
2608	Unit 2 Transformers in the Unit 1 Transformer Yard
2610	Diesel Fuel Oil Storage Tank 2A
2612	Diesel Fuel Oil Storage Tank 2C
2801	Unit 2 "A" Cooling Tower and Switchgear Building
2802	Unit 2 "B" Cooling Tower and Switchgear Building
2803	Unit 2 "C" Cooling Tower and Switchgear Building
2804	Unit 2 Turbine Building Back Entrance (Frisker Building)
2806	Unit 2 HVAC Chiller Building
2807	Unit 2 Helper Cooling Tower
N/A	Atlanta Steel Building
N/A	Auxiliary Boilers
N/A	Building No. 1C (Welding and HP Training)
N/A	Cable Storage Building
N/A	Circulating Water System Chlorination Facility
N/A	Compressed Gas Storage Building
N/A	Computer Building
N/A	Cooling Tower Maintenance Support Building
N/A	Deep Well Pumps
N/A	Discharge Structure
N/A	Service Building and Annex
N/A	Document Storage Building
N/A	Edwin I. Hatch Operations Training Facility
N/A	Fabrication Shop
N/A	Fire Protection Equipment Building
N/A	Fire Protection Tanks
N/A	Gas and Diesel Fuel Pumps
N/A	FLEX Equipment Storage Building
N/A	Greenhouse
N/A	Health Physics Dress Out Building
N/A	High Voltage Switch House
N/A	Hydrogen Gas Storage Area
N/A	Investment Recovery Area

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
N/A	Maintenance Support Building
N/A	Microwave Tower
N/A	Outage & Modification Building Second Floor
N/A	Outage Support Warehouse
N/A	PACE (Environmental) Building
N/A	PESB (Security) Building
N/A	Pumphouse Behind Lower Level Radwaste
N/A	Quality Control Area
N/A	Radwaste Dilution Water Valve Pit
N/A	Rifle Range Building
N/A	Security Annex Training Building
N/A	Security Post
N/A	Sewage Treatment
N/A	Simulator Building Second Floor
N/A	Site Projects Building
N/A	Skills Training Building
N/A	Staff Development Building (Cork and Hook Building)
N/A	Start-up Boiler Building
N/A	Supply Shed
N/A	Support Building
N/A	Tire Shop
N/A	Tractor Shed
N/A	Training Flow Loop Building
N/A	Turbine Rotor Storage Building
N/A	Unit 1 & 2 PSW Chemical Treatment
N/A	Unit 1 Dechlorination Treatment
N/A	Unit 1 Overflow Weir
N/A	Unit 2 Dechlorination Treatment
N/A	Unit 2 Overflow Weir
N/A	Vehicular Maintenance Garage
N/A	Hatch Energy Education Center
N/A	Warehouse No. 1
N/A	Warehouse No. 2

Related Fire Areas/Zones	
Fire Area/Zone	Fire Area/Zone Description
N/A	Warehouse No. 4 & 5
N/A	Waste Oil Facility
N/A	Water Treatment Building
N/A	Weld Shop
N/A	Weld Test Building
N/A	Williams Building
N/A	Yard

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation
Not applicable, areas are screened-out.

Fire Suppressant Runoff – Liquid Effluent Evaluation
Not applicable, areas are screened-out.

Administrative Controls – Pre-Fire Plans to Minimize the Risk of Radioactive Release
Not applicable, areas are screened-out.

Fire Brigade Training to Minimize Radioactive Release
Not applicable, areas are screened-out.

Non-Power Operations
Not applicable, areas are screened-out.

Conclusion
Not applicable, areas are screened-out.

Compartment Identification – Areas excluded from the scope of 10 CFR 50.48(c)

Compartment Selection and Justification Basis

Fire Area 0810, the Spent Fuel Storage Facility, is excluded from the NFPA 805 fire protection licensing basis as it is a separately-licensed facility and is not within the scope of 10 CFR 50.48(c) (NFPA 805) and therefore is not included in this review.

Related Fire Areas	
Fire Area/Zone	**Fire Area/Zone Description**
0810	Spent Fuel Storage Facility

Related Pre-Fire Plans	
Pre-Fire Plan	**Pre-Fire Plan Description**
A-43966 Sh. 113A	Spent Fuel Storage Facility
A-43966 Sh. 113B	Spent Fuel Storage Facility

Smoke and Byproducts of Combustion – Airborne Effluent Evaluation

Not applicable, area is not reviewed.

Fire Suppressant Runoff – Liquid Effluent Evaluation

Not applicable, area is not reviewed.

Administrative Controls – Pre-Fire Plans to Minimize the Risk of Radioactive Release

Not applicable, area is not reviewed.

Fire Brigade Training to Minimize Radioactive Release

Not applicable, area is not reviewed.

Non-Power Operations

Not applicable, area is not reviewed.

Conclusion

Fire Area 0810, the Spent Fuel Storage Facility, is excluded from the NFPA 805 fire protection licensing basis as it is a separately-licensed facility and is not within the scope of 10 CFR 50.48(c) (NFPA 805) and therefore is not included in this review.

F. Fire-Induced Multiple Spurious Operations Resolution

6 Pages Attached

MSO Process Summary

The following provides the guidance from FAQ 07-0038, Revision 3, along with the process and 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 2, Chapter 3)
- Generic lists of MSOs (e.g., from Owners Groups and/or later versions of NEI 00-01, if endorsed by NRC for use in assessing MSOs)
- Self-assessment results (e.g., NEI 04-06 assessments performed to addressed RIS 2004-03)
- PRA insights (e.g., NEI 00-01 Revision 2, Appendix F)
- Operating Experience (e.g., licensee event reports, NRC Inspection Findings, etc.)

Results of Step 1:

The following information sources were used to identify the potential HNP MSOs of concern:

- Piping and Instrumentation Diagrams (P & IDs)
- Procedures
- Safe Shutdown Analysis
- PRA insights
- Internal Events PRA
- BWROG generic list of MSOs
- Operating Experience

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.

The documentation of the results of the expert panel should include how the expert panel was conducted including the members of the expert panel, their experience, education, and areas of expertise. The documentation should include the list of MSOs reviewed as well as the source for each MSO. This documentation should provide a list the MSOs that were included in the PRA and a separate list of MSOs that were not kept for further analysis (and the reasons for rejecting these MSOs for further analysis).

Describe the expert panel process (e.g., when it was held, what training was provided to the panel members, what analyses were reviewed to identify MSOs, how was consensus achieved on which MSOs to keep and any dispute resolution process criteria used in decision process, etc.).

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.

Results of Step 2:

The 2009 Expert Panel consisted of a two-day meeting with representatives from SNC (and contractors) with experience in fire protection, post-fire safe shutdown, circuit analysis, system engineering, plant operations and PRA. The panel conducted document reviews, and held discussions on potential fire-induced spurious operations that could potentially impact plant safety. Documents that were used as guidance included:

- Boiling Water Reactor Owners Group (BWROG) Working MSO list [file named *BWR Generic MSO List update rev d (file dated 10/20/09)*, obtained from BWROG representative. This list was used as input by the BWROG to NEI 00-01, Rev. 2, Appendix G, submitted to the NRC on 6/5/09]
- NFPA 805 FAQ 07-0038, Lessons Learned on Multiple Spurious Operations, Revision 1 and NRC Comments
- NEI 00-01, Guidance for Post-Fire Safe-Shutdown Circuit Analysis, Rev. 2, dated 6/5/09 [ML091770265]

The 2012 supplemental Expert Panel consisted of a one-day phone conference meeting with representatives from SNC (and contractors) with experience in fire protection, post-fire safe shutdown, circuit analysis, system engineering, PRA, and plant operations. The panel conducted document reviews, and held discussions focused on the new and clarified BWR MSO scenarios. Documents that were used as guidance included:

- Boiling Water Reactor Owners Group (BWROG) Generic MSO list (Table G-1 of Appendix G to NEI 00-01, Rev. 3)
- NFPA 805 FAQ 07-0038, Lessons Learned on Multiple Spurious Operations, Revision 1 and NRC Comments
- NEI 00-01, *Guidance for Post-Fire Safe-Shutdown Circuit Analysis*, Rev. 2, dated 6/09 and Rev. 3, dated 10/2011
- Regulatory Guide 1.189, *Fire Protection for Nuclear Power Plants*, Rev. 2, dated 11/2/2009

Training was conducted in the form of an introductory overview and slide presentation. Topics discussed included:

- Purpose and scope of the safe shutdown analysis
- PRA overview
- Overview training on the MSO issue, including
    - Background on Fire-Induced MSOs
    - Format and Status of the Hatch SSA and Fire PRA

Detailed discussion on the types of circuit failures was not held since the focus of the panel was on system and component level failures. Key points included:

- The proposed scenarios should not have presupposed limits on the number of fire-induced hot shorts or spurious operations (e.g., do not assume only one or two, one at a time, etc.)
- The focus would not be on individual fire area locations, and rather specific components/systems that are susceptible to the particular MSO scenario. This allows the Expert Panel to provide the necessary guidance to facilitate follow-on scenario vulnerability analysis in accordance with the criteria of NEI 00-01. The training discussion and examples in NEI 00-01, Revision 2 were used as input, as well as lessons learned from performance of other plant Expert Panels. focus system/component approach, in order to allow the analysis following the Expert

The Expert Panel started with a review of the BWROG generic scenarios. The BWROG Generic MSO List includes scenarios related to the following functions:

- Reactivity Control
- RCS Inventory Control (Makeup)
- RCS Pressure Control
- Decay Heat Removal
- Support Functions

By using the BWROG Generic MSO List as guidance, a step-by-step discussion was held, typically by reviewing P&IDs and simplified system training diagrams, postulating scenarios, discussing the potential consequences and likelihood, discussing operator response, and recommending additional courses of action. Key considerations, in addition to consequences were:

- Whether the scenario of concern was currently modeled in the Hatch SSA
- Whether procedures addressed the potential scenarios of concern
- Additional analyses or justification that may be necessary to document exclusion of a particular scenario

In addition to the item-by-item review of the BWROG Generic MSO list, an additional "brainstorming" review was conducted by the Expert Panel in order to look for additional plant specific scenarios and general system pinch points that may not have existed specifically on the generic list. The additional reviews were conducted on the following key systems:

- Plant Service Water
- RHR Service Water
- RCIC System
- HPCI System
- Core Spray
- Main Steam
- CRD
- RWCU
- RHR – LPCI Mode
- RHR – Torus Cooling Mode

- RHR – Shutdown Cooling Mode
- ASD Cooling Mode
- Drywell Pneumatics
- Condensate and Feedwater
- Electrical System

The Expert Panel also discussed the potential for individual scenarios that, if combined, would be more severe, or warrant inclusion as a separate MSO scenario. The panel did not identify any specific vulnerabilities.

The MSO expert panel for HNP was originally convened in October, 2009 using the guidance in NEI 00-01, Revision 2. In October 2011, Revision 3 to NEI 00-01 was issued. This revision contains updates to the BWR generic MSO list, including nine new scenarios and many clarifications to existing scenarios. SNC updated the Hatch MSO analysis to address the most current generic list and also to incorporate additional operating experience for MSO identification and resolution. To ensure the integrity of the MSO Expert Panel process, a supplemental Panel was convened on October 17, 2012 to review the new and clarified scenarios, as well as additional implementing considerations from the original 2009 Panel.

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, including associating that routing with fire areas, fire zones and/or Fire PRA physical analysis units, as applicable.

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 conditions associated with MSOs do not require 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. In other instances, the update of the PRA may not be warranted if the contribution is negligible. 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.

Results of Step 3:

The NSCA and Fire PRA analyses were updated in the Transition Process to reflect the treatment of applicable MSO scenarios. This included the identification of equipment, identification of cables, and the routing of cables by plant locations. The MSOs were examined for applicability to the Fire PRA and any differences or adjustments to the

expert panel conclusions are discussed in Appendix A to Calculation H-RIE-FIREPRA-U00-005. The HNP Results are documented in:

- Calculation SENH-17-002, Circuit Analysis for Fire Safety Analysis – FDM Output
- SENH-15-009, "Nuclear Safety Capability Assessment Report"
- Calculation H-RIE-FIREPRA-U00-005, *Hatch Fire PRA Task 5 Fire Induced Risk Model*

Step 4 – Evaluate for NFPA 805 Compliance

The MSO combinations included in the NSCA should be evaluated with respect to compliance with the deterministic requirements of NFPA 805, as discussed in Section 4.2.3 of NFPA 805. For those situations in which the MSO combination does not meet the deterministic requirements of NFPA 805 (VFDR), the issue with the components and associated cables should be mitigated by other means (e.g., performance-based approach per Section 4.2.4 of NFPA 805, plant modification, etc.).

The performance-based approach may include the use of feasible and reliable recovery actions. The use of recovery actions to demonstrate the availability of a success path for the nuclear safety performance criteria requires that the additional risk presented by the use of these recovery actions be evaluated (NFPA 805 Section 4.2.4).

Results of Step 4:

The MSO combination components of concern were also evaluated as part of the HNP NSCA. 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 fire risk evaluations. The process and results for Fire Risk Evaluations are summarized in Section 4.5 of the Transition Report.

Step 5 – Document Results

The results of the process should be documented. The results should provide a detailed description of the MSO identification, analysis, disposition, and evaluation results (e.g., references used to identify MSOs; the composition of the expert panel, the expert panel process, and the results of the expert panel process; disposition and evaluation results for each MSO, etc.). 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.

Results of Step 5:

The HNP results are documented in:

- Calculation SENH-12-002, *Expert Panel for Addressing Multiple Spurious Operations*
- SENH-15-009, "Nuclear Safety Capability Assessment Report"
- Calculation H-RIE-FIREPRA-U00-005, *Hatch Fire PRA Task 5 Fire Induced Risk Model*

Summary of Results and Risk Insights

As part of Step 4 of the process outlined above, MSO combinations were reviewed for their impact on deterministic compliance (i.e., fire area by fire area reviews) to identify fires that could result in the potential MSO combinations. As part of the process, VFDRs were created where the deterministic requirements of NFPA 805 Section 4.2.3 were not met. These VFDRs were addressed by demonstrating compliance with the performance-based approach of Section 4.2.4 of NFPA 805 (See Section 4.5 and Attachment C of this document).

The spurious operations reviewed as part of the process included components that were part of the original HNP 10 CFR 50 Appendix R post-fire safe shutdown analysis, as well as components and interactions that were added as the industry issue evolved. Changes to industry accepted circuit failure modes were also considered.

Spurious operations, both single and multiple, have a tangible impact on fire risk and are included in the FPRA model. Fire-induced spurious operations can lead to initiating events (e.g., SRV[s] spuriously open) and can also affect mitigation of initiators (e.g., RCIC isolation). Given the potential significance of fire-induced MSOs, the expert panel systematically searched for and identified MSO failures not already captured by the internal events PRA model, which was used as the starting point for the FPRA.

Fire-induced MSOs are included in the FPRA model, and their associated risk is included in the quantification of each fire scenario, the total plant fire risk, and evaluation of each VFDR. The VFDRs are identified in Attachment C, Table B-3 and a summary of the FPRA results is provided in Attachment W.

G. Recovery Actions Transition

25 Pages Attached

Attachment G is redacted in its entirety.

H. NFPA 805 Frequently Asked Question Summary Table

2 Pages Attached

Note: The NFPA 805 FAQ process will continue through the transition of non-pilot NFPA 805 plants. Final closure of the FAQs will occur when RG 1.205 is revised to endorse a new revision of NEI 04-02 that incorporates the outstanding FAQs.

This table includes the approved FAQs that have not been incorporated into the current endorsed revision of NEI 04-02 and utilized in this submittal:

Table H-1 - NEI 04-02 FAQs Utilized in LAR Submittal

No.	Rev.	Title	FAQ Ref.	Closure Memo
06-0008	9	NFPA 805 Fire Protection Engineering Evaluations	ML090560170	ML073380976
06-0022	3	Acceptable Electrical Cable Construction Tests	ML090830220	ML091240278
07-0030	5	Establishing Recovery Actions	ML103090602	ML110070485
07-0032	2	Clarification of 10 CFR 50.48(c), 10 CFR 50.48(a) and GDC 3 clarification	ML081300697	ML081400292
07-0035	2	Bus Duct Counting Guidance for High Energy Arcing Faults	ML091610189	ML091620572
07-0038	3	Lessons learned on Multiple Spurious Operations	ML103090608	ML110140242
07-0039	2	Lessons Learned - NEI B-2 Table	ML091420138	ML091320068
07-0040	5	Non-Power Operations Clarification	ML17331B109	ML17331B108
08-0042	0	Fire Propagation from Electrical Cabinets	ML080230438 ML091460350	ML092110537
08-0043	1	Electrical Cabinet Fire Location	ML083540152 ML091470266	ML092120448
08-0044	0	Large Oil Fires	ML081200099 ML091540179	ML092110516
08-0049	0	Cable Fires	ML081200309 ML091470242	ML092100274
08-0052	0	Transient Fire Growth Rate and Control Room Non-Suppression	ML081500500 ML091590505	ML092120501
08-0054	1	Demonstrating Compliance with Chapter 4 of NFPA 805	ML103510379	ML15016A280
09-0056	2	Radioactive Release Transition	ML102810600	ML102920405
09-0057	3	New Shutdown Strategy	ML100330863	ML100960568
10-0059	5	NFPA 805 Monitoring	ML120410589	ML120750108
12-0062	1	UFSAR Content	ML121430035	ML121980557
12-0063	1	Fire Brigade Make-Up	ML121670141	ML121980572

Table H-1 - NEI 04-02 FAQs Utilized in LAR Submittal

No.	Rev.	Title	FAQ Ref.	Closure Memo
12-0064	1	Hot work/transient fire frequency: influence factors	ML122550050	ML12346A488
12-0067	1	Transformer Oil Collection Drain Basin Inspections	ML13035A039	ML13037A425
13-0004	1	Sensitive Electronics	ML13182A708	ML13322A085
13-0005	5	Cable Fires Special Cases: Self Ignited and Caused by Welding and Cutting	ML13319B181	ML13319B181
13-0006	0	Modeling Junction Box Scenarios in a Fire PRA	ML13178A341	ML13331B213
13-0069	3	"Sufficient Training and Knowledge" of Fire Brigade	ML13035A197	ML14210A144
14-0008	1	Main Control Board Treatment	ML14167A296	ML14190B307
14-0009	1	Treatment of Well Sealed MCC Electrical Panels Greater than 440V	ML15118A810	ML15114A441
14-0070	0	Use of Non-Fire Treated Wood	ML15335A159	ML15336A556

I. Definition of Power Block

2 Page Attached

Each structure within the HNP owner-controlled area was assessed against the criteria provided in NFPA 805 Section 1.5 to determine whether it meets the definition of power block. Structures with circuits included in the Nuclear Safety Compliance Assessment are included in the definition of power block. Areas which may contain radioactive materials which can be released to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) are included within the scope of the power block definition.

For the purposes of establishing the structures included in the Fire Protection program in accordance with 10 CFR 50.48(c) and NFPA 805, plant structures listed in the following table are considered to be part of the power block.

Table I-1 – Power Block Definition

Power Block Structures	Fire Area(s)
U1 Circulating Water Pump Pit	1605
U2 Circulating Water Pump Pit	2605
U1 Condensate Storage Tank	1603
U2 Condensate Storage Tank	2603
Control Building (U1 & U2)	0001, 0002, 0007, 0014, 0024, 0025, 0028, 0031, 0040, 1003, 1004, 1005, 1006, 1008, 1009, 1010, 1013, 1015, 1016, 1017, 1018, 1019, 1020, 1023, 2003, 2004, 2005, 2006, 2008, 2009, 2010, 2013, 2014, 2015, 2016, 2017, 2018, 2019, 2020, 2021, 2023
U1 Cooling Towers, Cooling Tower Switchgear Buildings, and Cooling Tower Electrical Building	1801, 1802, 1803
U2 Cooling Towers, Cooling Tower Switchgear Buildings, and Cooling Tower Electrical Building	2801, 2802, 2803
Diesel Fuel Oil Storage Tanks	1610, 1611, 1612, 2610, 2612
Diesel Generator Building	0401, 1401, 1402, 1403, 1404, 1405, 1406, 1407, 1408, 1409, 1410, 1411, 1412, 2401, 2402, 2403, 2404, 2405, 2406, 2407, 2408, 2409
Diesel Generator 1B, Unit 1 and 2 Pullboxes and Associated Ductbanks (West of DG Building)	0608
Division I, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (East of DG Building)	0604
Division I, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (North of DG Building)	0610
Division I, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (West of DG Building)	0606
Division II, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (East of DG Building)	0605
Division II, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (North of DG Building)	0612

Table I-1 – Power Block Definition

Power Block Structures	Fire Area(s)
Division II, Unit 1 and 2 Underground Pullboxes and Associated Ductbanks (West of DG Building)	0607
Fire Pump House	0702, 0703, 0704
Intake Structure	0501
Miscellaneous Underground Yard Pullboxes	0611
Outside Cable Buses West of DG Building (Excluding Duct Runs in the Existing Switchyard Fire Areas – 1606/2608)	0609
U1 Radwaste Building	1301
U1 Radwaste Addition Building	1302
U2 Radwaste Building	2301
Off Gas Recombiner Building	1608
Reactor Building (U1 & U2)	0201, 1201, 1203, 1205, 1609, 2201, 2203, 2205
U1 Reactor Building Railroad Airlock and Nitrogen Storage Tank	1604
Hot Machine Shop and Unit 2 Nitrogen Storage Tank	2604
U1 Service Water Valve Pits	1601, 1602
U2 Service Water Valve Pits	2601, 2602
U1 and U2 Main 500 kV Switchyard High Voltage Switchyard and Switch House	0801
U1 Transformer Yard	1606, 2608
U2 Transformer Yard	2606
Turbine Building (U1 & U2)	0101, 1101, 1102, 1103, 1104, 1105, 2101, 2102, 2103, 2104
Waste Gas Treatment Building	0601
Off Gas Stack (Main Stack)	0602
Service Building (Fan Room and Tunnel Only)	0811, 2101
U1 HVAC Chiller Building	1806
U2 HVAC Chiller Building	2806
Fire Protection Tanks	Yard

J. Fire Modeling V&V

11 Pages Attached

Fire modeling tools are used in the HNP NFPA 805 transition process in support of the Fire PRA. The fire models listed in Table J-1 were used within the Fire PRA to assess the extent of fire generated conditions for the different fire scenarios postulated in the Fire PRA for quantification of CDF, LERF, ΔCDF and ΔLERF. Table J-1 includes the model (calculation) identification, the purpose of the model application, technical references for the model, and the validation work available for it.

The models identified in Table J-1 are shown to have been appropriately applied within the range of their applicability or when used outside that range, basis for the use has been provided.

Table J-1 V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Flame Height (Method of Heskestad)	Calculates the vertical extension of the flame region of a fire.	Technical Reference: • NUREG-1805, Chapter 3, 2004 • NUREG-1824, Volume 3, 2007 • NUREG-1934, Chapter 2, 2012 • Society of Fire Protection Engineers (SFPE) Handbook of Fire Protection Engineering, 5th Edition, Chapter 13, Heskestad, 2016	• The correlation is contained in NUREG-1805, for which V&V is documented in NUREG-1824. • The correlation is documented in an authoritative publication of the "SFPE Handbook of Fire Protection Engineering." • The correlation has been applied within its limits of applicability and the validated range reported in NUREG-1824 or, if applied outside the validated range, the model has been justified as acceptable, either by qualitative analysis, or by quantitative sensitivity analysis. The methodology for justifying application of the fire model outside the range is in accordance with methods documented in NUREG-1934. The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2 • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2, Attachment A, Fire Modeling Workbook Methodology for HNP

Table J-1 V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Plume Centerline Temperature (Method of Heskestad)	Calculates the vertical separation distance, based on temperature, to a target in order to determine the vertical extent of the zone of influence (ZOI)	Technical Reference: • NUREG-1805, Chapter 9, 2004 • NUREG-1824, Volume 3, 2007 • NUREG-1934, Chapter 2, 2012 • SFPE Handbook of Fire Protection Engineering, 5th Edition, Chapter 13, Heskestad, 2016 • NUREG/CR-6850, Volume 2, Appendix H, 2005	• The correlation is contained in NUREG-1805, for which V&V is documented in NUREG-1824. • The correlation is documented in an authoritative publication of the "SFPE Handbook of Fire Protection Engineering." • The correlation has been applied within its limits of applicability and the validated range reported in NUREG-1824 or, if applied outside the validated range, the model has been justified as acceptable, either by qualitative analysis, or by quantitative sensitivity analysis. The methodology for justifying application of the fire model outside the range is in accordance with methods documented in NUREG-1934. The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2 • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2, Attachment A, Fire Modeling Workbook Methodology for HNP
Radiant Heat Flux (NUREG-1805 Solid Flame II (Method of Shokri and Beyler)	Calculates the horizontal separation distance, based on heat flux, to a target in order to determine the horizontal extent of the ZOI.	Technical Reference: • NUREG-1805, Chapter 5. 2004 • NUREG-1824, Volume 3, 2007 • NUREG-1934, Chapter 2, 2012 • SFPE Handbook of Fire Protection Engineering, 5th Edition, Chapter 66, Beyler, C., 2016 • NUREG/CR-6850, Volume 2, Appendix H, 2005	• The correlation is contained in NUREG-1805, for which V&V is documented in NUREG-1824. • The correlation is documented in an authoritative publication of the "SFPE Handbook of Fire Protection Engineering." • The correlation has been applied within its limits of applicability and the validated range reported in NUREG-1824 or, if applied outside the validated range, the model has been justified as acceptable, either by qualitative analysis, or by quantitative sensitivity analysis. The methodology for justifying application of the fire model outside the range is in accordance with methods documented in NUREG-1934. The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2 • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2, Attachment A, Fire Modeling Workbook Methodology for HNP

Table J-1 V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Ceiling Jet Temperature (Method of Alpert)	Calculates the horizontal separation distance, based on temperature at the ceiling of a room, to a target in order to determine the horizontal extent of the ZOI.	Technical Reference: • FIVE-Rev1, Referenced by EPRI Report 1002981, 2002 • NUREG-1824, Volume 4, 2007 • NUREG-1934, Chapter 2, 2012 • SFPE Handbook of Fire Protection Engineering, 5th Edition, Chapter 14, Alpert, R., 2016 • NUREG/CR-6850, Volume 2, Appendix H, 2005	• The correlation is used in the FIVE-Rev1 fire model, for which V&V is documented in NUREG-1824. • The correlation is documented in an authoritative publication of the "SFPE Handbook of Fire Protection Engineering." • The correlation has been applied within its limits of applicability and the validated range reported in NUREG-1824 or, if applied outside the validated range, the model has been justified as acceptable, either by qualitative analysis, or by quantitative sensitivity analysis. The methodology for justifying application of the fire model outside the range is in accordance with methods documented in NUREG-1934. The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2. • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2, Attachment A, Fire Modeling Workbook Methodology for HNP

Table J-1 V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
HGL Calculations using Consolidated Model of Fire Growth and Smoke Transport (CFAST, Version 6.1.1 And Version 7.2.1)	Calculates the upper and lower gas layer temperature, interface height, and soot concentration. Evaluates the time at which control room abandonment is necessary based on smoke obscuration and average HGL temperature.	Technical Reference: • NIST Special Publication 1086, CFAST Version 6 • NIST-TN-1889v3, CFAST Version 7.2.1 • NIST-TN-1889v2, CFAST Version 7.2.1 • NUREG-1824, Volume 5, 2007 • NUREG-1824, Supplement 1, 2014 • NUREG-1934, Chapter 2, 2012	• V&V of the CFAST code is documented in the NIST Special Publication 1086. • The V&V of CFAST specifically for Nuclear Power Plant applications is documented in NUREG-1824 its Supplement. • It is concluded in NUREG-1824, Volume 5, Chapter 6, "Model Validation", that CFAST models the HGL height, temperature and smoke concentration in an appropriate manner. • The model has been applied within its limits of applicability and within the validated range reported in NUREG-1824 or, if applied outside the validated range, the model has been justified as acceptable, either by qualitative analysis, or by quantitative sensitivity analysis. The methodology for justifying application of the fire model outside the range is in accordance with methods documented in NUREG-1934. The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2 • H-RIE-FIREPRA-U00-008B, "Hatch Fire PRA Task 11c, Evaluation of the Control Room Abandonment Times", Version 2 • H-RIE-FIREPRA-U00-008C, Hatch Fire PRA Task 11B, Multi-Compartment Analysis", Version 2. • H-RIE-FIREPRA-U00-008D, Hatch Fire PRA Task 8D, Exposed Structural Steel Analysis", Version 2.

Table J-1 V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Smoke Detection Actuation Correlation (Method of Heskestad and Delichatsios)	Alpert Ceiling Jet correlation is used to determine temperature and the Heskestad and Delichatsios temperature to smoke density correlation for smoke detection timing estimates	Technical Reference: • NUREG-1805, Chapter 11, 2004 • NUREG-1824, Volume 4, 2007 • NUREG-1934, Chapter 2, 2012 • SFPE Handbook of Fire Protection Engineering, 5th Edition, Chapter 40, Custer R., Meacham B., and Schifiliti, R., 2016 • SFPE Handbook of Fire Protection Engineering, 5th Edition, Chapter 14, Alpert, R., 2016	• The smoke detection correlation is contained in NUREG-1805. • Alpert's ceiling jet correlation V&V is documented in NUREG-1824. • The correlation has been applied within its limits of applicability and the validated range reported in NUREG-1824 or, if applied outside the validated range, the model has been justified as acceptable, either by qualitative analysis, or by quantitative sensitivity analysis. The methodology for justifying application of the fire model outside the range is in accordance with methods documented in NUREG-1934. • The temperature to smoke density correlation is documented in an authoritative publication of the "SFPE Handbook of Fire Protection Engineering." The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2, Attachment A, Fire Modeling Workbook Methodology for HNP
Heat Detection Actuation Correlation	Estimates heat detector timing based on the Alpert ceiling jet temperature, velocity, and thermal response of sprinkler.	Technical Reference: • NUREG-1805, Chapter 11, 2004 • NFPA Fire Protection Handbook, 19th Edition, Chapter 3-9, Budnick, E., Evans, D., and Nelson, H., 2003 • NUREG-1824, Volume 4, 2007 • NUREG-1934, Chapter 2, 2012	• The heat detection correlation is contained in NUREG-1805. • The correlation is documented in an authoritative publication of the NFPA Fire Protection Handbook. • Alpert's ceiling jet correlation V&V is documented in NUREG-1824. • The correlation has been applied within its limits of applicability and the validated range reported in NUREG-1824 or, if applied outside the validated range, the model has been justified as acceptable, either by qualitative analysis, or by quantitative sensitivity analysis. The methodology for justifying application of the fire model outside the range is in accordance with methods documented in NUREG-1934. The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2, Attachment A, Fire Modeling Workbook Methodology for HNP

Table J-1 V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Corner and Wall Effects	Determines a heat release rate adjustment factor for fires that are proximate to a wall or corner.	Technical Reference: • SFPE Handbook of Fire Protection Engineering, 5th Edition, Chapter 25, Lattimer, 2016. • Zukoski, E.E., "Properties of Fire Plumes," Combustion Fundamentals of Fire, Cox, G., Ed., Academic Press, London, 1995. • NIST-GCR-90-580, "Development of an Instructional Program for Practicing Engineers Hazard I Users," Barnett, J. R. and Beyler, C. L., NIST GCR, NIST, Gaithersburg, MD, July, 1990. • IMC 0609, Appendix F, "Fire Protection Significance Determination Process"	• The correlation is documented in an authoritative publication of the "SFPE Handbook of Fire Protection Engineering." • The heat release rate input to plume and ceiling jet correlations is adjusted by using a "location factor" when the fire is located within two feet of a wall or corner. This location factor doubled the heat release rate for both the plume and ceiling jet correlations for a fire near a wall, and quadrupled it for a fire near a corner. • Although not specifically Verified and Validated in NUREG-1824, the correlation is documented in recognized Fire Protection Engineering publications. • The correlation is widely accepted and utilized in the industry, for example, it is recommended by IMC 0609. • The correlation has been applied within its limits of applicability and in a manner consistent with the referenced studies or has been qualitatively justified as acceptable. The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2. • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2, Attachment A, Fire Modeling Workbook Methodology for HNP.

Table J-1 V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Correlation for Flame Spread over Horizontal Cable Trays (FLASHCAT)	Predicts the growth and spread of a fire within a vertical stack of horizontal cable trays.	Technical Reference: • NUREG/CR-7010, Section 9, 2012 • NUREG/CR-6850, Volume 2, Appendix R, 2005	• The correlation is recommended by NUREG/CR-7010 and follows guidance set forth in NUREG/CR-6850. • The FLASH-CAT model is validated in NUREG/CR-7010, Section 9.2.3, through experimentally measured HRRs compared with the predictions of the FLASH-CAT model. • The correlation has been applied to configurations consistent with those reported in NUREG/CR-7010 or the correlation has been qualitatively justified as acceptable. The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2 • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2, Attachment A, Fire Modeling Workbook Methodology for HNP. • H-RIE-FIREPRA-U00-008B, "Hatch Fire PRA Task 11c, Evaluation of the Control Room Abandonment Times ", Version 2.
Stratified Hot Layer Method	Simple layer growth calculation using the governing equations from CFAST. Used to calculate the upper layer temperature and layer depth in the FMWB. Interacts with the Plume Centerline Temperature calculations.	Technical Reference • NIST-TN-1889v1, CFAST, Version 7	• This model is developed using the governing equations from the CFAST fire simulation model. • The V&V of CFAST specifically for Nuclear Power Plant applications is documented in NUREG-1824 and its Supplement. • It is concluded in NUREG-1824, Volume 5, Chapter 6, "Model Validation", that CFAST models the HGL height, temperature and smoke concentration in an appropriate manner. The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2, Attachment A, Fire Modeling Workbook Methodology for HNP.

Table J-1 V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Fire Modeling Workbook (FMWB)	Used to calculate the zone of influence associated with fire scenarios. Includes application of the following calculations: • Flame Height (Method of Heskestad) • Plume Centerline Temperature (Method of Heskestad) • Radiant Heat Flux (NUREG-1805 Solid Flame II (Method of Shokri and Beyler) • Ceiling Jet Temperature (Method of Alpert) • Smoke Detection Actuation Correlation (Method of Heskestad and Delichatsios) • Heat Detection Actuation Correlation • Corner and Wall Effects • Correlation for Flame Spread over Horizontal Cable Trays (FLASHCAT) • Heat Soak Method for Evaluating Time to Cable Damage	HNP Calculation: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Revision 3, Attachment A, Fire Modeling Workbook Methodology for HNP	• The FMWB is a collection of fire modeling correlations that are already documented in NUREG-1805 FDT[s], "Fire Dynamics Tools (FDT[s]): Quantitative Fire Hazard Analysis Methods for the US Nuclear Regulatory Commission Fire Protection Inspection Program," December 2004, and Fire Induced Vulnerability Evaluation (FIVE), "EPRI Fire Induced • Vulnerability Evaluation Methodology", Revision 1, Referenced by EPRI Report 1002981, 2002. • The fire modeling correlations within the Fire Modeling Workbook (FMWB) have been verified, by "black box" testing, to ensure that the results were identical to the verified and validated models. "Black box" testing (or functional testing) is testing that ignores the internal mechanism of a system or component and focuses solely on the outputs generated in response to selected inputs and execution conditions. • The results from the FMWB were compared to those produced by the NUREG-1805 FDT[s] and FIVE-Rev1, when identical inputs were entered into both. Since the correlations from NUREG-1805 FDT[s] and FIVE, Rev1, were verified and validated in NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," Final Report, April 2007, and the results match the results produced by the FMWB, the FMWB is verified and validated with respect to NUREG-1824. The applicability of the V&V basis to the model implementation in the HNP FPRA is described in the following HNP Calculations: • H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8/11a, Fire Scenario Development and Detailed Fire Modeling", Version 2, Attachment A, Fire Modeling Workbook Methodology for HNP

Table J-1 V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Heat Soak Method for Evaluating Time to Cable Damage	Used to estimate the time to damage for cables subject to varying time dependent temperatures	HNP Calculation: • TECH-FRA-02, A Heat Soak Method for Evaluating Time to Cable Damage, Revision 0	• The heat soak damage integral method has been verified to ensure that method was capable of reproducing the values listed in the NUREG/CR-6850, Appendix H tables. Additionally, verification was performed to demonstrate that the approach does not fail cables at low exposure temperatures and yield expected values for a time-dependent exposure. • The heat soak damage integral method was validated using the intermediate scale test data that was used to develop the THIEF model from NUREG/CR-6931. The validation results demonstrate that the heat soak method reproduces the conservatism of the strict application of Appendix H of NUREG/CR-6850 while still maintaining a slight degree of conservatism when compared to the higher fidelity THIEF method.

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22. H-RIE-FIREPRA-U00-008C, Hatch Fire PRA Task 11B, Multi-Compartment Analysis", Version 2
23. H-RIE-FIREPRA-U00-008D, Hatch Fire PRA Task 8D, Exposed Structural Steel Analysis", Version 2
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K. Existing Licensing Action Transition

21 Pages Attached

Licensing Action Number: 1

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.3, for Lack of an Area-Wide Automatic Fire Suppression System in the Control Room

Basis Date: November 16, 1981

Required Post Transition: No

Basis:

Exemption request per March 19, 1981 GPC letter to the NRC provides justification for lack of an area-wide automatic fire suppression system in the Control Room, which was approved by the NRC in a letter dated November 16, 1981.

This licensing action is not being credited for compliance with NFPA 805 because Fire Area 0024 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2, and no credit is taken for prior approval of the deterministic compliance strategy for this fire area.

Applicable Fire Areas:

Fire Area	Description
0024	Control Complex Control Building - el 147 ft and 164 ft

Licensing Action Number: 2

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of Automatic Fire Suppression Systems and 3-Hour Fire Rated Barriers in the Unit 1 4160V Transformer Room and Unit 1 West 600V Switchgear Room

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letters dated April 28, 1983 and May 27, 1983, provides justification for lack of an automatic fire suppression system and lack of a 3-hour fire rating of the perimeter walls in both the Unit 1 4160V Transformer Room and West 600V Switchgear Room, which was approved by the NRC in a letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because:

1. Fire Areas 1016 and 1019 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these areas, and

2. Calculation SMNH-16-061, Engineering Evaluation of Fire Area Boundaries within the Control Building, concludes that the barriers surrounding Fire Areas 1016 and 1019 are adequate for the hazard such that prior NRC approval of these barriers does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas:

Fire Area	Description
0014	Working Floor and HP Area Control Building - el 130 ft
1016	West 600V Switchgear Room Control Building - el 130 ft
1017	East 600V Switchgear Room Control Building - el 130 ft
1018	West DC Switchgear Room Control Building - el 130 ft
1019	Transformer Room Control Building - el 130 ft
1020	East DC Switchgear Room Control Building - el 130 ft

Licensing Action Number: 3

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of 3-Hour Fire Rated Barriers in the Unit 1 and Unit 2 Switchgear Rooms, Unit 2 Transformer Room, and Control Building Working Floor

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letters dated April 28, 1983 and May 27, 1983, provides justification for lack of 3-hour fire-rated barriers surrounding the control room elevator shaft and stairwell, Unit 1 and Unit 2 East and West DC Switchgear Rooms, Unit 1 and Unit 2 East 600V Switchgear Room, Unit 2 West 600V Switchgear Room, and Unit 2 4160V Transformer Room, which was approved by the NRC in a letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because Calculation SMNH-16-061, Engineering Evaluation of Fire Area Boundaries within the Control Building, concludes that the barriers surrounding the elevator shaft and stairwell in Fire Area 0001, and Fire Areas 1017, 1018, 1020, 2016, 2017, 2018, 2019, and 2020, are adequate for the hazard such that prior NRC approval of these barriers does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas:

Fire Area	Description
0001	Working Floor Control Building - el 112 ft
0014	Working Floor and HP Area Control Building - el 130 ft
1017	East 600V Switchgear Room Control Building - el 130 ft
1018	West DC Switchgear Room Control Building - el 130 ft
1020	East DC Switchgear Room Control Building - el 130 ft
2014	Unit 2 Switchgear Access Hallway Control Building - el 130 ft
2016	West 600V Switchgear Room Control Building - el 130 ft
2017	East 600V Switchgear Room Control Building - el 130 ft
2018	West DC Switchgear Room Control Building -el 130 ft
2019	Transformer Room Control Building - el 130 ft
2020	East DC Switchgear Room Control Building - el 130 ft

Licensing Action Number: 4

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of a 3-Hour Fire Rated Barrier or 1-Hour Barrier and Area-Wide Automatic Fire Detection and Suppression in the Unit 1 and Unit 2 Reactor Buildings

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letters dated April 28, 1983 , May 27, 1983, November 16, 1983, and November 30, 1983, provides justification for the presence of a water curtain in lieu of a physical barrier separating the north and south halves of the Unit 1 and Unit 2 Reactor Buildings, lack of area-wide suppression and detection, presence of unsealed penetrations and unrated blowout panels in credited barriers, and separation of redundant equipment by less than 20 feet without adequate separation in the Unit 1 and 2 Reactor Buildings, which was approved by the NRC in a letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because:

1. Fire Areas 1203, 1205, 2203, and 2205 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these areas, and

2. Calculation SMNH-16-059, Engineering Evaluation of Fire Area Boundaries within the Unit 1 and Unit 2 Reactor Buildings, concludes that the barriers surrounding Fire Areas 1203, 1205, 2203, and 2205 are adequate for the hazard such that prior NRC approval of these barriers and water curtains does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas:

Fire Area	Description
1203	Reactor Building South Unit 1 Reactor Building - All Elevations
1205	Unit 1 Reactor Building North Unit 1 Reactor Building - All Elevations
2203	Reactor Building North Unit 2 Reactor Building - All Elevations
2205	Unit 2 Reactor Building South Unit 2 Reactor Building - All Elevations

Licensing Action Number: 5

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of an Area-Wide Automatic Fire Suppression System in the Unit 2 Control Building Health Physics Area

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letter dated May 27, 1983, provides justification for lack of an area-wide automatic fire suppression system in the Unit 2 Health Physics Area of the Control Building, which was approved by the NRC in a letter dated April 18, 1984.

The licensing action is not being credited for compliance with NFPA 805 because Fire Area 0014 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2, and no credit is taken for prior approval of the deterministic compliance strategy for this fire area.

Applicable Fire Areas:

Fire Area	Description
0014	Working Floor and HP Area Control Building - el 130 ft

Licensing Action Number: 6

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of 3-Hour Fire Rated Barriers in the Unit 2 Control Building Switchgear Hallway

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letters dated April 28, 1983 and May 27, 1983, provides justification for lack of a 3-hour fire-rated barrier between the Unit 2 Control Building Switchgear Access Hallway and the Control Building Common Corridor, which was approved by the NRC in a letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because:

1. Fire Areas 0014 and 2014 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these fire areas, and

2. Calculation SMNH-16-061, Engineering Evaluation of Fire Area Boundaries within the Control Building, concludes that the separation between Fire Areas 0014 and 2014 is adequate for the hazard such that prior NRC approval of this separation does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas:

Fire Area	Description
0014	Working Floor and HP Area Control Building - el 130 ft
2014	Unit 2 Switchgear Access Hallway Control Building - el 130 ft

Licensing Action Number: 7

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of a 3-Hour Fire Rated Barrier in the Station Battery Rooms

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letters dated April 28, 1983 and May 27, 1983, provides justification for lack of 3-hour fire rated doors separating the Unit 1 and 2 Station Battery Rooms from the Control Building Working Floor El. 112', which was approved by the NRC in a letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because Calculation SMNH-16-042, Engineering Evaluation of Fire Door Assemblies, concludes that the doors between Fire Area 0001 and Fire Areas 1004, 1005, 2004, and 2005 are functionally equivalent such that prior NRC approval of these doors does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas:

Fire Area	Description
0001	Working Floor Control Building - el 112 ft
1004	Station Battery Room 1A Control Building - el 112 ft
1005	Station Battery Room 1B Control Building - el 112 ft
2004	Station Battery Room 2A Control Building - el 112 ft
2005	Station Battery Room 2B Control Building - el 112 ft

Licensing Action Number: 8

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of a 3-Hour Fire Rated Barrier in the Unit 2 Turbine Building Condenser Bay

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letter dated May 27, 1983, provides justification for lack of 3-hour fire-rated barriers surrounding the Unit 2 Main Condenser Area, which was approved by the NRC in a letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because Calculation SMNH-16-060, Engineering Evaluation of Fire Area Boundaries within the Unit 1 and Unit 2 Turbine Building, concludes that the barriers surrounding Fire Area 2101 are adequate for the hazard such that prior NRC approval of these barriers does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas:

Fire Area	Description
2101	Unit 2 Turbine Building Unit 2 Turbine Building el 112 ft, 130 ft, and 147 ft

Licensing Action Number: 9

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of 3-Hour Fire Rated Barriers in the Turbine Buildings, East Cableway, and West Cableway

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letter dated May 27, 1983, provides justification for lack of 3-hour fire-rated barriers separating the East Cableway from the Unit 1 and Unit 2 Turbine Building Working Floors and Condenser Bays, and separating the West Cableway from the Unit 1 and Unit 2 Turbine Buildings, which was approved by the NRC in a letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because:

1. Fire Areas 1101, 1104, 2101, and 2104 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these fire areas, and

2. Calculation SMNH-16-060, Engineering Evaluation of Fire Area Boundaries within the Unit 1 and Unit 2 Turbine Building, concludes that the barriers surrounding Fire Areas 1101, 1104, 2101, and 2104 are adequate for the hazard such that the prior NRC approval of these barriers does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas:

Fire Area	Description
1101	Unit 1 Turbine Building Unit 1 Turbine Building - el 112 ft, 130 ft, and 147 ft
1104	East Cableway Unit 1 Turbine Building - el 130 ft
2101	Unit 2 Turbine Building Unit 2 Turbine Building el 112 ft, 130 ft, and 147 ft
2104	East Cableway Turbine Building - el 130 ft

Licensing Action Number: 10

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of an Area-Wide Suppression System in Diesel Building Switchgear Room 2G

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letter dated May 27, 1983, provides justification for lack of an automatic fire suppression system in the Unit 2 Diesel Generator Building Switchgear Room 2G, which was approved by the NRC in a letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because Fire Area 2409 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for this fire area.

Applicable Fire Areas:

Fire Area	Description
2409	Switchgear Room 2G Diesel Generator Building - el 130 ft

Licensing Action Number: 11

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of 3-Hour Fire Rated Barriers or 1-Hour Barriers and Area-Wide Automatic Fire Detection and Suppression in the Control Building Common Corridor

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letters dated April 28, 1983 and May 27, 1983, provides justification for lack of 3-hour separation between the Control Building Common Corridor and the Fan Room, lack of 3-hour separation between the Control Building Common Corridor and the Unit 2 Control Building Switchgear Access Hallway, lack of full area sprinkler protection, and lack of 1-hour separation between redundant equipment, which was approved by the NRC in letter dated April 18, 1984.

This licensing action will not be credited for compliance with NFPA 805 because:

1. Fire Areas 0014 and 2101 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these fire areas, and

2. Calculation SMNH-16-062, Engineering Evaluation of NFPA 805 Power Block Building Separation, concludes that the barriers surrounding Fire Areas 0014 and 0811 are adequate for the hazard such that prior NRC approval of these barriers does not need to be maintained for NFPA 805 compliance, and

3. Calculation SMNH-16-061, Engineering Evaluation of Fire Area Boundaries within the Control Building, concludes that the separation between Fire Areas 0014 and 2014 is adequate for the hazard such that prior NRC approval of this separation does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas:

Fire Area	Description
0014	Working Floor and HP Area Control Building - el 130 ft
0811	Service Building Fan Room
2014	Unit 2 Switchgear Access Hallway Control Building - el 130 ft

Licensing Action Number: 12

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of an Area-Wide Automatic Fire Suppression System in the Intake Structure

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per July 1, 1982 GPC letter to the NRC, supplemented by letter dated May 27, 1983, provides justification for lack of an automatic fire suppression system in the Intake Structure, which was approved by the NRC in a letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because Fire Area 0501 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for this fire area.

Applicable Fire Areas:

Fire Area	Description
0501	Intake Structure North Boundary of Plant Site

Licensing Action Number: 13

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of a 3-Hour Fire Rated Barrier in the Control Building East Corridor

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Exemption request per May 27, 1983 GPC letter to the NRC provides justification for lack of 3-hour fire-rated barriers separating the Control Building 112' elevation East Corridor from the Unit 2 Turbine Building, and at the open stairwell between the Control Building 112' elevation East Corridor and the Unit 2 130' elevation East Cableway, which was approved by the NRC in letter dated April 18, 1984.

The licensing action is not being credited for compliance with NFPA 805 because:

1. Fire Areas 1101, 2101, and 2104 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these fire areas, and

2. Calculation SMNH-16-060, Engineering Evaluation of Fire Area Boundaries within the Unit 1 and Unit 2 Turbine Building, concludes that the barriers surrounding Fire Areas 1101, 2101, and 2104 are adequate for the hazard such that prior NRC approval of these barriers does not need to be maintained for NFPA 805 compliance, and

3. Calculation SMNH-16-061, Engineering Evaluation of Fire Area Boundaries within the Control Building, concludes that the barriers surrounding Fire Areas 0007, 2101, and 2104 are adequate for the hazard such that prior NRC approval of these barriers does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas:

Fire Area	Description
0007	Control Building East Corridor and HP Cold Lab Control Building Elevation - 112 ft 0 in.
1101	Unit 1 Turbine Building Unit 1 Turbine Building - el 112 ft, 130 ft, and 147 ft
2101	Unit 2 Turbine Building Unit 2 Turbine Building el 112 ft, 130 ft, and 147 ft
2104	East Cableway Turbine Building - el 130 ft

Licensing Action Number: 14

Licensing Action Description: Deviations from the Requirements of NFPA Standards 13, 14, and 15 with Respect to Hanger Selection and Spacing

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Request per May 27, 1983 GPC letter to the NRC, supplemented by letter dated December 20, 1983, provides justification for the sprinkler hanger design, selection and spacing criteria that does not meet the more stringent requirements of NFPA 13, 14, and 15, which was approved by the NRC in letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because Calculation SMNH-16-063, Engineering Evaluation of Sprinkler System Restraint Configurations, concludes that the systems are adequate for the hazard such that prior NRC approval of these systems does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas: None

Licensing Action Number: 15

Licensing Action Description: Deviation from the criteria of NFPA Standards Nos. 13, 14, and 15 with Respect to Component Selection

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Request per May 27, 1983 GPC letter to the NRC, supplemented by letter dated December 20, 1983, provides justification for the use of closed head directional spray nozzles and multibushing reductions to allow for the location of spray away from easily damaged equipment, which was approved by the NRC in letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because Calculation SMNH-16-064, Engineering Evaluation of Sprinkler Location and Spray Pattern Configurations, concludes that the systems are adequate for the hazard such that prior NRC approval of these systems does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas: None

Licensing Action Number: 16

Licensing Action Description: Deviations from the criteria of NFPA Standard Nos. 13, 15, and 72E with Respect to Sprinkler Head/Nozzle and Detector Placement

Basis Date: April 18, 1984

Required Post Transition: No

Basis:

Request per May 27, 1983 GPC letter to the NRC, supplemented by letter dated December 20, 1983, provides justification for the positioning of sprinkler heads and fire detectors to improve response time of the systems because of the nature of the facility, which was approved by the NRC in letter dated April 18, 1984.

This licensing action is not being credited for compliance with NFPA 805 because Calculation SMNH-16-064, Engineering Evaluation of Sprinkler Location and Spray Pattern Configurations, concludes that the systems are adequate for the hazard such that prior NRC approval of these systems does not need to be maintained for NFPA 805 compliance.

Applicable Fire Areas: None

Licensing Action Number: 17

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.J, for Lack of Emergency Lighting

Basis Date: January 2, 1987

Required Post Transition: No

Basis:

Exemption request per May 16, 1986 GPC letter to the NRC, supplemented by letters dated October 31, 1986, December 9, 1986, and December 11, 1986, provides justification for lack of 8-hour battery-powered emergency lights within the Control Room and Yard, which was approved by the NRC in letter dated January 2, 1987.

This licensing action will not be credited for compliance with NFPA 805 because there are no deterministic requirements for emergency lighting within NFPA 805; therefore, although adequate levels of emergency lighting must be maintained on site, the exemption from the requirements of 10 CFR 50 Appendix R is not required.

Applicable Fire Areas:

Fire Area	Description
0024	Control Complex Control Building - el 147 ft and 164 ft
1601	Unit 1 Service Water Valve Pit 1A East of the Diesel Generator Building - el 130 ft
1602	Unit 1 Service Water Valve Pit 1B East of the Diesel Generator Building - el 130 ft
2601	Unit 2 Service Water Valve Pit 2A East of the Diesel Generator Building - el 130 ft
2602	Unit 2 Service Water Valve Pit 2B East of the Diesel Generator Building - el 130 ft

Licensing Action Number: 18

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2.c, for Lack of 1-Hour Fire Rated Barriers in the Unit 1 and Unit 2 Reactor Buildings

Basis Date: January 2, 1987

Required Post Transition: No

Basis:

Exemption request per May 16, 1986 GPC letter to the NRC, supplemented by letters dated October 31, 1986, December 9, 1986, and December 11, 1986, provides justification for lack of 1-hour fire-rated barriers separating redundant shutdown divisions within the suppression system water curtain boundary between Fire Areas 1203 and 1205 and between Fire Areas 2203 and 2205, which was approved by the NRC in letter dated January 2, 1987 as corrected by the NRC in a letter dated March 24, 1987.

This licensing action will not be credited for compliance with NFPA 805 because Fire Areas 1203, 1205, 2203, and 2205 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these fire areas.

Applicable Fire Areas:

Fire Area	Description
1203	Reactor Building South Unit 1 Reactor Building - All Elevations
1205	Unit 1 Reactor Building North Unit 1 Reactor Building - All Elevations
2203	Reactor Building North Unit 2 Reactor Building - All Elevations
2205	Unit 2 Reactor Building South Unit 2 Reactor Building - All Elevations

Licensing Action Number: 19

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.1.a, for RHR and RCIC Pump Repairs

Basis Date: January 2, 1987

Required Post Transition: No

Basis:

Exemption request per May 16, 1986 GPC letter to the NRC, supplemented by letters dated October 31, 1986 and December 9, 1986, provides justification for the requirement to make repairs to circuits in the Control Room to maintain cooling capacity for the RHR and RCIC pumps after a fire to maintain hot shutdown.

This licensing action will not be credited for compliance with NFPA 805 because Fire Area 0024 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for this fire area.

Applicable Fire Areas:

Fire Area	Description
0024	Control Complex Control Building - el 147 ft and 164 ft

Licensing Action Number: 20

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Sections III.G.2.a and III.G.2.b, for Lack of 1-Hour Fire Rated Barriers in the Unit 1 and 2 Reactor Buildings

Basis Date: January 2, 1987

Required Post Transition: No

Basis:

Exemption request per May 16, 1986 GPC letter to the NRC, supplemented by letters dated October 31, 1986, November 14, 1986, and December 9, 1986, provides justification for lack of 1-hour fire-rated barriers in conjunction with fire detection and automatic suppression between redundant divisions within the Unit 1 Reactor Building North of Column Line R7, and within the Unit 2 Reactor Building South of Column Line R19, which was approved by the NRC in a letter dated January 2, 1987.

This licensing action will not be credited for compliance with NFPA 805 because Fires Areas 1205 and 2205 are being transitioned as performance-based areas in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for these fire areas.

Applicable Fire Areas:

Fire Area	Description
1205	Unit 1 Reactor Building North Unit 1 Reactor Building - All Elevations
2205	Unit 2 Reactor Building South Unit 2 Reactor Building - All Elevations

Licensing Action Number: 21

Licensing Action Description: Exemption from 10 CFR 50 Appendix R, Section III.G.2.b, for Lack of 20 Feet of Redundant Cable Separation in the Intake Structure

Basis Date: January 2, 1987

Required Post Transition: No

Basis:

Exemption request per May 16, 1986 GPC letter to the NRC provides justification for a separation distance of less than 20 feet between redundant shutdown divisions in the Intake Structure, which was approved by the NRC in letter dated January 2, 1987 as corrected by the NRC in a letter dated March 24, 1987, as corrected by the NRC in a letter dated March 24, 1987.

This licensing action will not be credited for compliance with NFPA 805 because Fire Area 0501 is being transitioned as a performance-based area in accordance with NFPA 805 Section 4.2.4.2 and no credit is taken for prior approval of the deterministic compliance strategy for this fire area.

Applicable Fire Areas:

Fire Area	Description
0501	Intake Structure North Boundary of Plant Site

L. NFPA 805 Chapter 3 Requirements for Approval (10 CFR 50.48(c)(2)(vii))

39 Pages Attached

Approval Request 1

NFPA 805 Section 3.2.3(1) states:

"Procedures shall be established for implementation of the fire protection program. In addition to procedures that could be required by other sections of the standard, the procedures to accomplish the following shall be established:

Inspection, testing, and maintenance for fire protection systems and features credited by the fire protection program."

SNC will utilize performance based methods to establish the appropriate inspection, testing, and maintenance frequencies for fire protection systems and features required by NFPA 805. Performance-based inspection, testing, and maintenance frequencies will be established as described in EPRI Technical Report TR-1006756. SNC requests NRC approval to utilize these performance-based methods as an acceptable variance from the requirements of NFPA 805 Chapter 3 requirements.

Basis for Request:

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

NFPA 805 Section 2.6.1, *Availability, Reliability, and Performance Levels,* requires that *"Acceptable levels of availability, reliability, and performance shall be established."*

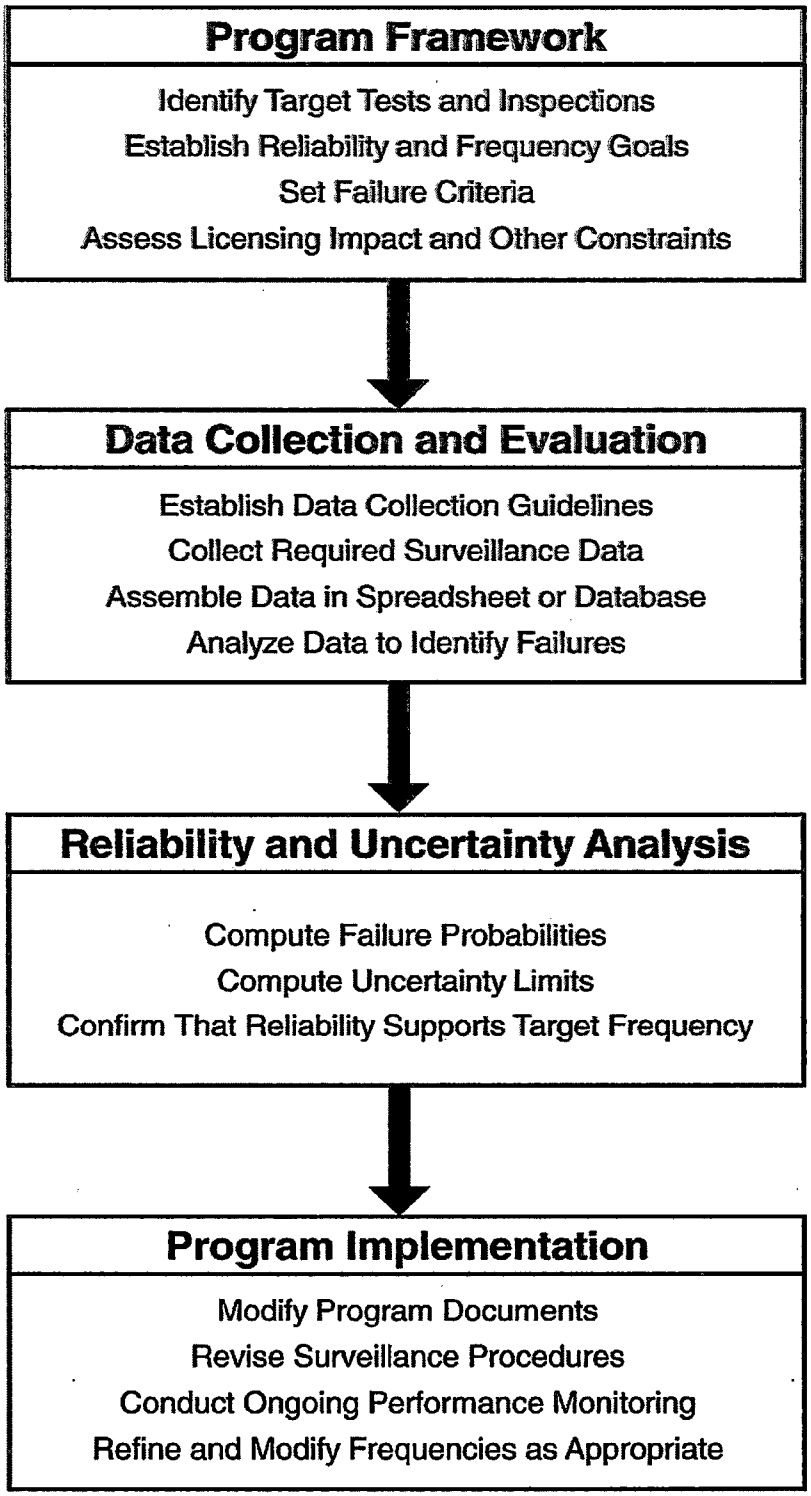
NFPA 805 Section 2.6.2 requires that "*Methods to monitor availability, reliability, and performance shall be established. The methods shall consider the plant operating experience and industry operating experience."*

The scope and frequency of the inspection, testing, and maintenance activities for fire protection systems and features required in the fire protection program have been established based on the previously approved Technical Specifications / License Controlled Documents and appropriate NFPA codes. This request does not involve the use of the EPRI Technical Report TR-1006756 to establish the scope of those activities as that is determined by the required systems review identified in Attachment C, Table C-2.

This request is specific to the use of EPRI Technical Report TR-1006756 to establish the appropriate inspection, testing, and maintenance frequencies for fire protection systems and features credited by the fire protection program. As stated in EPRI Technical Report TR-1006756, Section 10.1, "*The goal of a performance-based surveillance program is to adjust test and inspection frequencies commensurate with equipment performance and desired reliability."* This goal is consistent with the stated requirements of NFPA 805 Section 2.6. The EPRI Technical Report TR-1006756

provides an accepted method to establish appropriate inspection, testing, and maintenance frequencies which ensure the required NFPA 805 availability, reliability, and performance goals are maintained.

The target tests, inspections and maintenance are those activities for the NFPA 805 required fire protection systems and features. The reliability and frequency goals are established to ensure the assumptions in the NFPA 805 engineering analysis remain valid. The failure criterion is established based on the required fire protection systems and features credited functions and ensure those functions are maintained. Data collection and analysis follows the Technical Report TR-1006756 document guidance. The failure probability is determined based on the Technical Report TR-1006756 guidance and a 95% confidence level is utilized. The performance monitoring is performed in conjunction with the monitoring program required by NFPA 805 section 2.6 and it ensures site specific operating experience is considered in the monitoring process. The following is a flow chart that identifies the basic process that is utilized:



EPRI TR-1006756 - Figure 10-1
Flowchart for Performance-Based Surveillance Program

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

Use of performance based test frequencies established per EPRI TR-1006756 methods combined with NFPA 805 Section 2.6, Monitoring Program, will ensure that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analysis. Therefore, there is no adverse impact to Nuclear Safety Performance Criteria by the use of the performance based methods in EPRI TR-1006756.

The radiological release performance criteria are satisfied based on the determination of limiting radioactive release (Refer to Attachment E of this LAR). Fire protection systems and features are credited as part of that evaluation. Use of performance based test frequencies established per EPRI TR-1006756 methods combined with NFPA 805 Section 2.6, Monitoring Program, will ensure that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analysis which includes those assumptions credited to meet the radioactive release performance criteria. Therefore, there is no adverse impact to radioactive release performance criteria.

Safety Margin and Defense-in-Depth:

Use of performance based test frequencies established per EPRI TR-1006756 methods combined with NFPA 805 Section 2.6, Monitoring Program, will ensure that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analysis which includes those assumptions credited in the risk evaluation safety margin discussions. In addition, the use of these methods in no way invalidates the inherent safety margins contained in the codes used for design and maintenance of fire protection systems and features. Therefore, the safety margin inherent and credited in the analysis has been preserved.

The three echelons of defense-in-depth are:

(1) To prevent fires from starting (combustible/hot work controls)
(2) Rapidly detect, control and extinguish fires that do occur, thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans)
(3) Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions)

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

Echelon 1 is not affected by the use of EPRI TR-1006756 methods. Use of performance based test frequencies established per EPRI TR-1006756 methods combined with NFPA 805 Section 2.6, Monitoring Program, will ensure that the availability and

reliability of the fire protection systems and features credited for DID are maintained to the levels assumed in the NFPA 805 engineering analysis. Therefore, there is no adverse impact to echelons 2 and 3 for the defense in depth.

Conclusion:

NRC approval is requested for use of the performance based methods contained in EPRI TR-1006756 to establish the appropriate inspection, testing, and maintenance frequencies for fire protection systems and features required by NFPA 805. As described above, this approach is considered acceptable because it:

(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 nuclear safety capability).

Approval Request 2

NFPA 805 Section 3.3.4 states:

"Thermal insulation materials, radiation shielding materials, ventilation duct materials, and soundproofing materials shall be noncombustible or limited combustible."

NFPA 805 Section 1.6.36 defines Limited Combustible as:

> *"Material that, in the form in which it is used, has a potential heat value not exceeding 3500 Btu/lb (8141 kJ/kg) and either has a structural base of noncombustible material with a surfacing not exceeding a thickness of 1/8 in. (3.2 mm) that has a flame spread rating not greater than 50, or has another material having neither a flame spread rating greater than 25 nor evidence of continued progressive combustion, even on surfaces exposed by cutting through the material on any plane."*

NFPA 805 Section 1.6.41 defines Noncombustible as:

> *"A material that, in the form in which it is used and under the conditions anticipated, will not ignite, burn, support combustion, or release flammable vapors when subjected to fire or heat."*

Plant Hatch utilizes a flexible closed-cell thermal (anti-sweat) insulation in power block buildings that does not meet the requirements of Section 3.3.4. Specifically, the closed-cell thermal insulation used at Plant Hatch does not meet the NFPA 805 definition of a limited combustible in that it has a potential heat value that exceeds the code limit of 3500 Btu/lb. These materials meet the limited combustible flame spread rating criteria.

In addition, blowout panels constructed of 2-inch thick expanded polystyrene bead boards covered with .040 inch thick aluminum stucco are used to permit pressure relief to the Unit 2 turbine building in case of a main steam line break in the Unit 2 reactor building main steam pipe chase. The blowout panels are provided between the Unit 2 main steam pipe chase (Fire Zone 2205H) and turbine building, above el. 147 ft. (Fire Zone 2101K) and between the Unit 2 vent room above the pipe chase (Fire Zone 2205H) and turbine building, above el. 164 ft. (Fire Zone 0101J). Unit 1 blowout panels are constructed from materials that meet NFPA 805, Section 3.3.4.

SNC requests NRC approval for these materials as an acceptable variance from the requirements of NFPA 805 Chapter 3.

Basis for Request:

The basis for the approval request of this deviation is:

- The closed-cell thermal insulation was considered during the development of the plant fire modeling in that dimensions and locations of the insulation were used to

estimate additional heat release rates, potential changes in conditions and the possibility of new targets within a fire scenario.

- The limited applications of exposed thermal insulation installed to prevent condensation on the surface of system piping and for industrial personnel safety do not adversely affect installed fire protection systems and features. As such, essential safety functions are maintained and capable of being performed.

- Appendix A to Branch Technical Position APCSB 9.5-1 required thermal insulation to be non-combustible or listed by a nationally recognized testing laboratory, such as Factory Mutual or Underwriters' Laboratory Inc. for flame spread, smoke and fuel contribution of 25 or less in its use configuration (ASTM E-84 Test). This previous NRC accepted definition only addressed flame spread and smoke generation and did not consider potential heat value. The thermal insulation used at Plant Hatch met these Appendix A requirements.

- There are no significant ignition sources or fire hazards near the main steam pipe chase blowout panels. The relief panels are covered with aluminum stucco, with the exception of minor exposed polystyrene along the edges, which shields the expanded polystyrene bead board from exposure to fire and prevents ignition and flame spread. Although there is minor exposed polystyrene along the edges of the panels, given the panel construction and the amount of exposed material, a fire would not propagate across the barrier to involve additional combustible material on the adjacent side of the boundary. Therefore, the installation of the relief panels will not result in any additional fire damage or flame spread.

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The aluminum stucco covered polystyrene bead board wall panels and the closed-cell thermal insulation do not affect nuclear safety. Detection and means of manual suppression are provided in the areas that contain NSCA-credited equipment. Due to the configuration of the aluminum stucco covered polystyrene bead board wall panels and the thermal insulation, propagation across barriers is not likely to occur and redundant success paths will not be impacted. Therefore, there is no impact on the nuclear safety performance criteria.

The aluminum stucco covered polystyrene bead board wall panels and the closed-cell thermal insulation have no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the wall panels or thermal insulation. The aluminum stucco covered polystyrene bead board wall panels and the closed-cell thermal insulation do not change the radiological release evaluation, which concluded that potentially contaminated water is contained and smoke is monitored. The aluminum stucco covered polystyrene bead board wall panels and the closed-cell thermal insulation for

which NRC approval is requested do not add additional radiological materials to the area or challenge system boundaries.

Safety Margin and Defense-in-Depth:

The areas where the aluminum stucco covered polystyrene bead board wall panels and the closed-cell thermal insulation are located have been analyzed in their current configuration. The precautions and limitations of the use of these materials do not affect the analysis of the fire event. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are:

(1) To prevent fires from starting (combustible/hot work controls)
(2) Rapidly detect, control and extinguish fires that do occur, thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans)
(3) Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions)

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

Echelon 1 is maintained despite the aluminum stucco covered polystyrene bead board wall panels and the closed-cell thermal insulation adding small amounts of combustibles to the areas they are used. Plant housekeeping and hot work procedures are in place to limit the start of fires within the plant.

Echelon 2 is maintained as the use of the aluminum stucco covered polystyrene bead board wall panels and closed-cell thermal insulation does not affect the plant's ability to rapidly detect, control and extinguish fires that do occur, thereby limiting damage. The use of these materials does not affect automatic suppression systems from performing their functions, portable fire extinguishers and hose stations are available for manual firefighting activities that if a fire were to occur the damage would be limited.

Echelon 3 is maintained because success paths are available in each area of the plant where redundant pathways or required safe shutdown-related cables are located; one pathway remains protected allowing essential safety functions to be performed. The potential openings created from the aluminum stucco covered polystyrene bead board wall panels do not result in compromising post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the presence of aluminum stucco covered polystyrene bead board wall blowout panels between the Unit 2 steam chase and the turbine building and closed-cell thermal insulation materials that do not meet the requirements of NFPA 805. Based on the assessment above, the level of risk encountered by this

configuration is acceptable. As described above, this approach is considered acceptable because it:

(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 nuclear safety capability).

Approval Request 3

NFPA 805 Section 3.3.5.1 states:

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

Fire zones at Hatch Nuclear plant (HNP) contain wiring above suspended ceilings that is not in compliance with NFPA 805 Section 3.3.5.1. SNC requests NRC approval for the wiring above suspended ceilings in the associated rooms as an acceptable variance from the requirements of NFPA 805 Chapter 3.

This request is applicable for the following fire zones:

- Control Building 112'-0" Elevation:
  - 0007D, Respirator Room
  - 0007E, Chromatography Lab

- Radio Chemistry Lab and Rad Protection Area in the Control Building (noted in plant documentation as Health Physics Area):
  - 0014A, R.C. Lab
  - 0014B, H.P. Hallway
  - 0014C, Health Physics Area Storage
  - 0014D, HP Reference Area
  - 0014E, HP Receiving Area
  - 0014G, HP Counting Room
  - 0014H, Hot Lab
  - 0014I, HP Foreman's Office
  - 0014J, HP Office
  - 0014K, Control Building Working Floor
  - 0014M, Men's Room
  - 0014N, Ladies' Rest Room

- Control Room Area in the Control Building
  - 0024B, Computer Room
  - 0024C, Main Control Room (MCR)

- Shift Supervisor's Areas in the Turbine Building (attached to the Control Building):
  - 0101F, Main Control Room Break Area
  - 0101G, Chart Storage Room and Hallway
  - 0101H, Shift Clerk/Operations Supervisor's Office

Some of the wiring (cables) above the suspended ceiling in the Main Control Room (Fire Zone 0024C) are not in armored cable, metallic conduit, or enclosed cable trays.

The cable trays located above the suspended ceiling in the MCR containing cables for power, control and instrumentation are installed without covers.

The other areas listed above have small quantities (kept to a minimum) of exposed wiring (cables) above the suspended ceilings for video, communication, and data. These field routed cables are low voltage and are not susceptible to self-ignition. These cables are generally not plenum rated but many are IEEE-383 qualified. These cables are individually routed or routed together in small quantities.

Basis for Request:

The basis for the approval request of this deviation is:

- Wiring is kept to a minimum to support communications and other similar uses.
- The majority of the power, control and instrumentation cables above ceilings are in compliance with this requirement (i.e., plenum rated equivalent, armored, or routed in metallic conduit or covered cable trays).
- As stated in the Fire Hazards Analysis, the cable insulation at HNP is predominantly IEEE 383-qualified. There are no other significant combustibles located above the ceilings that could create a significant fire exposure to the small quantities of cables that are not enclosed. Fires involving cables are therefore expected to develop slowly.
- In the evaluated areas, excluding the control room, there are small quantities of low voltage video, communication, and data cables, which are not susceptible to self-ignition.
- Fires involving cable trays without covers are not expected to propagate beyond the cable tray of origin. As discussed in FAQ 13-0005, *Cable Fires Special Cases: Self Ignited and Caused by Welding and Cutting*, the EPRI Fire Events Database shows that self-ignited tray fires have only led to localized failures in a small number of cables within a single raceway. No event has led to sustained open flaming fires, or damage to cables beyond the initially impacted raceway. Given that the only significant combustibles above the ceiling are the cable trays, a fire involving the cable trays is expected to be contained within the tray of origin and not spread to additional raceways.
- The Control Room areas (Fire Zones 0024B and 0024C) are equipped with smoke detection both above and below the ceiling. The Main Control Room (Fire Zone 0024C) is also continuously manned with operators who have fire brigade training.
- Smoke from a fire above the ceiling can be manually exhausted to maintain a tenable environment in the MCR. The MCR (Fire Zone 0024C) environmental control system supplies HVAC for the MCR (Fire Zone 0024C), which is common

for Units 1 and 2, as shown in Drawing H-16056 and H-26120. There are two MCR (Fire Zone 0024C) exhaust fans operated only for purging smoke from the MCR (Fire Zone 0024C) in the event of a fire. The main control room is capable of venting smoke at a rate of 11,500 cfm to the outside via the Reactor Building vent plenum.

- In the HP Counting Room (Fire Zone 0014G) and Hot Lab (Fire Zone 0014H), partial water spray suppression and spot-type smoke detection are present above the ceiling. The other evaluated areas outside of the control room area have no automatic fire suppression or detection above the ceiling; however, these areas contain only small quantities of low-voltage wiring/cable that are not susceptible to self-ignition and will not contribute significantly to a fire above the ceiling. These areas are also normally occupied such that manual detection and fire brigade manual suppression of such a fire is likely.

- The above-ceiling space for the shift supervisor's area (Fire Area 0101) is used as an air plenum but smoke above the ceiling would not be recirculated to any other areas of the plant. The air-handling unit for this area is located in the ceiling space of the storage room per Drawing H-26117, and the condenser is located on the roof of the building. Air is supplied via ducted registers to each room (storage, office, and kitchen) and is returned to the air-handling unit from the ceiling space. Registers are located in each room to allow return air to flow into the ceiling space. A fan exhausts kitchen air to the Turbine Building atmosphere. The exhaust fan is manually energized from a local switch in the kitchen area and is operated as necessary. Makeup air is provided to the system from the cable spreading room supply air header.

- Other than Fire Area 0101, there are no plenum areas above suspended ceilings in the scope of this request. Smoke from a fire above a suspended ceiling is therefore not expected spread to adjacent or remote areas in the plant.

- The current configuration of cables and wiring above the ceilings was considered during the development of the plant fire modeling in support of the fire risk model.

- When replacing or installing new cables above suspended ceilings, SNC will meet the requirements of NFPA 805, Section 3.3.5.1. An implementation item will ensure revision to applicable design and/or cable installation guidance to apply the requirements of this section to future changes to the plant (See Attachment S, Table S-3, Implementation Item IMP-3).

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The presence of unenclosed cables above the suspended ceilings does not affect nuclear safety. The majority of the cables in the evaluated areas is either in compliance with this requirement or is IEEE-383 qualified. With the exception of the wiring above

the control room ceiling, wiring that is not in compliance is mostly small gauge, low-voltage cable not susceptible to self-ignition. Fires involving the power, control and instrumentation cables above the ceiling of the control room are not expected to spread beyond the tray of origin due to lack of other combustibles and credible ignition sources. Therefore, there is no impact on the nuclear safety performance criteria.

The cables above the suspended ceilings have no impact on the radiological release performance criteria. The radiological release review (Refer to Attachment E of this LAR) was performed based on the manual fire suppression activities in areas containing, or potentially containing, radioactive materials and is not dependent on the type of cables or locations of suspended ceilings. The cables do not add additional radiological materials to the areas or challenge any system boundaries. Therefore, the location of cables does not change the radiological release evaluation conclusion that potentially contaminated water is contained and smoke is monitored.

Safety Margin and Defense-in-Depth:

The video, communication, and data cables are screened out of the analysis in the FPRA based on approved analytical methods and are therefore not required for maintenance of safety margin. Cables in uncovered cable trays above the control room ceiling are not expected to propagate a fire beyond the tray of origin, as discussed in FAQ 13-0005. In addition, the areas with wiring above suspended ceilings have been analyzed in their current configuration. Therefore, the inherent safety margin and conservatisms in the analysis methods remain unchanged. Precautions and limitations established by Implementation Item IMP-3 will ensure that these materials do not impact future analyses of fire events.

The three echelons of defense-in-depth are:

(1) To prevent fires from starting (combustible/hot work controls)
(2) Rapidly detect, control and extinguish fires that do occur, thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans)
(3) Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions)

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

The exposed, non-plenum rated electrical cables located above suspended ceilings do not significantly affect echelon 1 of the defense-in-depth concept of preventing fires from occurring. The limited quantity of this wiring above suspended ceilings will not result in open, sustained flaming involving multiple trays and will therefore not cause fire damage to redundant components necessary for nuclear safety capability. Echelon 1 will also be maintained via the establishment of design and/or cable installation procedure guidance which includes the requirements of NFPA 805 Section 3.3.5.1 (IMP-3). The presence of the cables above suspended ceilings does not significantly diminish echelons 2 and 3. Most of the areas contain small quantities of low-voltage wiring/cable and are normally occupied such that quick manual detection and fire

brigade manual suppression is likely. The MCR which contains cables in trays without covers contain power, control, and instrumentation wiring, is provided with smoke detection above and below the suspended ceiling and is continuously-manned. Most of the cables at HNP are IEEE-383 qualified. A reasonable balance of the three echelons are provided; therefore, defense-in-depth is achieved.

Conclusion:

NRC approval is requested for the existence of exposed, electrical wiring above suspended ceilings. The current configuration of cables and wiring above the ceilings has been considered within the fire risk model. Based on the assessment above, the level of risk encountered by this configuration is acceptable. As described above, this approach is considered acceptable because it:

(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 nuclear safety capability).

Approval Request 4

NFPA 805 Section 3.3.5.2 states:

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

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

SNC requests NRC approval for the use of nonmetallic conduit in embedded applications and for the use of flexible PVC coated metallic conduits in lengths up to 6-feet as acceptable variances from the requirements of NFPA 805, Chapter 3.

Basis for Request:

The basis for the approval request of the deviation for the use of PVC coated flexible metallic conduits in lengths up to 6-feet is:

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

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

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

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

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

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

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

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

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

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

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

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The use of nonmetallic conduit in embedded applications and the use of flexible PVC coated metallic conduits in lengths up to 6-feet does not affect nuclear safety as the material in which conduits are run are located such that they are not subject to failure mechanisms that potentially result in circuit damage or damage to external targets. Additionally, NFPA 70 allows for the use of rigid nonmetallic conduit for underground and embedded applications. If a fire were to occur in a fire area containing these conduits, existing controls such as fire-rated barriers, electrical raceway fire barrier systems, spatial separation, etc. would ensure redundant cabling and circuitry would not be affected by the fire. Therefore, there is no impact on the nuclear safety performance criteria.

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

Safety Margin and Defense-in-Depth:

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

The three echelons of defense-in-depth are:

(1) To prevent fires from starting (combustible/hot work controls)
(2) Rapidly detect, control and extinguish fires that do occur, thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans)
(3) Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions)

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

The current configuration of the conduit at Plant Hatch does not impact fire protection defense-in-depth.

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

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

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

Conclusion:

NRC approval is requested for the use of nonmetallic conduit in embedded applications and for the use of flexible PVC coated metallic conduits in lengths up to 6-feet. The engineering analysis performed determined that the performance-based approach utilized to evaluate a variance from the requirements of NFPA 805 Chapter 3:

(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;

(B) Maintains safety margins; and

(C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire shutdown capability).

Approval Request 5

NFPA 805 Section 3.5.2 states:

"The tanks shall be interconnected such that fire pumps can take suction from either or both. A failure in one tank or its piping shall not allow both tanks to drain. The tanks shall be designed in accordance with NFPA 22, Standard for Water Tanks for Private Fire Protection."

NFPA 22, 1976 Edition, Section 8-2.11 states:

"An approved check valve shall be placed horizontally in the discharge pipe and shall be located in a pit under the tank when the tank is on an independent tower. When the tank is located over a building, the check valve shall ordinarily be placed in a pit, preferably outside the building. When yard room is not available, the check valve may be located on the ground floor or in the basement of a building provided that it is adequately protected against breakage."

Check valves are not provided in the discharge piping from either fire protection water storage tank as required by NFPA 22.

NFPA 805 Section 3.5.10 states:

"An underground yard fire main loop, designed and installed in accordance with NFPA 24, Standard for the Installation of Private Fire Service Mains and Their Appurtenances, shall be installed to furnish anticipated water requirements."

NFPA 24, 1973 Edition, Section 3202 states:

"Where there is more than one source of water supply, a check valve shall be installed in each connection, except that, where cushion tanks are used with automatic fire pumps, no check valve is required in the cushion tank connection."

Check valves are not provided in the discharge piping from either fire protection water storage tank as required by NFPA 24.

SNC requests NRC approval for the lack of check valves in the fire protection water storage tanks discharge piping as an acceptable variance from the requirements of NFPA 805 Chapter 3.

Basis for Request:

The basis for the approval request of this deviation is:

- Both fire protection water storage tanks are equipped with low-level water alarms that are set to alarm at the control panel, located in the Main Control Room, when the water level is lowered 12 inches from the required tank level, which is less than 10,000 gallons of the 300,000 gallon capacity tanks. If a low level alarm

is received, prompt action would be taken per the applicable abnormal operating procedure to investigate the cause and take corrective actions.

- Control isolation valves are available for isolation of each tank and the associated discharge piping in the event of a leak.

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The lack of check valves on fire protection water storage tank discharge piping does not adversely affect nuclear safety capability given the history of reliability of the existing configuration. Additionally, if a leak that affected the water supply from both fire water storage tanks were to develop, prompt action would be taken based on available indication within the Main Control Room to ensure an adequate volume of water is available in the event of a fire. Therefore, there is no impact on the nuclear safety performance criteria.

The lack of check valves on fire protection water storage tank discharge piping has no impact on the radiological release performance criteria. The radioactive release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the listed deviation or parts associated with the deviation. The identified deviation does not change the radiological release evaluation, which concluded that potentially contaminated water is contained and smoke is monitored. The identified deviation for which NRC approval is requested does not add additional radiological materials to the area or challenge system boundaries.

Safety Margin and Defense-in-Depth:

The lack of check valves on the fire protection water storage tank discharge piping will not impact the ability of the plant to achieve and maintain the fuel in a safe and stable condition. Given the history of reliability of the fire protection water storage tank piping and the capability to isolate leaks that do develop, an adequate volume of water will be available in the event of a fire. Therefore, the safety margin inherent in the analysis for a fire event is maintained.

The three echelons of defense-in-depth are:

(1) To prevent fires from starting (combustible/hot work controls)
(2) Rapidly detect, control and extinguish fires that do occur, thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans)
(3) Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions)

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

The lack of check valves on the fire protection water storage tank discharge piping does not affect echelons 1 or 3. Echelon 2 is maintained by the installed control valves, low-level water alarms, and abnormal operating procedure that ensure that the fire protection water storage tanks are not drained in the event of a leak. The current deviations from the requirements of NFPA 22, 1976 Edition and NFPA 24, 1973 Edition, do not directly result in compromising automatic fire suppression functions, manual fire suppression functions, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the lack of check valves on the fire protection water storage tank discharge piping. The level of risk encountered by maintaining this configuration is acceptable. The engineering analysis performed determined that the performance-based approach utilized to evaluate a variance from the requirements of NFPA 805 Chapter 3:

(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;

(B) Maintains safety margins; and

(C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

Approval Request 6

NFPA 805 Section 3.5.3 states:

"Fire pumps, designed and installed in accordance with NFPA 20, Standard for the Installation of Stationary Pumps for Fire Protection, shall be provided to ensure that 100 percent of the required flow rate and pressure are available assuming failure of the largest pump or pump power source."

A review of NFPA 20, 1972 Edition, identified nine sections that required NRC approval for electric fire pump deviations from the listed code requirements. Sections and requirements where deviations were identified in the code review are listed below:

Section 433.a requires power supply protective devices, when installed in the power supply circuits at utility plants, substations, or plant load distribution centers ahead of the fire pump feeder circuits, to hold indefinitely stalled rotor current conditions of the fire pump motor(s) under maximum plant load.

Section 433.b requires power supply protective devices, when installed in the fire pump feeder circuit, to hold indefinitely stalled rotor current of the fire pump motor(s) and other necessary associated fire pump installation electrical accessories.

Section 513.e requires circuits which are depended upon for proper operation of the controller to not have over-current protective devices connected in them.

Section 514.b.1 requires that no other overcurrent protective devices be used in the motor circuit on the load side of the circuit breaker.

Section 514.b.8 requires that the motor branch circuit's interrupting rating shall be adequate for the circuit in which it is used, and in no case be less than 14,000 amperes.

Section 514.b.9 requires that the motor branch circuit's interrupting rating should be obtained by the purchaser based upon the maximum possible short-circuit current at the pump room.

Contrary to the above requirements of NFPA 20, there are other over current devices installed on the load side of the circuit breakers to protect the Class 1E bus allowing continual service to critical plant safety equipment. Therefore, the circuit breaker for the electric motor-driven fire pump is not expected to hold stalled rotor current conditions indefinitely under maximum plant load. Additionally, the interrupting rating is less than the minimum allowable 14,000 amperes due to requirements of the Class 1E design circuit protection.

Additionally, the following sections of NFPA 20, 1972 are not specifically met:

Section 5.11.c requires that all controllers be specifically approved for fire pump service.

Section 512.a requires that the controller be located as close to as is practical and within sight of the motor.

Section 515.c.3.(a) requires that the controller be equipped with a handle or lever which operates to close the motor-circuit switching mechanism mechanically for non-automatic continuous running operation of the motors independent of any electric control circuits or magnets (or equivalent devices) and independent of the pressure-activated control switch. Means shall be incorporated for mechanically latching or holding of the handle or level for manual operation in the actuated position. The mechanical latching shall not be automatic, but at the option of the operator.

The electric fire pump controller is not UL-listed for fire pump service, the 1E bus is located in the Diesel Generator Building and not within sight of the motor, and is not equipped with a handle or lever which operates to close the motor-circuit switching mechanism mechanically for non-automatic continuous running operation of the motor.

SNC requests NRC approval for the above listed fire pump deviations against the identified sections of NFPA 20 as acceptable variances from the requirements of NFPA 805 Chapter 3.

Basis for Request:

Based on a review of NFPA 20, 1972 Edition, six different deviations affecting nine code sections were identified and are listed below:

1) Configuration of overcurrent devices installed on the load side of the electric fire pump circuit breakers allow continual service to critical plant equipment which make it unable to hold electric fire pump stalled rotor current conditions indefinitely under maximum plant load. (NFPA 20, 1972 Edition Sections 433.a and 433.b)
2) Use of existing Class 1E emergency power supply circuit breakers for the electric motor-driven fire pump and circuit design. (NFPA 20, 1972 Edition, Sections 513.e and 514.b.1)
3) Inadequate interrupting rating for the fire pump circuit. (NFPA 20, 1972 Edition, Sections 514.b.8 and 514.b.9)
4) Use of non-listed components within the fire pump controller configuration. (NFPA 20, 1972 Edition, Section 511.c)
5) Portions of the controller function located out of sight of the motor. (NFPA 20, 1972 Edition, Section 512.a)
6) Lack of a handle or lever for the current motor-circuit switching mechanism of the fire pump controller. (NFPA 20, 1972 Edition, Section 515.c.3.(a))

The bases for the approval request of the above listed deviations are:

- Operating experience and test results have demonstrated the adequacy of the existing circuit breaker and electric motor-driven fire pump configuration. The pump motor, impeller, and circuit breaker configuration have a history of reliability

and have proven not to fail the primary incoming power supply to the switchgear or cause inadvertent trips and/or pump start failure under dead start, full load conditions or normal operating load conditions. The motor-driven fire pump receives power from the plant 1E emergency bus, which is more reliable than a typical power supply configuration for a commercial building. The 1E bus also serves critical plant equipment and is therefore supplied by offsite power as well as an emergency generator. The non-UL-listed components of the fire pump controller have been evaluated against the requirements of NFPA 20 and, despite not being listed, are determined to be equivalent and adequate for their use in their current configuration.

- According to the annex information in the 2016 edition of NFPA 20, the intent of the emergency-run mechanical control is to provide a means for starting and running the fire pump when normal electric/magnetic operation of the contactor is not possible (i.e., failure of the pump starter). The fire pump motor start is currently configured to protect the 1E bus from overcurrent damage that could result in interruption of power for other critical equipment on the 1E bus, however, the configuration still meets the intent of the NFPA 20 requirement. In the event of automatic starter failure, alarms indicating failure to start are provided in the control room. The breaker at the 1E bus can be operated mechanically in the event of loss of control power. The electric fire pump is inspected, tested, and maintained at periodic intervals to ensure reliable operation. There are also two redundant diesel fire pumps with both automatic and manual start capability that will remain available in the unlikely event that the electric fire pump is not available.
- A single 70-gal/min 125-psi, pressure maintaining pump (jockey pump) is provided to keep the system filled and pressurized during low flow drawoffs and leakages. A 2500-gal/min electric motor-driven fire pump will start automatically on decrease in water pressure below the operating pressure of the jockey pump. If the demand exceeds the capacity of the electric motor-driven fire pump or upon failure of this pump, a lower pressure will start one of two diesel-driven pumps. A continuing pressure drop will start the second diesel-driven pump. All pumps are located inside the fire protection pump house. Alarms indicating pump running, drive availability, or failure to start are provided in the control room.
- In the unlikely event of a circuit breaker trip, administrative procedures require personnel to immediately investigate the condition and restore the power supply when it is confirmed to be electrically safe to perform such action. In the interim, both diesel-driven fire pumps are available and are designed to automatically start if the electric pump fails to supply adequate pressure to the fire protection water system. NFPA 20 Sections 433.a, 433.b, 513.e, 514.b.1, 514.b.8, and 514.b.9 do not discuss the presence of redundant diesel-driven fire pumps in a nuclear power plant application; they discuss the circuit breaker requirement for a single electric-motor-driven fire pump. Under normal operating conditions, three fire pumps with independent power supplies and controls are available to serve the maximum sprinkler system and hose stream hydraulic demands at Plant

Hatch. As indicated in inspection, testing, and maintenance test results, the fire pumps remain capable of performing their intended design function.

- Despite the control breaker for the electric fire pump being located in the Diesel Generator Building and not within sight of the of the electric fire pump motor which it serves, site personnel are trained to operate the controller function and are knowledgeable of its location so as to not prove a hindrance in functionality. The Diesel Generator Building is located less than 300 ft (91 m) from the electric fire pump and its controller, is within sight of the Fire Pump House, and can be easily accessed by site personnel.

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The inability to maintain indefinite stalled rotor current conditions, use of the power supply circuit breakers for the electric motor-driven fire pump, an inadequate interrupting rating, a non-listed controller with portions not located within sight of the motor, and lack of a mechanical motor-circuit switching mechanism do not adversely affect nuclear safety capability given the history of reliability of the existing configuration in addition to the backup capability of the diesel-driven fire pumps, the ability of the breaker to be operated mechanically in the event of loss of control power and the precautions implemented per plant procedures. Therefore, there is no impact on the nuclear safety performance criteria.

The radioactive release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the listed deviations or parts associated with said deviations. The identified deviations do not change the radiological release evaluation, which concluded that potentially contaminated water is contained and smoke is monitored. The identified deviations for which NRC approval is requested do not add additional radiological materials to the area or challenge system boundaries.

Safety Margin and Defense-in-Depth:

The inability to maintain an indefinite stalled rotor current condition under maximum plant load, use of the existing Class 1E emergency power supply circuit breakers for the electric motor-driven fire pump and circuit design, an inadequate interrupting rating for circuits, portions of the controller function which are located out of sight of the motor, and the current motor-circuit switching mechanism not equipped with a handle or lever are considered adequate in their current configuration. Given the history of reliability of the power supply, the electric fire pump and its related circuits, the backup capability of the two diesel-driven fire pumps, the ability of the breaker to be operated mechanically in the event of loss of control power and the precautions implemented per plant procedures, the safety margin is not considered to be adversely affected. Therefore, the safety margin inherent in this analysis for a fire event has been preserved.

The three echelons of defense-in-depth are:

(1) To prevent fires from starting (combustible/hot work controls)

(2) Rapidly detect, control and extinguish fires that do occur, thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans)
(3) Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions)

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

The inability to maintain an indefinite stalled rotor current condition under maximum plant load, use of Class 1E emergency power supply circuit breakers, inadequate interrupting ratings for circuits, positioning portions of the pump controller out of sight of the motor, use of non-listed pump controller components, and not providing a mechanical means for the current motor-circuit switching do not affect echelons 1 and 3. Echelon 2 is not adversely affected as operating experience and test results have demonstrated the adequacy of the existing circuit breaker and electric motor-driven fire pump configuration, including the inability to achieve indefinitely stalled rotor current conditions, held under maximum plant load. Also, site personnel are trained to operate the controller function and are knowledgeable of its location so as to not prove a hindrance in functionality. The current deviations from NFPA 20, 1972 Edition, do not result in compromising automatic fire suppression functions, manual fire suppression functions, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the listed deviations. The level of risk encountered by maintaining this configuration is acceptable. The engineering analysis performed determined that the performance-based approach utilized to evaluate a variance from the requirements of NFPA 805 Chapter 3:

(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;

(B) Maintains safety margins; and

(C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire shutdown capability).

Approval Request 7

NFPA 805 Section 3.5.5 states:

"Each pump and its driver and controls shall be separated from the remaining fire pumps and from the rest of the plant by rated fire barriers."

A floor-to-ceiling fire-rated barrier is not present between Fire Zones 0702A (Water Pump Room) and 0702B (West Fire Pump Room) separating the electric fire pump room from the sanitary water pump room. Additionally, the exterior walls of the Fire Pump House are not rated fire barriers.

The Fire Pump House consists of fire rated barriers separating the three fire pumps from one another. However, the electric motor driven fire pump located in Fire Zone 0702B is not separated from the sanitary water pumps located in Fire Zone 0702A by a fire rated barrier. The wall separating Fire Zones 0702A and 0702B is of non-rated concrete block construction and contains openings near the ceiling and at the non-fire rated, normally open door. Additionally, none of the three fire pump rooms are separated from exterior areas surrounding the Fire Pump House due to non-fire rated exterior walls. The walls to the exterior are non-rated sheet metal.

SNC requests NRC approval for the lack of fire rated barriers between the fire pumps and other areas of the plant as an acceptable variance from the requirements of NFPA 805 Chapter 3.

Basis for Request:

The basis for the approval request of this deviation is:

- The fire pumps are properly separated from one another with fire rated barriers. As such, the loss of one pump would not impact the ability to provide 100% of the required fire water demand.
- There are no significant intervening combustibles between the fire pump house and nearby structures. The nearest structures to the fire pump house are the fire protection tanks which are located approximately 10 feet away. These tanks do not present a hazard to the fire pumps or their functionality.
- Both the diesel driven pumps and the electric fire pump are provided with automatic sprinkler protection and portable CO2 fire extinguishers. Additionally, a fire hydrant is located just south of the Fire Pump House.
- With the exception of the fire water storage tanks, the Fire Pump House is located greater than 50 feet from the nearest significant structure.
- There are no credible fire scenarios that would impact all three fire pumps.

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The lack of fire rated separation between the fire pumps and exterior areas of the plant does not affect nuclear safety as the fire pumps are not relied upon for nuclear safety capability functions. A failure of any one pump will not affect the starting of the remaining pumps. Therefore, there is no impact on the nuclear safety performance criteria.

The lack of fire rated separation between the fire pumps and other areas of the plant has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the location of the fire pumps. The location of the fire pumps does not change the radiological release evaluation performed that concluded that potentially contaminated water is contained and smoke is monitored. The configuration of the fire pumps does not add additional radiological materials to the area or challenge system boundaries.

Safety Margin and Defense-in-Depth:

The lack of fire rated separation between fire pumps and other areas of the plant does not impact the ability to supply the required fire water in a fire event. The fire pumps are adequately separated from one another by rated fire barriers and protected by automatic suppression systems. Should a fire occur at one of the pumps or should a pump fail, the functionality of the remaining fire pumps will be unaffected. Therefore, the safety margin inherent and credited in the analysis is maintained.

The three echelons of defense-in-depth are:

(1) To prevent fires from starting (combustible/hot work controls)
(2) Rapidly detect, control and extinguish fires that do occur, thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans)
(3) Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions)

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

The location of the fire pumps does not affect echelon 1 because the location of the fire pumps does not affect the ability to prevent fires from starting. Echelon 2 is not adversely affected by the location of the fire pumps, in relation to other parts of the plant, as there are three fire pumps at HNP, all separated from one another by fire-rated barriers. Echelon 3 is maintained by adequate rated barriers between fire pumps to ensure that two fire pumps remain operable if a fire affects one pump. The lack of fire rated separation between fire pumps and other areas of the plant does not result in

compromising the automatic fire suppression functions, manual fire suppression functions, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the lack of fire rated barriers between the fire pumps and other areas of the plant.

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

(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;

(B) Maintains safety margins; and

(C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

Approval Request 8

NFPA 805 Section 3.5.16 states:

"The fire protection (FP) 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."

Contrary to the requirements of NFPA 805 Section 3.5.16, the fire protection water supply is used for the following non-fire protection purposes:

- Decay Heat Removal (DHR). In the event normal cooling tower make-up (demineralized water transfer) is lost, water supplied from the electric fire pump may be used as emergency make-up to the secondary loop. This is detailed in system operating procedure 34SO-G71-001-0.

- Loss of Plant Service Water (Condensate Pumps and Mechanical Vacuum Pump). If a significant loss of Plant Service Water (PSW) exists and cannot be promptly restored, water from the fire protection water system may be used to provide an alternate source of cooling to the condensate pump and/or mechanical vacuum pump. This is detailed in abnormal operating procedures 34AB-P41-001-1/2.

- Standby Liquid Control. If the Standby Liquid Control (SBLC) storage tank needs to be refilled to provide additional boron for shutting down the reactor, water from the fire protection water system may be used to refill the SBLC tank if the normal demineralized water supply is not available. This is detailed in system operating procedures 34SO-C41-003-1/2 and in emergency operating procedures 31EO-EOP-109-1/2.

- Spent Fuel Pool and Reactor Vessel Cooling. In the event of a loss of fuel pool cooling or decreasing reactor well/fuel pool water level, water from the fire protection water system may be used if make-up from the condensate storage tank and demineralized water system are not available. This is detailed in abnormal operating procedures 34AB-G41-001-1/2 and 34AB-G41-002-1/2.

- Alternate Reactor Pressure Vessel (RPV) Water Level Control. In the event of a loss of normal methods to control the water level in the RPV, water from the fire protection water system may be used to the control the water level of the RPV

and fill the Condensate Storage Tanks (CST). This is detailed in emergency operating procedures 31EO-EOP-110-1/2.

- Primary Containment Flooding. In the event water level drops below the top of the active fuel, water from the fire protection water system may be used to flood primary containment. This is detailed in emergency operating procedures 31EO-EOP-112-1/2.

- Cleaning Main Condenser Tubes. The fire protection water system may be used for cleaning main condenser tubes during outages. This evolution is controlled via the work management process.

SNC requests NRC approval to use the fire protection water supply system for the above listed non-fire protection functions as an acceptable variance from the requirements of NFPA 805 Chapter 3.

Basis for Request:

The manual use of the fire protection water for these non-fire protection system water demands would have no adverse impact on the ability of the fire protection system to provide required flow and pressure, based on the following:

Decay Heat Removal

The DHR system is primarily used during refueling outages when the decay heat load is high so system use is limited. In addition, water from the fire protection water system will only be used if the normal make-up supply from the demineralized water transfer system is unavailable. If required, the maximum water flow to the DHR cooling tower is 260 gpm.

Loss of Plant Service Water (Condensate Pumps and Mechanical Vacuum Pump)

The mechanical vacuum pump can operate for up to 2 hours without PSW cooling. After that time, the flow to the mechanical vacuum pump would be approximately 125 gpm. The condensate pump requires approximately 6 gpm of water for cooling to the pump bearings.

Standby Liquid Control

The maximum amount of fire protection water required to refill the SBLC tank, based on the overflow capacity of the tank, is 5150 gallons. In addition, water from the fire protection water system will only be used if the normal supply from the demineralized water transfer system is unavailable.

Spent Fuel Pool and Reactor Vessel Cooling

The level of water in the Spent Fuel Pool and Reactor Vessel can be replenished through the use of a fire hose on the refueling floor by directing the hose stream

either into the Spent Fuel Pool or the open Reactor Vessel. If fire protection water is the only available source of makeup water to restore Spent Fuel Pool Level, and access to the Refueling Floor is denied, then a flow is established for injection of fire protection water through PSW piping to the Spent Fuel Pool. One or more fire pumps may be used to inject fire protection water into the spent fuel pool as directed by the Shift Supervisor.

Alternate RPV Water Level Control

If the normal and alternative sources are unavailable, fire protection water can be injected through the Unit 1 Condensate Transfer Tie-in through 1P11-F038A/B from hydrant 1Y43-F314J, Unit 2 Condensate Transfer Pump Discharge Check Valve 2P11-F027B from hydrant 1Y43-F314K, or directly into the CST through the manway hatch. Two 2½" hose streams may be used to make up condensate storage tank level.

Primary Containment Flooding

If the normal and alternative sources are unavailable, fire protection water can be injected through the Unit 2 Condensate Transfer Pump Discharge Check Valve 2P11-F027B from hydrant 1Y43-F314K, or directly into the CST through the manway hatch. One 2½" hose stream may be used to flood the primary containment or to make up condensate storage tank level.

Cleaning Main Condenser Tubes

Main condenser tube cleaning is performed at an estimated flow rate of 26 gpm and only during outages when the condenser is available for maintenance. This evolution typically lasts 5 days which equates to an estimated total volume of 185,000 gallons. This amount of flow and volume is well within the makeup capacity of the 700 gpm automatic makeup pumps.

Concurrent flow demands from emergency use and fire suppression is highly unlikely. If concurrent flow demands were to occur, the fire protection water system will recover quickly and operators have the option of starting additional fire pumps to ensure appropriate flow and pressure. Each of the three fire pumps is rated for 2500 gpm capacity at 125 psi. The operation of the second fire pump will ensure that the pressure and flow margins are maintained for the fire suppression systems. Simultaneous use with the other non-fire protection emergency uses is not likely.

The fire protection storage tanks water levels are controlled by automatic makeup pumps and level control valves from two deep-wells. Each 700 gpm pump is capable of refilling either tank within 8 hours. Two automatic makeup levels are provided; one at the low-level set point (29 feet) and another at the low-low level set point (28 feet). These water supplies are strained and filtered for normal makeup. In the event of a fire resulting in starting of one of the main fire pumps, the filters may be bypassed to ensure that full make-up capacity is available. Routine surveillance checks by plant operators

using a local tank level indicator verify that the tank level is kept above the minimum level.

Use of fire protection water for DHR, Loss of PSW, Alternate Boron Injection, Fuel Pool Cooling, Alternate RPV Water Level Control, and Primary Containment Flooding is strictly controlled by Emergency Operating, Abnormal Operating, and Infrequent Operations procedures. Use of fire protection water for cleaning main condenser tubes is controlled by the work management process. Personnel utilizing the fire protection water for this purpose are also in contact with the Control Room ensuring the ability to secure the non-fire protection system water demand should a fire occur. Communications are by the plant Public Address (PA) system, plant telephone system, and two-way radio systems.

Following the use of fire protection water for DHR, Loss of PSW, Alternate Boron Injection, Spent Fuel Pool and Reactor Vessel Cooling, Alternate RPV Water Level Control, and Primary Containment Flooding, restoration procedures are included in the applicable Emergency Operating, Abnormal Operating, and Infrequent Operations procedures. These procedures realign the system to normal operating conditions and allow for the fire protection water systems to be returned to standby and fire protection water tanks refilled if required.

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The non-fire use of fire protection water system is strictly controlled by plant operating procedures and the work control process. The non-fire protection flow demands ensure that there is no impact on the ability of the automatic suppression systems to perform their function as hydraulically designed. The ability to isolate the non-fire protection flow ensures there is no impact on the ability of the fire protection water supply to perform its design function of supplying water for both automatic and manual suppression activities with no loss of margin. Therefore, there is no impact on the nuclear safety performance criteria.

The non-fire use of fire protection water has no impact on the radiological release performance criteria. The radioactive release review was performed based on the potential location of radiological concerns and is not dependent on the non-fire use of fire protection water. In addition, these non-fire applications of fire protection water do not add additional radiological materials to any plant area or challenge plant boundaries.

Safety Margin and Defense-in-Depth:

The use of fire protection water for non-fire protection functions requires control room approval. The non-fire protection flow demands ensure that there is no impact on the ability of both automatic and manual fire suppression systems to perform their function as hydraulically designed. The ability to isolate the non-fire protection flow applications or activate additional fire pumps ensures there is no impact on fire suppression efforts. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

The three echelons of defense-in-depth are:

(4) To prevent fires from starting (combustible/hot work controls)
(5) Rapidly detect, control and extinguish fires that do occur, thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans)
(6) Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions)

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

The use of fire protection water for non-fire protection functions does not affect echelons 1 and 3. This evaluation demonstrates that both automatic and manual fire suppression functions are not adversely impacted and will be available when needed. Therefore, the non-fire uses of the fire protection water system do not adversely impact fire protection defense-in-depth.

Since the control room would be aware of a water based fire protection system activation concurrent with fire protection water used for non-fire protection emergency the operator would have the option of starting a second fire pump to ensure adequate flow and pressure is provided to both operations. The automatic makeup to the fire protection storage tanks also ensures there will be an adequate quantity of fire protection water.

Conclusion:

NRC approval is requested for approval of the temporary "off-normal" manual use of the fire protection water supply. The level of risk encountered by maintaining this current practice is acceptable. The performance-based method used for this analysis provides an equivalent level of fire protection to NFPA 805, Section 3.5.16 and:

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

Approval Request 9

NFPA 805 Section 3.6.1 states:

"For all power block buildings, Class III standpipe and hose systems shall be installed in accordance with NFPA 14, Standard for the Installation of Standpipe, Private Hydrant, and Hose Systems."

Section 1-6.1.3 of NFPA 14, 1983 edition, defines Class III standpipe systems as follows:

"For use by either fire departments and those trained in handling heavy hose streams (2½ in. hose) or by the building occupants (1½ in. hose)".

Contrary to this requirement, reactor building hose stations located below the 130' elevation only include 1½ inch hose connections as required for Class II service. A listing of the specific hose stations is provided in Tables 1 and 2 below.

Table 1: Unit 1 Reactor Building Class II Hose Stations

Hose Station	Fire Zone	Description
HS-R01	1205C	Northwest Corner Room - Below el. 130'
HS-R02	1205Z	HPCI Pump Room - Below el. 130'
HS-R03	1205B	Northeast Corner Room - el. 87'
HS-R04	1203B	Southeast Corner Room - el. 87'
HS-R05	1203C	RCIC Southwest Corner Room - el. 87'
HS-R06	1205A	Torus Reactor Building North - Below el. 130'
HS-R07	1205A	Torus Reactor Building North - Below el. 130'
HS-R08	1203A	Torus Reactor Building South - Below el. 130'
HS-R09	1203A	Torus Reactor Building South - Below el. 130'

Table 2: Unit 2 Reactor Building Class II Hose Stations

Hose Station	Fire Zone	Description
2HS-R01	2203B	Northeast Corner Room – el. 87'
2HS-R02	2205B	Southeast Corner Room - Below el. 130'
2HS-R03	2205Z	HPCI Pump Room - Below el. 130'
2HS-R04 (el. 108'-8")	2205C	Southwest Corner Room - Below el. 130'
2HS-R04A (el. 87')	2205C	Southwest Corner Room - Below el. 130'
2HS-R05 (el. 110'-7")	2203C	RCIC Northwest Corner Room - Below el. 130'
2HS-R05A (el. 87')	2203C	Northwest Corner Room - Below el. 130'
2HS-R06	2203A	Torus Reactor Building North - Below el. 130'
2HS-R07	2205A	Torus Reactor Building South - Below el. 130'
2HS-R08	2205A	Torus Reactor Building South - Below el. 130'
2HS-R09	2203A	Torus Reactor Building North - Below el. 130'

SNC requests NRC approval for the current installation of Class II standpipe and hose systems listed above in lieu of Class III standpipe and hose systems as required by NFPA 805 Chapter 3.

Basis for Request:

The bases for the approval request of this deviation are as follows:

- For both Units 1 and 2, all of the Class II standpipe and hose systems are equipped with 75 feet of hose. In all cases, there is sufficient hose to reach all portions of the fire zone. All of these hose stations are fed by an 8-inch main that is routed in a loop configuration near the perimeter walls of the Reactor Buildings. The hose stations and main loop are located below grade and below the level of the fire pump house, which is at elevation 130'. Water pressure is therefore gained by the elevation difference. Although several of these fire zones have potential oil fire scenarios, the current 1½" hose connections are capable of providing adequate flow and pressure to support manual firefighting activities in these fire zones. Therefore, the existing standpipe and hose station configurations are adequate for the hazards in these fire zones.
- For the Class II standpipe and hose systems in Fire Zones 1205B, 1205C, 1203B 2205B, 2205C, and 2203B, an additional Class III hose station is available in the fire zone directly above (Fire Zones 1205F, 1203F, 2205F and 2203F) located near the top of the open stairwell. These additional Class III hose stations provide full redundant manual suppression coverage of these fire zones. These fire zones also have fire extinguishers on the 97' elevation and the 118' elevation.
- Fire Zones 1203C and 2203C have multiple levels of metal grate platform accessed by metal grate open stairwells between the top of the fire zone at

elevation 130' and the bottom of the fire zone at elevation 87'. Class III hose stations are available in the fire zones above, 1203F and 2203F respectively. The Class III hose stations are located approximately 75 feet from the top of the enclosed stairway leading down to Fire Zones 1203C and 2203C. These hose stations therefore provide some, but not full, redundant manual suppression coverage in Fire Zone 1205C. Fire Zones 1203C and 2203C are also protected by automatic wet-pipe sprinkler systems. Water flow alarms will initiate response by the plant fire brigade.

- Fire Zones 1205Z and 2205Z have upper level metal grate platform accessed by a metal grate open stairwell. Additional 1½-inch fire hose stations are located in the adjacent Fire Zones 1205B and 2205B near the doorways leading to Fire Zones 1205Z and 2205Z. These additional hose stations provide redundant coverage to the entire floor level of Fire Zones 1205Z and 2205Z, which contains the majority of the hazards. Fire Zones 1205Z and 2205Z are also protected by automatic wet-pipe sprinkler systems. These fire zones also have fire extinguishers on the 97' elevation and the 118' elevation. The north wall boundary to Fire Zone 2205B is protected by directional spray nozzles. The wet-pipe sprinkler system is expected to control the fire and initiate a water flow alarm both locally and in the MCR to initiate response by the plant fire brigade.
- Unit 1 Hose Stations HS-R06, HS-R07, HS-R08 and HS-R09 are installed to protect the contiguous Torus Fire Zones 1205A and 1203A. Similarly, Unit 2 Hose Stations 2HS-R06, 2HS-R07, 2HS-R08 and 2HS-R09 are installed to protect the contiguous Torus Fire Zones 2205A and 2203A. The spacing and hose lengths are sufficient to reach all portions of these fire zones with almost fully-overlapping redundant coverage. There are also wall-mounted fire extinguishers located next to two of the four hose stations in each unit. These fire zones are partially protected by a common wet-pipe sprinkler system and linear heat detection system at the open boundary between Fire Area 1203 (Fire Zone 1203A) and Fire Area 1205 (Fire Zone 1205A) in Unit 1 and between Fire Area 2203 (Fire Zone 2203A) and Fire Area 2205 (Fire Zone 2205A) in Unit 2. The suppression systems are designed to prevent fire propagation across the open fire area boundaries between 1203A and 1205A in Unit 1 (2203A and 2205A in Unit 2). Actuation of the installed suppression or detection system provides both local and MCR alarms to initiate response by the plant fire brigade.
- The fire brigade members are trained in the use standpipe and hose systems, as well as the use backup capabilities located in adjacent fire areas/fire zones when needed. NMP-TR-426, Fire Training Program, indicates that backup lines (i.e., safety lines) from independent water supplies are used to reinforce and protect personnel in case the initial attack line proves inadequate. NMP-TR-426 also describes the responsibility of the Site Lead Fire Instructor to ensure adequate protection for personnel on training attack lines by always providing backup lines. The fire brigade will properly use the hose stations in adjacent fire zones identified herein which provide additional/redundant hose coverage.

Acceptance Criteria Evaluation:

Nuclear Safety and Radiological Release Performance Criteria:

The installation of Class II standpipe and hose systems in lieu of Class III standpipe and hose systems does not adversely affect nuclear safety capability based on the following:

These fire zones are located at the lowest plant elevations where available water pressure is the highest. The existing standpipe and hose station configurations are adequate for the hazards in these fire zones. Fire suppression and/or detection systems are also installed in the majority of these fire zones. The existing standpipe and hose systems provide adequate fire hose coverage from both inside the fire zone and from hose systems in adjacent fire zones which provide additional/redundant manual suppression coverage. The site fire brigade is trained to use adjacent standpipe and hose systems as backup/safety lines when needed. Therefore, there is no impact on the nuclear safety performance criteria.

The radioactive release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials. The existing standpipe and hose systems were evaluated in the radiological release evaluation, which concluded that potentially contaminated water is contained and smoke is monitored. The Class II standpipe and hose systems for which NRC approval is requested do not add additional radiological materials to the area or challenge system boundaries.

Safety Margin and Defense-in-Depth:

The installation of Class II standpipe hose connections is considered adequate for the hazards in the areas protected. Given the technical bases listed above, particularly the redundant overlapping hose coverage from adjacent fire zones, safety margin is not considered to be adversely affected. Therefore, the safety margin inherent in this analysis for a fire event has been preserved.

The three echelons of defense-in-depth are:

(1) To prevent fires from starting (combustible/hot work controls)
(2) Rapidly detect, control and extinguish fires that do occur, thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans)
(3) Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions)

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

The currently installed configuration of standpipe and hose systems does not affect the prevention of fires from starting and therefore has no impact on echelon 1.

Based on the mitigating factors and redundancies, the currently installed configuration

of standpipe and hose systems does not compromise manual fire suppression functions, automatic fire suppression functions, or post-fire safe shutdown capability. Echelon 2 is therefore not adversely impacted. These mitigating factors and redundancies include:

- The evaluated Class II standpipe and hose systems have sufficient hose to reach all portions of the fire zone in which they are located.
- There is either full or partial overlapping redundant hose coverage from adjacent fire zones. Areas that have only partial overlapping redundant coverage are also protected by automatic wet-pipe sprinkler systems which are expected to limit the fire size and initiate fire brigade response.
- The fire brigade is trained to use standpipe and hose systems located in adjacent fire areas/fire zones.
- There is adequate available flow and pressure from the mains feeding the evaluated standpipe and hose systems.

Based on the mitigating factors and redundancies described above, the currently installed configuration of Class II standpipe and hose systems provides an adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed. Echelon 3 is therefore not adversely impacted.

Conclusion:

NRC approval is requested for the use of Class II standpipe and hose systems in the plant areas described above. The level of risk associated with maintaining this configuration has been determined to be acceptable. Consistent with the performance-based approach utilized to evaluate a variance from the requirements of NFPA 805 Chapter 3, this engineering analysis has determined that the currently installed configuration of standpipe and hose systems:

(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;

(B) Maintains safety margins; and

(C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire shutdown capability).

M. License Condition Changes

35 Pages Attached

Replace the current HNP fire protection license conditions 2.C(3) for Unit 1 and 2.C(3)a for Unit 2 with the standard license condition based on Regulatory Position 3.1 of RG 1.205.

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Southern Nuclear Operating Company 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 licensee amendment request dated _____ , supplemented by letters dated _____, and as approved in the safety evaluation report dated _____. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, 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) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

(a) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. 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 plant change evaluation.

(b) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-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 plant change evaluation.

Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an

NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the component, system, procedure, or physical arrangement functionality using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the component, system, procedure, or physical arrangement functionality using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are:

- Fire Alarm and Detection Systems (Section 3.8);
- Automatic and Manual Water-Based Fire Suppression Systems (Section 3.9);
- Gaseous Fire Suppression Systems (Section 3.10); and,
- Passive Fire Protection Features (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

(2) Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in NRC safety evaluation report dated _____ to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in (2) above.

(2) The licensee shall implement the following modifications to its facility to complete transition to full compliance with 10 CFR 50.48(c) by _____:
See plant specific list of modifications identified in Attachment S.

(3) The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

==========

License conditions 2.C(3) and 2.C(3)a shall be superseded:

Unit 1

(3) Fire Protection

Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained in the updated Fire Hazards Analysis and Fire Protection Program for the Edwin I. Hatch Nuclear Plant Units 1 and 2, which was originally submitted by letter from GPC to the Commission dated July 22, 1986. Southern Nuclear may make changes to the 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.

Unit 2

(a) Fire Protection

Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained in the updated Fire Hazards Analysis and Fire Protection Program for the Edwin I. Hatch Nuclear Plant Units 1 and 2, which was originally submitted by letter from GPC to the Commission dated July 22, 1986. Southern Nuclear may make changes to the 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.

It is SNC's understanding that implicit in the revocation of this license condition, all prior FPP SERs and commitments have been superseded in their entirety by the revised license condition.

No other license conditions need to be revised or superseded.

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 FPP which meets the requirements in 10 CFR 50.48(a) and 50.48(c):

A review was conducted of the HNP Renewed Operating License DPR-57 and NPF-5, by licensing staff and the NFPA 805 Transition Team. The review was performed by reading the Operating license and performing electronic searches. Outstanding LARs that have been submitted to the NRC were also reviewed for potential impact on the license conditions.

FACILITY OPERATING LICENSE– MARKUPS

6 Pages Attached

for sample analysis or instrument calibration, or associated with radioactive apparatus or components

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain, and is subject to, the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not in excess of 2,804 megawatts thermal.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 287, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance Requirement (SR) contained in the Technical Specifications and listed below, is not required to be performed immediately upon implementation of Amendment No. 195. The SR listed below shall be successfully demonstrated before the time and condition specified:

SR 3.8.1.18 shall be successfully demonstrated at its next regularly scheduled performance.

(3) Fire Protection

Insert Attachment

~~Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained in the updated Fire Hazards Analysis and Fire Protection Program for the Edwin I. Hatch Nuclear Plant, Units 1 and 2, which was originally submitted by letter dated July 22, 1986. Southern Nuclear may make changes to the fire protection program without prior Commission approval only if the changes~~

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~~would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.~~

(4.a) Physical Protection

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," with revisions submitted through May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 265, as supplemented by a change approved by License Amendment No. 274.

(4.b) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre- staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(4.c) The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

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(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain, and is subject to, the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions[2] specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not in excess of 2,804 megawatts thermal, in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 232, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission.

(a) Fire Protection

Insert Attachment

~~Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained~~

2 The original licensee authorized to possess, use, and operate the facility was Georgia Power Company (GPC). Consequently, certain historical references to GPC remain in certain license conditions.

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~~in the updated Fire Hazards Analysis and Fire Protection Program for the Edwin I. Hatch Nuclear Plant Units 1 and 2, which was originally submitted by letter from GPC to the Commission dated July 22, 1986. Southern Nuclear may make changes to the fire protection program without prior Commission approval only if the changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.~~

(b.1) Physical Protection

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," with revisions submitted through May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 209, as supplemented by a change approved by License Amendment No. 219.

(b.2) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

ATTACHMENT

Insert the following in license conditions 2.C(3) and 2.C(3)a as follows:

Southern Nuclear Operating Company 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 licensee amendment request dated _____ , supplemented by letters dated _____, and as approved in the safety evaluation report dated _____. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, 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) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

(a) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. 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 plant change evaluation.

(b) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-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 plant change evaluation.

Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding

technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the component, system, procedure, or physical arrangement functionality using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the component, system, procedure, or physical arrangement functionality using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are:

- Fire Alarm and Detection Systems (Section 3.8);
- Automatic and Manual Water-Based Fire Suppression Systems (Section 3.9);
- Gaseous Fire Suppression Systems (Section 3.10); and,
- Passive Fire Protection Features (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

(2) Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in NRC safety evaluation report dated _____ to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in (2) above.

(2) The licensee shall implement the modifications to its facility to complete transition to full compliance with 10 CFR 50.48(c) by _____ as described in Attachment S, Table S-2, "Plant Modifications Committed," of SNC letter NL-18-0282. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications

(3) The licensee shall implement the items as listed in Attachment S, Table S-3, "Implementation Items," of SNC letter NL-18-0282, dated _____, within 365 days after NRC approval.

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23 Pages Attached

for sample analysis or instrument calibration, or associated with radioactive apparatus or components

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain, and is subject to, the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not in excess of 2,804 megawatts thermal.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 288, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance Requirement (SR) contained in the Technical Specifications and listed below, is not required to be performed immediately upon implementation of Amendment No. 195. The SR listed below shall be successfully demonstrated before the time and condition specified:

SR 3.8.1.18 shall be successfully demonstrated at its next regularly scheduled performance.

(3) Fire Protection

Southern Nuclear Operating Company 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 licensee amendment request dated _____ , supplemented by letters dated _____, and as approved in the safety evaluation report dated _____. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical

specification, license condition or requirement would require prior NRC approval, 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) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. 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 plant change evaluation.

2) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-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 plant change evaluation.

(b) Other Changes that May Be Made Without Prior NRC Approval

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1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the component, system, procedure, or physical arrangement functionality using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the component, system, procedure, or physical arrangement functionality using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are:

- Fire Alarm and Detection Systems (Section 3.8);
- Automatic and Manual Water-Based Fire Suppression Systems (Section 3.9);
- Gaseous Fire Suppression Systems (Section 3.10); and,
- Passive Fire Protection Features (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA-805.

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Amendment No.

2) Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in NRC safety evaluation report dated _____ to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in (2) above.

2) The licensee shall implement the modifications to its facility to complete transition to full compliance with 10 CFR 50.48(c) by _____ as described in Attachment S, Table S-2, "Plant Modifications Committed," of SNC letter NL-18-0282. The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

3) The licensee shall implement the items as listed in Attachment S, Table S-3, "Implementation Items," of SNC letter NL-18-0282, dated _____, within 365 days after NRC approval.

(4.a) Physical Protection

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements

revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," with revisions submitted through May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 265, as supplemented by a change approved by License Amendment No. 274.

(4.b) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre- staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(4.c) The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

(5) FSAR Supplement

The licensee's Final Safety Analysis Report Supplement, dated September 5, 2001, shall be included in the next Updated Final Safety Evaluation Analysis Report update, required by 10 CFR 50.71(e).

(6) Safety Analysis Report

The licensee's Final Safety Analysis Report Supplement, dated September 5, 2001, submitted pursuant to 10 CFR 54.21(d), describes certain future inspection activities to be completed before the period of extended operations begins. The licensee shall complete those activities no later than August 6, 2014.

(7) Integrated Surveillance Program

The licensee shall implement a staff-approved reactor vessel integrated surveillance program for the extended period of operation which satisfies the requirements of 10 CFR Part 54. Such a program will be implemented through a staff-approved Boiling Water Reactor Vessel and Internals Project program or through a staff-approved plant-specific program. The plant-specific program, if needed, will be developed in a manner that is consistent with other aging management programs, will include consideration of the 10 program attributes utilized for other aging management programs, and will provide a technical justification for any

program attribute not covered by the plant-specific surveillance material testing program. The plant-specific program, if needed, will include the following actions:

(a) Capsules will periodically be removed to determine the rate of embrittlement.

(b) Capsules will be removed at neutron fluence levels that provide relevant data for assessing the integrity of the Plant Hatch, Unit 1 reactor pressure vessel (in particular, for the determination of reactor pressure vessel pressure-temperature limits through the period of extended operation).

(c) Capsules will contain material to monitor the impact of irradiation on the Plant Hatch Unit 1 reactor pressure vessel and will contain dosimetry to monitor neutron fluence.

Before the renewal term begins, the licensee will notify the NRC of its decision to implement the integrated surveillance program or a plant-specific program, and provide the appropriate revisions to the Updated Final Safety Analysis Report Supplement summary descriptions of the vessel surveillance material testing program.

(8) Design Bases Accident Radiological Consequences Analyses

Southern Nuclear is authorized to credit administering potassium iodide to reduce the 30 day post-accident thyroid radiological dose to the operators in the main control room until May 31, 2012. Should design basis changes be completed rendering the crediting of potassium iodide no longer necessary prior to May 31, 2012, Southern Nuclear will remove the crediting of potassium iodide from the design basis accident radiological consequences analyses (reference Unit 2 FSAR paragraph 15.3.3.4.2.2) in the next Updated Final Safety Analysis Report as required by 10 CFR 50.71(e).

(9) Alternative Source Term

1) Southern Nuclear Operating Company, Inc (SNC, the licensee) shall complete actions by April 30, 2010, as described in SNC's letters dated October 18, 2007, and March 13, 2008, to complete the design modifications to the HNP turbine building ventilation exhaust systems. Specifically, the HNP Units 1 and 2 turbine building exhaust fans shall be capable of being manually switched over from normally operating power supplies, to a Class 1E circuit that will be isolated by an appropriately rated safety related, environmentally and seismically qualified circuit breaker. For further protection and isolation, the licensee shall also use fuses

that will be located in a seismically qualified manual transfer switch housing. The aforementioned circuit breaker and fuses shall be adequately coordinated with the upstream load center breaker over the entire range. These devices shall be adequately rated to prevent adverse effects of a fault to the rest of the distribution system.

2) SNC shall implement modifications by May 31, 2010, as described in Enclosure 1, section 2.7.3.2, of the LAR and section 5.7 of SNC's letter dated February 25, 2008 (NL 08-0175) to modify the design for the air supply to the turbine building exhaust ventilation dampers, such that operating air to the dampers will be supplied from a non-interruptible instrument air source to eliminate single failure point vulnerability to loss of system/instrument air.

3) SNC shall complete actions by May 31, 2010, as described in SNC's letter dated February 25, 2008 (NL-08-0175) to install and implement the capability for Standby Liquid Control System hand switch jumpers for HNP Units 1 and 2.

4) SNC shall complete actions by May 31, 2012 for HNP Unit 1, as described in SNC's letters dated February 25, 2008 (NL-08-0175) and July 2, 2008 (NL-08-1022), to modify the following Main Steam Isolation Valve alternate leakage treatment boundary valves, such that they can be closed in the event of a loss of offsite power without requiring local operation:

1N38-F101A, 1N38-F101B, 1N33-F012, 1N33-F013

5) SNC shall implement actions by May 31, 2010, as described in SNC's letter dated February 27, 2008, to assure that temperature switches which monitor charcoal bed temperature meet the environmental qualification requirements of 10 CFR 50.49.

(10) TSTF-448, Control Room Habitability

Upon implementation of the Amendments adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.4.4, in accordance with TS 5.5.14.c.(i), the assessment of CRE habitability as required by Specification 5.5.14.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.14.d, shall be considered met. Following implementation:

a. The first performance of SR 3.7.4.4, in accordance with Specification 5.5.14.c.(i), shall be within the next 18 months.

b. The first performance of the periodic assessment of CRE habitability, Specification 5.5.14.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, of the next successful tracer gas test.

c. The first performance of the periodic measurement of CRE pressure, Specification 5.5.14.d, shall be within 24 months, plus the 6 months allowed by SR 3.0.2, from the date of the most recent successful pressure measurement test.

(11) Degraded Voltage Protection

SNC shall implement the Degraded Voltage modifications to eliminate the manual actions in lieu of automatic degraded voltage protection to assure adequate voltage to safety-related equipment during design basis events by completion of the Unit 1 2020 Spring Outage, U1R29.

D. Southern Nuclear shall not market or broker power or energy from Edwin I. Hatch Nuclear Plant, Unit 1.

3. This renewed license is effective as of the date of issuance and shall expire at midnight, August 6, 2034.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications
Appendix B – Environmental Protection Plan

Date of Issuance: January 15, 2002

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(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain, and is subject to, the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions[2] specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not in excess of 2,804 megawatts thermal, in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 233, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission.

(a) Fire Protection

Southern Nuclear Operating Company 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 licensee amendment request dated _____ , supplemented by letters dated _____, and as approved in the safety evaluation report dated _____. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, 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) and 10 CFR 50.48(c),

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the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

1) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

a) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. 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 plant change evaluation.

b) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-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 plant change evaluation.

2) Other Changes that May Be Made Without Prior NRC Approval

a) Changes to NFPA 805, Chapter 3, fundamental Fire Protection Program

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Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the component, system, procedure, or physical arrangement functionality using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the component, system, procedure, or physical arrangement functionality using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are:

- Fire Alarm and Detection Systems (Section 3.8);
- Automatic and Manual Water-Based Fire Suppression Systems (Section 3.9);
- Gaseous Fire Suppression Systems (Section 3.10); and,
- Passive Fire Protection Features (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA-805.

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b) Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in NRC safety evaluation report dated _____ to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

3) Transition License Conditions

a) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in (2) above.

b) The licensee shall implement the modifications to its facility to complete transition to full compliance with 10 CFR 50.48(c) by _____ as described in Attachment S, Table S-2, "Plant Modifications Committed," of SNC letter NL-18-0282. The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

c) The licensee shall implement the items as listed in Attachment S, Table S-3, "Implementation Items," of SNC letter NL-18-0282, dated _____, within 365 days after NRC approval.

(b.1) Physical Protection

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including

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amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," with revisions submitted through May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 209, as supplemented by a change approved by License Amendment No. 219.

(b.2) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

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(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(b.3) The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

(c) FSAR Supplement

The licensee's Final Safety Analysis Report Supplement dated September 5, 2001, shall be included in the next Updated Final Safety Analysis Report update, required by 10 CFR 50.71(e).

(d) Safety Analysis Report

The licensee's Final Safety Analysis Report Supplement dated September 5, 2001, submitted pursuant to 10 CFR 54.21(d), describes certain future inspection activities to be completed before the period of extended operations begins. The licensee shall complete those activities no later than June 13, 2018.

(e) Integrated Surveillance Program

The licensee shall implement a staff-approved reactor vessel integrated surveillance program for the extended period of operation which satisfies the requirements of 10 CFR Part 54. Such a program will be implemented through a staff-approved

Boiling Water Reactor Vessel Internals Project program or through a staff-approved plant-specific program. The plant-specific program, if needed, will be developed in a manner consistent with other aging management programs, will include consideration of the 10 program attributes utilized for other aging management programs, and will provide a technical justification for any program attribute not covered by the plant-specific surveillance material testing program. The plant-specific program, if needed, will include the following actions:

i. Capsules will periodically be removed to determine the rate of embrittlement.

ii. Capsules will be removed at neutron fluence levels that provide relevant data for assessing the integrity of the Plant Hatch Unit 2 reactor pressure vessel (in particular, for the determination of reactor pressure vessel pressure-temperature limits through the period of extended operation).

iii. Capsules will contain material to monitor the impact of irradiation on the Plant Hatch Unit 2 reactor pressure vessel and will contain dosimetry to monitor neutron fluence.

Before the renewal term begins, the licensee will notify the NRC of its decision to implement the integrated surveillance program or a plant-specific program, and provide the appropriate revisions to the Updated Final Safety Analysis Report Supplement summary descriptions of the vessel surveillance material testing program.

(f) Design Bases Accident Radiological Consequences Analyses

Southern Nuclear is authorized to credit administering potassium iodide to reduce the 30 day post-accident thyroid radiological dose to the operators in the main control room until May 31, 2011. Should design basis changes be completed rendering the crediting of potassium iodide no longer necessary prior to May 31, 2011, Southern Nuclear will remove the crediting of potassium iodide from the design basis accident radiological consequences analyses (reference Unit 2 FSAR paragraph 15.3.3.4.2.2) in the next Updated Final Safety Analysis Report update as required by 10 CFR 50.71(e).

(g) Alternative Source Term

i) Southern Nuclear Operating Company, Inc (SNC, the licensee) shall complete actions by April 30, 2010, as described in SNC's letters dated October 18, 2007, and March 13, 2008, to complete the design modifications to the HNP turbine building ventilation exhaust systems. Specifically, the HNP Units 1 and 2 turbine building exhaust fans shall be capable of being

manually switched over from normally operating power supplies, to a Class - 1E circuit that will be isolated by an appropriately rated safety related, environmentally and seismically qualified circuit breaker. For further protection and isolation, the licensee shall also use fuses that will be located in a seismically qualified manual transfer switch housing. The aforementioned circuit breaker and fuses shall be adequately coordinated with the upstream load center breaker over the entire range. These devices shall be adequately rated to prevent adverse effects of a fault to the rest of the distribution system.

ii) SNC shall implement modifications by May 31, 2010, as described in Enclosure 1, section 2.7.3.2, of the LAR and section 5.7 of SNC's letter dated February 25, 2008, (NL 08-0175) to modify the design for the air supply to the turbine building exhaust ventilation dampers, such that operating air to the dampers will be supplied from a non-interruptible instrument air source to eliminate single failure point vulnerability to loss of system/instrument air.

iii) SNC shall complete actions by May 31, 2010, as described in SNC's letter dated February 25, 2008 (NL-08-0175) to install and implement the capability for Standby Liquid Control System hand switch jumpers for HNP Units 1 and 2.

iv) SNC shall complete actions by May 31, 2011, for HNP Unit 2, as described in SNC's letters dated February 25, 2008 (NL-08-0175) and July 2, 2008 (NL-08-1022), to modify the following Main Steam Isolation Valve alternate leakage treatment boundary valves, such that they can be closed in the event of a loss of offsite power without requiring local operation:

2N11-F004A, 2N11-F004B, 2N33-F003, 2N33-F004

v) SNC shall implement actions by May 31, 2010, as described in SNC's letter dated February 27, 2008, to assure that temperature switches which monitor charcoal bed temperature meet the environmental qualification requirements of 10 CFR 50.49.

(h) TSTF-448 Control Room Habitability

Upon implementation of the Amendments adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.4.4, in accordance with TS 5.5.14.c.(i), the assessment of CRE habitability as required by Specification 5.5.14.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.14.d, shall be considered met. following implementation:

i) The first performance of SR 3.7.4.4, in accordance with Specification 5.5.14.c.(i), shall be within the next 18 months.

ii) The first performance of the periodic assessment of CRE habitability, Specification 5.5.14.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, of the next successful tracer gas test.

iii) The first performance of the periodic measurement of CRE pressure, Specification 5.5.14.d, shall be within 24 months, plus the 6 months allowed by SR 3.0.2, from the date of the most recent successful pressure measurement test.

(i) Degraded Voltage Protection

SNC shall implement the Degraded Voltage modifications to eliminate the manual actions in lieu of automatic degraded voltage protection to assure adequate voltage to safety-related equipment during design basis events by completion of the Unit 2 2019 Spring Outage, U2R25.

D. This renewed license is subject to the following antitrust conditions:

(1) As used herein:

(a) "Entity" means any financially responsible person, private or public corporation, municipality, county, cooperative, association, joint stock association or business trust, owning, operating or proposing to own or operate equipment or facilities within the state of Georgia (other than Chatham, Effingham, Fannin, Towns and Union Counties) for

the generation, transmission, or distribution of electricity, provided that, except for municipalities, counties, or rural electric cooperatives, "entity" is restricted to those which are or will be public utilities under the laws of the State of Georgia or under the laws of the United States, and are or will be providing retail electric service under a contract or rate schedule on file with and subject to the regulation of the Public Service Commission of the State of Georgia or any regulatory agency of the United States, and provided further, that as to municipalities, counties, or rural electric cooperatives, "entity" is restricted to those which provide electricity to the public at retail within the State of Georgia (other than Chatham, Effingham, Fannin, Towns and Union Counties) or to responsible and legally qualified organizations of such municipalities, counties, and/or cooperatives in the State of Georgia (other than Chatham, Effingham, Fannin, Towns and Union Counties) to the extent they may bind their members.

(b) "Power Company" means Georgia Power Company, any successor, assignee of this license, or assignee of all or substantially all of Georgia Power Company's assets, and any affiliate or subsidiary of Georgia Power Company to the extent it engages in the ownership of any bulk power supply generation or transmission resource in the State of Georgia (but specifically not including (1) flood rights and other land rights acquired in the State of Georgia incidental to hydroelectric generation facilities located in another state and (2) facilities located west of the thread of the stream on that part of the Chattahoochee River serving as the boundary between the states of Georgia and Alabama).

(2) Power Company recognizes that it is often in the public interest for those engaging in bulk power supply and purchases to interconnect, coordinate for reliability and economy, and engage in bulk power supply transactions in order to increase interconnected system reliability and reduce the costs of electric power. Such arrangements must provide for Power Company's costs (including a reasonable return) in connection therewith and allow other participating entities full access to the benefits available from interconnected bulk power supply operations and must provide net benefits to Power Company. In entering into such arrangements neither Power Company nor any other participant should be required to violate the principles of sound engineering practice or forego a reasonably contemporaneous alternative arrangement with another, developed in good faith in arms length negotiations (but not including arrangements between Power Company and its affiliates or subsidiaries which impair entities' rights hereunder more than they would be impaired were such arrangements made in good faith between Power Company a non-affiliate or non-subsidiary) which affords it greater benefits. Any such arrangement must provide for adequate notice and joint planning procedures consistent with sound engineering practice, and must relieve Power Company from obligations undertaken by it in the event such procedures are not followed by any participating entity.

Power Company recognizes that each entity may acquire some or all of its bulk power supply from sources other than Power Company.

In the implementation of the obligations stated in the succeeding paragraphs, Power Company and entities shall act in accordance with the foregoing principles, and these principles are conditions to each of Power Company's obligations herein undertaken.

(3) Power Company shall interconnect with any entity which provides, or which has undertaken firm contractual obligations to provide, some or all of its bulk power supply from source other than Power Company on terms to be included in an interconnection agreement which shall provide for appropriate allocation of the costs of interconnection facilities; provided, however, that if an entity undertakes to negotiate such a firm contractual obligation, the Power Company shall, in good faith, negotiate with such entity concerning any proposed interconnection. Such interconnection agreement shall provide, without undue preference or discrimination, for the following, among other things, insofar as consistent with the operating necessities of Power Company's and any participating entity's systems:

(a) maintenance and coordination of reserves, including, where appropriate, the purchase and sale thereof,

(b) emergency support,

(c) maintenance support,

(d) economy energy exchanges,

(e) purchase and sale of firm and non-firm capacity and energy,

(f) economic dispatch of power resources within the State of Georgia, provided, however, that in no event shall such arrangements impose a higher percentage of reserve requirements on the participating entity than that maintained by Power Company for similar resources.

(4) Power Company shall sell full requirements power to any entity. Power Company shall sell partial requirements power to any entity. Such sales shall be made pursuant to rates on file with the Federal Power Commission, or any successor regulatory agency, and subject to reasonable terms and conditions.

(5) (a) Power Company shall transmit ("transmission service") bulk power over its system to any entity or entities with which it is interconnected, pursuant to rate schedules on file with the Federal Power Commission which will fully compensate Power Company for the use of its system, to the extent that such arrangements can be accommodated from a functional engineering standpoint and to the extent that Power Company has surplus line capacity or reasonably available funds to finance new construction for this purpose. To the extent the entity or entities are able, they shall reciprocally provide transmission service to Power Company. Transmission service will be provided under this subparagraph for the delivery of power to an entity for its or its members' consumption and retail distribution or for casual resale to another entity for (1) its consumption or (2) its retail distribution. Nothing contained herein shall require the Power Company to transmit bulk power so as to have the effect of making the Tennessee Valley Authority ("TVA") or its distributors, directly or indirectly, a source of power supply outside the area determined by the TVA Board of Directors by resolution of May 16, 1966 to be the area for which the TVA or its distributors were the primary source of power supply on July 1, 1957, the date specified in the Revenue Bond Act of 1959, 16 USC 831 n-4.

(b) Power Company shall transmit over its system from any entity or entities with which it is interconnected, pursuant to rate schedules on file with the Federal Power Commission which will fully compensate Power Company for the use of its system, bulk power which results from any such entity having excess capacity available from self-owned generating resources in the State of Georgia, to the extent such excess necessarily results from economic unit sizing or from failure to forecast load accurately or from such generating resources becoming operational earlier than the planned in-service date, to the extent that such arrangements can be accommodated from a functional engineering standpoint, and to the extent Power Company has surplus line capacity available.

(6) Upon request, Power Company shall provide service to any entity purchasing partial requirements service, full requirements service or transmission service from Power Company at a delivery voltage appropriate for loads served by such entity, commensurate with Power Company's available transmission facilities. Sales of such service shall be made pursuant to rates on file with the Federal Power Commission or any successor regulatory agency, and subject to reasonable terms and conditions.

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(7) Upon reasonable notice, Power Company shall grant any entity the opportunity to purchase an appropriate share in the ownership of, or, at the option of the entity, to purchase an appropriate share of unit power from each of the following nuclear generating units at Power Company's costs, to the extent the same are constructed and operated: Hatch 2, Vogtle 1, Vogtle 2, and any other nuclear generating unit constructed by Power Company in the State of Georgia which, in the application filed with USAEC or its successor agency, is scheduled for commercial operation prior to January 1, 1989.

An entity's request for a share must have regard for the economic size of such nuclear unit(s), for the entity's load size, growth and characteristics, and for demands upon Power Company's system from other entities and Power Company's retail customers, all in accordance with sound engineering practice. Executory agreements to accomplish the foregoing shall contain provisions reasonably specified by Power Company requiring the entity to consummate and pay for such purchase by an early date or dates certain. For purposes of this provision, "unit power" shall mean capacity and associated energy from a specified generating unit.

(8) Southern Nuclear shall not market or broker power or energy from Edwin I. Hatch Nuclear Plant, Unit 2. Georgia Power Company shall continue to be responsible for compliance with the obligations imposed on it in its antitrust license conditions. Georgia Power Company is responsible and accountable for the actions of Southern Nuclear, to the extent that Southern Nuclear's actions may, in any way, contravene the existing antitrust license conditions.

(9) To effect the foregoing conditions, the following steps shall be taken:

(a) Power Company shall file with the appropriate regulatory authorities and thereafter maintain in force as needed an appropriate transmission tariff available to any entity;

(b) Power Company shall file with the appropriate regulatory authorities and thereafter maintain in force as needed an appropriate partial requirements tariff available to any entity; Power Company shall have its liability limited to the partial requirements service actually contracted for and the entity shall be made responsible for the security of the bulk power supply resources acquired by the entity from sources other than the Power Company;

(c) Power Company shall amend the general terms and conditions of its current Federal Power Commission tariff and thereafter maintain in force as needed provisions to enable any entity to receive bulk power at transmission voltage at appropriate rates;

(d) Power Company shall not have the unilateral right to defeat the intended access by each entity to alternative sources of bulk power supply provided by the conditions to this license; but Power Company shall retain the right to seek regulatory approval of changes in its tariffs to the end that it be adequately compensated for services it provides, specifically including, but not limited to, the provisions of Section 205 of the Federal Power Act;

(e) Power Company shall use its best efforts to amend any outstanding contract to which it is a party that contains provisions which are inconsistent with the conditions of this license;

(f) Power Company affirms that no consents are or will become necessary from Power Company's parent, affiliates or subsidiaries to enable Power Company to carry out its obligations hereunder or to enable the entities to enjoy their rights hereunder;

(g) All provisions of these conditions shall be subject to and implemented in accordance with the laws of the United States and of the State of Georgia, as applicable, and with rules, regulations, and orders of agencies of both, as applicable.

3. This renewed license is effective as of the date of issuance and shall expire at midnight, June 13, 2038.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications
Appendix B – Environmental Protection Plan

Date of Issuance: January 15, 2002

Renewed License No. NPF-5
Amendment No.

N. Technical Specification Changes

7 Pages Attached

Delete the following Technical Specification for Units 1 and 2:

- Section 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

  d. Fire Protection Program implementation; and

No other Technical Specifications need to be revised or deleted.

HNP implemented the following process for determining that this is the only Technical Specification required to be revised or deleted to implement the new FPP which meets the requirements in 10 CFR 50.48(a) and 50.48(c).

- A review was conducted of the FNP Technical Specifications by SNC licensing and NFPA 805 Transition Team. 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 condition.

HNP determined that these changes to the Technical Specifications are adequate for adoption of the new fire protection licensing basis, for the following reason:

- The requirement for establishing, implementing, and maintaining FP procedures is contained in the regulation (10 CFR 50.48(a) and 50.48(c) NFPA 805 Chapter 3).

The markups and retypes follow.

TECHNICAL SPECIFICATIONS – MARKUPS

2 Pages Attached

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1;

c. Quality assurance for effluent and environmental monitoring;

~~d. Fire Protection Program implementation; and~~ DELETED

e. All programs and manuals specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1;

c. Quality assurance for effluent and environmental monitoring;

~~d. Fire Protection Program implementation; and~~ DELETED

e. All programs and manuals specified in Specification 5.5.

TECHNICAL SPECIFICATIONS – RETYPE

2 Pages Attached

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1;

c. Quality assurance for effluent and environmental monitoring;

d. DELETED

e. All programs and manuals specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1;

c. Quality assurance for effluent and environmental monitoring;

d. DELETED

e. All programs and manuals specified in Specification 5.5.

O. Orders and Exemptions

4 Pages Attached

Exemptions

Rescind the following exemptions granted against 10 CFR 50, Appendix R

- Exemption from 10 CFR 50 Appendix R, Section III.G.3, for Lack of an Area-Wide Automatic Fire Suppression System in the Control Room.
  - Main Control Room – Fire Area 0024

  This exemption was provided in a safety evaluation dated November 16, 1981

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of Automatic Fire Suppression Systems and 3-Hour Fire Rated Barriers in the Unit 1 4160V Transformer Room and Unit 1 West 600V Switchgear Room
  - Unit 1 West 600V Switchgear Room – Fire Area 1016
  - Unit 1 4160V Transformer Room – Fire Area 1019

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of 3-Hour Fire Rated Barriers in the Unit 1 and Unit 2 Switchgear Rooms, Unit 2 Transformer Room, and Control Building Working Floor
  - Unit 1 Control Building Working Floor El. 112 Feet – Fire Area 0001
  - Unit 1 East 600V Switchgear Room – Fire Area 1017
  - Unit 1 West DC Switchgear Room – Fire Area 1018
  - Unit 1 East DC Switchgear Room – Fire Area 1020
  - Unit 2 West 600V Switchgear Room – Fire Area 2016
  - Unit 2 East 600V Switchgear Room – Fire Area 2017
  - Unit 2 West DC Switchgear Room – Fire Area 2018
  - Unit 2 4160V Transformer Room – Fire Area 2019
  - Unit 2 East DC Switchgear Room – Fire Area 2020

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of a 3-Hour Fire Rated Barrier or 1-Hour Barrier and Area-Wide Automatic Fire Detection and Suppression in the Unit 1 and Unit 2 Reactor Buildings
  - Unit 1 Reactor Building South of Column Line R7 – Fire Area 1203
  - Unit 1 Reactor Building North of Column Line R7 – Fire Area 1205
  - Unit 2 Reactor Building North of Column Line R19 – Fire Area 2203
  - Unit 2 Reactor Building South of Column Line R19 – Fire Area 2205

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of an Area-Wide Automatic Fire Suppression System in the Unit 2 Control Building Health Physics Area
  - Unit 2 Control Building Switchgear Hallway – Fire Area 0014

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of 3-Hour Fire Rated Barriers in the Unit 2 Control Building Switchgear Hallway
  - Unit 2 Switchgear Access Hallway – Fire Area 2014

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of a 3-Hour Fire Rated Barrier in the Station Battery Rooms
  - Unit 1 Station Battery Room 1A – Fire Area 1004
  - Unit 1 Station Battery Room 1B – Fire Area 1005
  - Unit 2 Station Battery Room 2A – Fire Area 2004
  - Unit 2 Station Battery Room 2B – Fire Area 2005

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of a 3-Hour Fire Rated Barrier in the Unit 2 Turbine Building Condenser Bay
  - Unit 2 Turbine Building Condenser Bay – Fire Zone 2101K

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of 3-Hour Fire Rated Barriers in the Turbine Buildings, East Cableway, and West Cableway
  - Unit 1 East Cableway – Fire Area 1104
  - Unit 2 Turbine Building West Cableway – Fire Zone 2101I
  - Unit 2 Turbine Building East Cableway – Fire Area 2104

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of an Area-Wide Suppression System in Diesel Building Switchgear Room 2G
  - Diesel Generator Building Switchgear Room 2G – Fire Area 2409

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of 3-Hour Fire Rated Barriers or 1-Hour Barriers and Area-Wide Automatic Fire Detection and Suppression in the Control Building Common Corridor
  - Common Control Building Corridor – Fire Area 0014

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of an Area-Wide Automatic Fire Suppression System in the Intake Structure
  - Intake Structure – Fire Area 0501

This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.G.2, for Lack of a 3-Hour Fire Rated Barrier in the Control Building East Corridor
  - Control Building East Corridor and HP Cold Lab – Fire Area 0007
  - Unit 1 Turbine Building – Fire Area 1101
  - Unit 2 Turbine Building – Fire Area 2101
  - Unit 2 East Cableway – Fire Area 2104

  This exemption was provided in a safety evaluation dated April 18, 1984

- Exemption from 10 CFR 50 Appendix R, Section III.J, for Lack of Emergency Lighting
  - Main Control Room – Fire Zone 0024C
  - All Yard Fire Areas

  This exemption was provided in a safety evaluation dated January 2, 1987

- Exemption from 10 CFR 50 Appendix R, Section III.G.2.c, for Lack of 1-Hour Fire Rated Barriers in the Unit 1 and Unit 2 Reactor Buildings
  - Unit 1 Reactor Building South of Column Line R7 – Fire Area 1203
  - Unit 1 Reactor Building North of Column Line R7 – Fire Area 1205
  - Unit 2 Reactor Building North of Column Line R19 – Fire Area 2203
  - Unit 2 Reactor Building South of Column Line R19 – Fire Area 2205

  This exemption was provided in a safety evaluation dated January 2, 1987

- Exemption from 10 CFR 50 Appendix R, Section III.G.1.a, for RHR and RCIC Pump Repairs
  - Control Complex – Fire Area 0024

  This exemption was provided in a safety evaluation dated January 2, 1987

- Exemption from 10 CFR 50 Appendix R, Sections III.G.2.a and III.G.2.b, for Lack of 1-Hour Fire Rated Barriers in the Unit 1 and 2 Reactor Buildings
  - Unit 1 Reactor Building North of Column Line R7 – Fire Area 1205
  - Unit 2 Reactor Building South of Column Line R19 – Fire Area 2205

  This exemption was provided in a safety evaluation dated January 2, 1987

- Exemption from 10 CFR 50 Appendix R, Section III.G.2.b, for Lack of 20 Feet of Redundant Cable Separation in the Intake Structure
  - Intake Structure – Fire Area 0501

  This exemption was provided in a safety evaluation dated January 2, 1987

Specific details regarding these exemptions are contained in Attachment K.

Orders

No Orders 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. 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 Nos. MD4538 and MD4539) 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 FPP will not affect measures required by B.5.b.

P. RI-PB Alternatives to NFPA 805 10 CFR 50.48(c)(4)

No risk-informed or performance-based alternatives to compliance with NFPA 805 (per 10 CFR 50.48(c)(4)) were utilized by Hatch Nuclear Plant Units 1 and 2.

Q. No Significant Hazards Evaluations

4 Pages Attached

A written evaluation of the significant hazards consideration of a proposed license amendment is required by 10 CFR 50.92, "Issuance of Amendment." Southern Nuclear Operating Company 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.

To the extent that these criteria apply to compliance with the requirements in NFPA 805, it is concluded that the proposed amendment does not involve a significant hazards consideration for the following reasons:

1. **Does the transition to NFPA 805 involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   Operation of Hatch Nuclear Plant Units 1 and 2 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 of NFPA 805 have been satisfied. The Updated Final Safety Analysis Report documents the analyses of design basis accidents at Hatch Nuclear Plant Units 1 and 2. The proposed amendment does not affect accident initiators, nor does it alter design assumptions, conditions, or configurations of the facility that would increase the probability of accidents previously evaluated. Further, the changes to be made for fire hazard protection and mitigation do not adversely affect the ability of structures, systems, or components to perform their design functions for accident mitigation, nor do they affect the postulated initiators or assumed failure modes for accidents described and evaluated in the Updated Final Safety Analysis Report. Structures, systems, or components required to safely shutdown the reactor and to maintain it in a safe shutdown condition will remain capable of performing their design functions.

   The purpose of the proposed amendment is to permit Hatch Nuclear Plant Units 1 and 2 to adopt a new fire protection licensing basis which complies with the requirements of 10 CFR 50.48(a) and (c) and the guidance in 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 10 CFR 50 Appendix R required 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 of NFPA 805 have been met.

NFPA taken as a whole, provides an acceptable alternative for satisfying General Design Criterion 3 (GDC 3) of Appendix A to 10 CFR 50, meets the underlying intent of the NRC's existing fire protection regulations and guidance, and provides for defense-in-depth. The goals, performance objectives, and performance criteria specified in Chapter 1 of the standard ensure that, if there are any increases in core damage frequency 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 the proposed amendment does not increase the probability of any accident previously evaluated. Equipment required to mitigate an accident remains capable of performing the assumed function(s). The proposed amendment will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated. The applicable radiological dose criteria will continue to be met. Therefore, the consequences of any accident previously evaluated are not increased with the implementation of the proposed amendment.

2. **Does the transition to NFPA 805 create the possibility of a new or different kind of accident from any kind of accident previously evaluated?**

Response: No.

Operation of Hatch Nuclear Plant Units 1 and 2 in accordance with the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not alter the requirements or functions for systems required during accident conditions. Implementation of the new fire protection licensing basis which complies with the requirements of 10 CFR 50.48(a) and (c) and the guidance Regulatory Guide 1.205 will not result in new or different accidents.

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

The purpose of the proposed amendment is to permit Hatch Nuclear Plant Units 1 and 2 to adopt a new fire protection licensing basis which complies with the requirements of 10 CFR 50.48(a) and (c) and the guidance in Regulatory Guide 1.205. The NRC considers that NFPA 805 provides an acceptable methodology and appropriate performance criteria for licensees to identify fire protection systems and features that are an acceptable alternative to the 10 CFR 50,

Appendix R required fire protection features (69 Fed. Reg. 33536, June 16, 2004).

The requirements of NFPA 805 address only fire protection and the impacts of fire on the plant that have previously been evaluated, with the exception of including requirements for radiological release performance criteria and non-Power Operation fire safety criteria, and alignment with plant down powers below hot shutdown. Based on this, implementation of the proposed amendment would not create the possibility of a new or different kind of accident from any kind of accident previously evaluated. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment. 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 the proposed amendment.

3. **Does the transition to NFPA 805 involve a significant reduction in the margin of safety?**

Response: No.

Operation of Hatch Nuclear Plant Units 1 and 2 in accordance with the proposed amendment does not involve a significant reduction in the margin of safety. 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 Updated Final Safety Analysis Report. The proposed amendment does not adversely affect the ability of structures, systems, or components to perform their design function. Structures, systems, or components 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 the proposed amendment is to permit Hatch Nuclear Plant Units 1 and 2 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 Regulatory Guide 1.205. The NRC considers that NFPA 805 provides an acceptable methodology and performance criteria for licensees to identify fire protection systems and features that are an acceptable alternative to the 10 CFR 50 Appendix R required 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 of NFPA 805 do not result in a significant reduction in the margin of safety.

The proposed changes are evaluated to ensure that risk and safety margins are kept within acceptable limits. The risk informed fire protection scenarios and

resolutions ensure fire risk analyses are performed and are only successful if adequate safety margin and defense-in-depth is maintained. Therefore, the transition to NFPA 805 does not involve a significant reduction in the margin of safety.

The requirements of NFPA 805 are structured to implement the NRC's mission to protect public health and safety, promote the common defense and security, and protect the environment. NFPA 805 is also 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 evaluations noted in items 1, 2 and 3 above, Southern Nuclear Operating Company has concluded that the proposed amendment presents no significant hazards consideration per the requirements set forth in 10 CFR 50.92(c), and, accordingly a finding of "no significant hazards consideration" is justified.

R. Environmental Considerations Evaluation

1 Page Attached

Southern Nuclear Operating Company has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Southern Nuclear Operating Company has determined that the proposed amendment meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50.

The purpose of the proposed amendment is to permit Hatch Nuclear Plant Units 1 and 2 to adopt a new fire protection licensing basis that complies with the requirements of 10 CFR 50.48(a) and (c) and the guidance in Regulatory Guide 1.205. The NRC considers that NFPA 805 provides an acceptable methodology and appropriate performance criteria for licensees to identify fire protection requirements that are an acceptable alternative to the 10 CFR 50 Appendix R required fire protection features (69 Fed. Reg. 33536, June 16, 2004).

The proposed amendment does not involve:

(1) A significant hazards consideration.
As stated in Attachment Q, the proposed amendment does not involve a significant hazards consideration.

(2) 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, the proposed amendment will not change the types or amounts of any effluents that may be released offsite.

(3) 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, the proposed amendment will not change the types or amounts of occupational exposures based on the results of the analysis performed and documented in Attachment E to this document based on firefighting activities.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is required to be developed in conjunction with the proposed amendment.

S. Modifications and Implementation Items

13 Pages Attached

Attachment S is redacted in its entirety.

T. Clarification of Prior NRC Approvals

1 Page Attached

There are no elements of the pre-transition fire protection program licensing basis that require clarification of prior NRC approval.

U. Internal Events PRA Quality

6 Pages Attached

In accordance with RG 1.205 position 4.3:

"The licensee should submit the documentation described in Section 4.2 of Regulatory Guide 1.200 to address the baseline PRA and application-specific analyses. For PRA Standard "supporting requirements" important to the NFPA 805 risk assessments, the NRC position is that Capability Category II is generally acceptable. Licensees should justify use of Capability Category I for specific supporting requirements in their NFPA 805 risk assessments, if they contend that it is adequate for the application. Licensees should also evaluate whether portions of the PRA need to meet Capability Category III, as described in the PRA Standard."

1. Full Scope Peer Review: November 2009

The HNP Full-Power Internal Events (including Internal Flooding) PRA has undergone a RG 1.200, Revision 2, Peer Review against the ASME PRA standard (ASME/ANS RA-Sa–2009) by a team of knowledgeable industry (vendor and utility) personnel. The Peer review was performed in November 2009 using the NEI 05-04 process, the ASME/ANS PRA Standard and Regulatory Guide 1.200, Rev. 2. The Peer Review was a full-scope review of the Technical Elements of the internal events, at-power PRA. The purpose of the review was to provide a method for establishing the technical adequacy of the PRA for the spectrum of potential risk-informed plant licensing applications for which the PRA may be used.

In the course of this review, 40 new Facts and Observations (F&Os) were prepared, including one (1) "Best Practices", 14 "Suggestions" and 25 "Findings". Most of the findings pertained to documentation issues.

The following information provides a summary of the review.

- The ASME/ANS RA-Sa–2009 PRA Standard contains a total of 326 numbered supporting requirements (SRs) in nine technical elements and one configuration control element. There were five not applicable SRs for the HNP review. They are: AS-84, IFEV-A8, LE-05, LE-06, and MU-01.

- Among 321 remaining applicable SRs, 95% of all SRs met Capability Category II or greater.

2. Resolution of Findings from RG 1.200 PRA Peer Review

The HNP Full-Power Internal Events (FPIE) PRA Fact and Observation (F&O) Independent Assessment (IA) was performed in April 2017 using the NEI 05-04 Appendix X process. The purpose was to review close-out of Finding-level F&Os issued from prior full scope PRA peer review performed against the ASME/ANS PRA Standard.

The IA Team reviewed a total of 25 Finding level F&Os. Of the 25 total reviewed findings, only three (3) were left OPEN and one (1) was left "Partially Closed". The rest of the reviewed findings were closed.

Five (5) SRs that were determined to meet CC I by the peer review team in November 2009. The IA Team reviewed the disposition of these five (5) SRs and determined that the SRs were dispositioned satisfactorily. The IA Team determined that these SRs met CC II.

The Findings that are not closed completely are summarized in the Table U-1. The IA Team considered the dispositions of the Finding-level F&Os as PRA maintenance. Therefore, a focused scoped peer review was not required and no new methods were identified.

3. Internal Events (including Internal Flooding) Base Risk

Table W-1 provides CDF and LERF values for the Full Power Internal Events (including Internal Flooding) PRA model.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding/Observation	Disposition
1-9	Reviewed the AS Notebook. Generally the discussion of the accident sequence modeling is adequate. However the following information was not present: (1) Discussion of environmental conditions associated with sequences. (2) Interface between accident sequences and plant damage states.	OPEN	Reviewed the AS Notebook. Generally the discussion of the accident sequence modeling is adequate. However the following information was not present: (1) Discussion of environmental conditions associated with sequences. (2) Interface between accident sequences and plant damage states. Basis: SR AS-B3 requires the identification of phenomenological conditions expected from each accident sequence. Possible Resolution: Include additional detail for each accident sequence. Particularly, there was no mention of the generation of harsh environments affecting temperature, pressure, debris, water levels or humidity that could impact the success of the system.	SNC disposition for NFPA 805 Application: Hatch credits equipment qualification for equipment located inside containment for LOCA and other sequences that cause harsh conditions in containment. For all other areas, the models do not credit use of equipment in the area of events that cause adverse environmental events, such as ISLOCA events and steam line breaks outside containment. The Internal flooding analysis evaluates the susceptibility of components to spray and flooding separately. This finding is a documentation issue and does not affect the NFPA 805 application.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding/Observation	Disposition
1-15	Reviewed the IF Notebook. The documentation of the initiating events is complete but does not facilitate reviews. The documentation in the IF Notebook does not represent the approach currently used in the modeling; it must be supplemented by the "FLOODING MODEL CHANGES" document. The combination of the two documents constitutes the model documentation.	PARTIALLY CLOSED	Reviewed the IF Notebook. The documentation of the initiating events is complete but does not facilitate reviews. The documentation in the IF Notebook does not represent the approach currently used in the modeling; it must be supplemented by the "FLOODING MODEL CHANGES" document. The combination of the two documents constitutes the model documentation. Basis: The documentation for IF covers two revisions of the IF model and the documentation is conflicting in some areas. Possible Resolution: Combine the current IF Notebook document with the information contained in the "FLOODING MODEL CHANGES" document into a new IF document representing the current modeling approach.	SNC disposition for NFPA 805 Application: The IA Team concluded that while HNP has performed technical work to disposition the Finding, the team found it hard to follow the documentation. Consequently, the team partially closed the Finding. Thus, this is a documentation issue with no impact on this application.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding/Observation	Disposition
4-5	In the development process of the flooding scenarios there is no credit taken for the manual isolation of floods. This approach was assumed to be conservative however, the propagation of flood water would be expanded if no operator action was taken therefore, affecting more areas and SSC's than initially projected.	OPEN	In the development process of the flooding scenarios there is no credit taken for the manual isolation of floods. This approach was assumed to be conservative however, the propagation of flood water would be expanded if no operator action was taken therefore, affecting more areas and SSC's than initially projected. Basis: The particular scenario investigated was %FL-42 (1203K-FWMP-1) which is a Plant Service Water leak affecting room 1203C. Using the documented data this room would fill up and overflow into other rooms containing PSA equipment required for accident mitigation. Possible Resolution: Operator actions should be developed and added to the scenario development and the PRA model to reflect how the plant would be operated in the event of this scenario. It may be beneficial to consider use of mitigation event trees to assure that all mitigation issues are considered.	SNC disposition for NFPA 805 Application: The IA Team noted the following during its review. *Hatch provided significant amount of information and justification for the model changes during the F&O closure process, which gave reasonable confidence that the current Hatch internal flooding model is developed appropriately and the changes can be classified as maintenance. However, the model changes have been confirmed to be significant and the current documentation supporting the internal flooding model is assessed to be not adequate to support this determination.* The IA Team concluded that while HNP has performed technical work to disposition the Finding, the team found it hard to follow the documentation. Consequently, the team did not close the Finding. Thus, this Finding is a documentation issue with no impact on this application.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding/Observation	Disposition
6-8	Consistency check did not include comparison of HEPs in regards to scenario context, plant history, procedures, operational practices, and experience.	OPEN	Consistency check did not include comparison of HEPs in regards to scenario context, plant history, procedures, operational practices, and experience. Basis: Section 8.3 of the HRA notebook evaluated consistency of HEPs based on the ratio of execution time and recovery time to check the reasonableness of the derived probability. This checks the reasonableness of the derivation of the probability. However, there was no check of consistency based on scenario context, plant history, procedures. Possible Resolution: Compare HEPs and determine if the HEP values, relative to each other, are representative of applicable scenario context, plant history, procedures, operational practices, and experience.	SNC disposition for NFPA 805 Application: A consistency check was performed as part of the review of the cutsets, review of the HRA combinations, and the review of the HRA with the Operations department. This needs to be formally documented in the Internal Events HRA notebook. The Finding has no impact on NFPA 805 application as a consistency check has been performed and documented in the HRA Notebook for Fire PRA. Thus, this Finding is a documentation issue with no impact on NFPA 805 application.

References

1. H-RIE-IEIF-U00-012, "Peer Review and F&O Closeout", Version 2.0
2. H-RIE-IEIF-U01, "U1 Model", Version 1.0
3. H-RIE-IEIF-U02, "U2 Model", Version 2.0

V. Fire PRA Quality

2 Pages Attached

In accordance with RG 1.205 position 4.3:

"The licensee should submit the documentation described in Section 4.2 of Regulatory Guide 1.200 to address the baseline PRA and application-specific analyses. For PRA Standard "supporting requirements" important to the NFPA 805 risk assessments, the NRC position is that Capability Category II is generally acceptable. Licensees should justify use of Capability Category I for specific supporting requirements in their NFPA 805 risk assessments, if they contend that it is adequate for the application. Licensees should also evaluate whether portions of the PRA need to meet Capability Category III, as described in the PRA Standard."

1) The Hatch 1/2 Fire PRA (FPRA) has undergone a RG 1.200, Revision 2, Peer Review against the ASME (ASME/ANS RA-Sa-2009) PRA Supporting Requirements (SRs) by a team of knowledgeable industry (vendor and utility) personnel. The peer review was performed in May 2016 using the NEI 07-12 process, the ASME/ANS PRA Standard and Regulatory Guide 1.200, Rev. 2. The peer review was a full-scope review of the Technical Elements of the internal fires, at-power PRA. The purpose of the review was to provide a method for establishing the technical adequacy of the PRA for the spectrum of potential risk-informed plant licensing applications for which the PRA may be used. The review was issued by the GE Hitachi BWR Owners Group in June of 2016. The conclusion of the review was that the HNP FPRA methodologies being used were appropriate and sufficient to satisfy the ASME/ANS PRA Standard RA-Sa-2009. The conclusion of the Hatch 1/2 Full scope peer review identified 61 findings, 0 unreviewed analysis methods (UAMs) and 1 best practice. The summary of the peer review findings exhibited the following statistics for the evaluation of elements to the combined Fire PRA Standard. For the Hatch 1/2 FPRA, 93.7% of the applicable SRs were assessed at Capability Category II or higher, including 83.5% of applicable SRs being assessed at Capability Category III. The HNP 1/2 FPRA had 1.2% of the applicable SRs assessed at the Capability Category level 1 and concluding that 5.1% of the applicable SRs be assessed as not MET.

2) An independent Assessment (IA) team was assembled and applied the NEI 07-12 Appendix X F&O closeout process that was endorsed by the NRC (ML17079A427) starting from the weeks of October 2, 2017 through November 10, 2017, including an onsite meeting review at the Southern Nuclear Operating Company (SNC) offices in Birmingham, Alabama in mid-October 2017. The IA team reviewed the dispositions for all F&Os and considered them as closed. For the 13 SRs that were determined to be not met by the full scope peer review in April 2016 and the three SRs that were assessed as meeting Capability Category I by the full scope peer review team in April 2016, the IA panel reviewed and reassessed these SRs as meeting Capability Category II or greater. Therefore, the Fire PRA meets Capability Category II, or greater, for all applicable supporting requirements of the ASME/ANS RA-Sa-2009 standard.

3) The IA team determined that the resolution of one Finding used a new method that was not peer reviewed in May 2016. Thus, the resolution of this Finding was considered an upgrade, requiring an additional focused scope peer review per the guidance in Appendix X to NEI 05-04/07-12/12-13. The focused scope peer review determined that the method was technically sound and provided a reasonable and realistic method for closing the F&O. The focused scope peer review did not identify any new F&Os.

References

1. H-RIE-FIREPRA-U00-014, "Hatch Fire PRA Task 14 Fire Risk Quantification", Version 2.0
2. H-RIE-FIREPRA-U00-016, "Document Resolution of Plant Hatch Fire PRA Model Peer Review Facts and Observations and Outcome of Independent Assessment Performed per NEI 07-12 Appendix X", Version 1.0
3. H-RIE-FIREPRA-U00-001, "Hatch Fire PRA Plant Boundary Definition and Partitioning Analysis (Task 01)", Version 2.0
4. H-RIE-FIREPRA-U00-002, "Hatch Fire PRA Task 2, Component Selection", Version 2.0
5. H-RIE-FIREPRA-U00-003, "Hatch Fire PRA Task 3 & 9, Cable Selection and Detailed Circuit Failure Analysis", Version 2.0
6. H-RIE-FIREPRA-U00-005, "Hatch Fire PRA Task 5, Fire Induced Risk Model", Version 2.0
7. H-RIE-FIREPRA-U00-006, "Hatch Fire PRA Task 6, Fire Ignition Frequency Development", Version 2.0
8. H-RIE-FIREPRA-U00-008A, "Hatch Fire PRA Task 8 & 11A, Detailed Fire Modeling", Version 2.0
9. H-RIE-FIREPRA-U00-010, "Hatch Fire PRA Task 10, Circuit Failure Likelihood Analysis", Version 2.0
10. H-RIE-FIREPRA-U00-008B, "Hatch Fire PRA Task 11B, Main Control Room Fire Modeling", Version 2.0
11. H-RIE-FIREPRA-U00-008C, "Hatch Fire PRA Task 11C, Multiple Compartment Fire Modeling", Version 2.0
12. H-RIE-FIREPRA-U00-012, "Hatch Fire PRA Task 12, FPRA Human Reliability Analysis", Version 2.0
13. H-RIE-FIREPRA-U00-013, "Hatch Fire PRA Task 13, Seismic Fire Interaction", Version 2.0

W. Fire PRA Insights

29 Pages Attached

Attachment W is redacted in its entirety.