Constellation Energy Nuclear Group
Nine Mile Point Nuclear Station, Unit 1

Transition to 10 CFR 50.48(c) - NFPA 805

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

CENG[SM]

a joint venture of

Constellation Energy

eDF

NINE MILE POINT
NUCLEAR STATION

Transition Report

Redacted Version

June, 2012

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Executive Summary

Nine Mile Point Nuclear Station, LLC (NMPNS) will transition the Nine Mile Point Nuclear Station, Unit 1 (NMP1) 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 The licensing basis per 10 CFR 50.48(b) and 10 CFR 50, Appendix R will be superseded.

Constellation Generation Group submitted a letter of intent to the Nuclear Regulatory Commission (NRC) on April 17, 2006, for NMP1 to adopt NFPA 805 in accordance with 10 CFR 50.48(c). By letter dated May 31, 2006, the NRC granted a three year enforcement discretion period.

By letter dated January 16, 2009, NMPNS requested that the period of enforcement discretion be extended until six months after the approval of the second pilot transition request. By letter dated March 9, 2009, the NRC approved the enforcement discretion extension request. In accordance with SECY-11-0061, Constellation Energy Nuclear Group (CENG) letter dated June 20, 2011, requested that the period of enforcement discretion be extended to June 29, 2012. By letter dated July 28, 2011, the NRC approved the enforcement discretion extension request.

The transition process consisted of a review and update of NMP1 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 (RG) 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
- 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 details to support the transition process and results.

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 License Amendment Request.

Acronym List

AC	Alternating Current
ADS	Automatic Depressurization System
AHJ	Authority Having Jurisdiction
AOV	Air Operated Valve
BTP	Branch Technical Position
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CCDP	Conditional Core Damage Probabilities
CLERP	Conditional Large Early Release Probabilities
CU	Cleanup
CDF	Core Damage Frequency
CENG	Constellation Energy Nuclear Group
CFR	Code of Federal Regulation
CGG	Constellation Generation Group
CRD	Control Rod Drive
CS	Core Spray
CSD	Cold Shutdown
CST	Condensate Storage Tank
CT	Current Transformer
CTS	Containment Spray
CTSRW	Containment Spray Raw Water
DHR	Decay Heat Removal
DID	Defense-in-Depth
DW	Drywell
EC	Emergency Condenser
ECCS	Emergency Core Cooling System
ECIV	Emergency Condenser Isolation Valve
EDG	Emergency Diesel Generator
EEE	Engineering Equivalency Evaluation
EEEE	Existing Engineering Equivalency Evaluation
EIR	Engineering Information Record
EPRI	Electric Power Research Institute
ERFBS	Electrical Raceway Fire Barrier System
ERV	Electromagnetic Relief Valve
ESW	Emergency Service Water
F&O	Facts and Observation
FA	Fire Area
FAA	Fire Area Analysis
FAQ	Frequently Asked Question
FM	Factory Mutual
FPEE	Fire Protection Engineering Evaluation
FPP	Fire Protection Program
FPSER	Fire Protection Safety Evaluation Report
FR	Federal Register
GDC	General Design Criteria

GET	General Employee Training
GL	Generic Letter
gpm	Gallons Per Minute
HEP	Human Error Probability
HRA	Human Reliability Analysis
HRE	Higher Risk Evolution
HSD	Hot Shutdown
HSS	High Safety Significant
HX	Heat Exchanger
IF	Initiating Frequency
KSF	Key Safety Function
LAR	License Amendment Request
LERF	Large Early Release Frequency
LFS	Limiting Fire Scenario
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
MCR	Main Control Room
MEFS	Maximum Expected Fire Scenario
MOV	Motor Operated Valve
MSO	Multiple Spurious Operation
MSIV	Main Steam Isolation Valve
NDD	Nuclear Division Directive
NEI	Nuclear Energy Institute
NEIL	Nuclear Electric Insurance Limited
NFPA	National Fire Protection Association
NMP1	Nine Mile Point Nuclear Station, Unit 1
NMPC	Niagara Mohawk Power Corporation
NMPNS	Nine Mile Point Nuclear Station, LLC
NPO	Non-Power Operation
NRC	Nuclear Regulatory Commission
NSCA	Nuclear Safety Capability Assessment
OMA	Operator Manual Action
P&ID	Piping & Instrumentation Drawing
PMG	Performance Monitoring Group
PRA	Probabilistic Risk Assessment
POSs	Plant Operating States
PWR	Pressurized Water Reactor
QU	Quantification
RA	Recovery Action
RAW	Risk Achievement Worth
RBCLC	Reactor Building Closed Loop Cooling
RCS	Reactor Coolant System
RG	Regulatory Guide
RI-PB	Risk-Informed Performance-Based
RIS	Regulatory Issue Summary
RP	Radiation Protection

RPV	Reactor Pressure Vessel
RSP	Remote Shutdown Panel
RSSB	Radwaste Solidification and Storage Building
RPS	Reactor Protection System
SDC	Shutdown Cooling
SE	Safety Evaluation
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SR	Supporting Requirement
SRV	Safety Relief Valve
SSA	Safe Shutdown Analysis
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SSR	Safety Shutdown Review
TS	Technical Specification
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
UL	Underwriter's Laboratories
UPS	Uninterruptible Power Supply
V&V	Verification & Validation
VFDR	Variance From Deterministic Requirements
yr	Year

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). Nine Mile Point Nuclear Station, LLC (NMPNS) is implementing the Nuclear Energy Institute (NEI) methodology contained in NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-based Fire Protection Program Under 10 CFR 50.48(c)," to transition Nine Mile Point Nuclear Station, Unit 1 (NMP1) 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 NMP1 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), for 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.

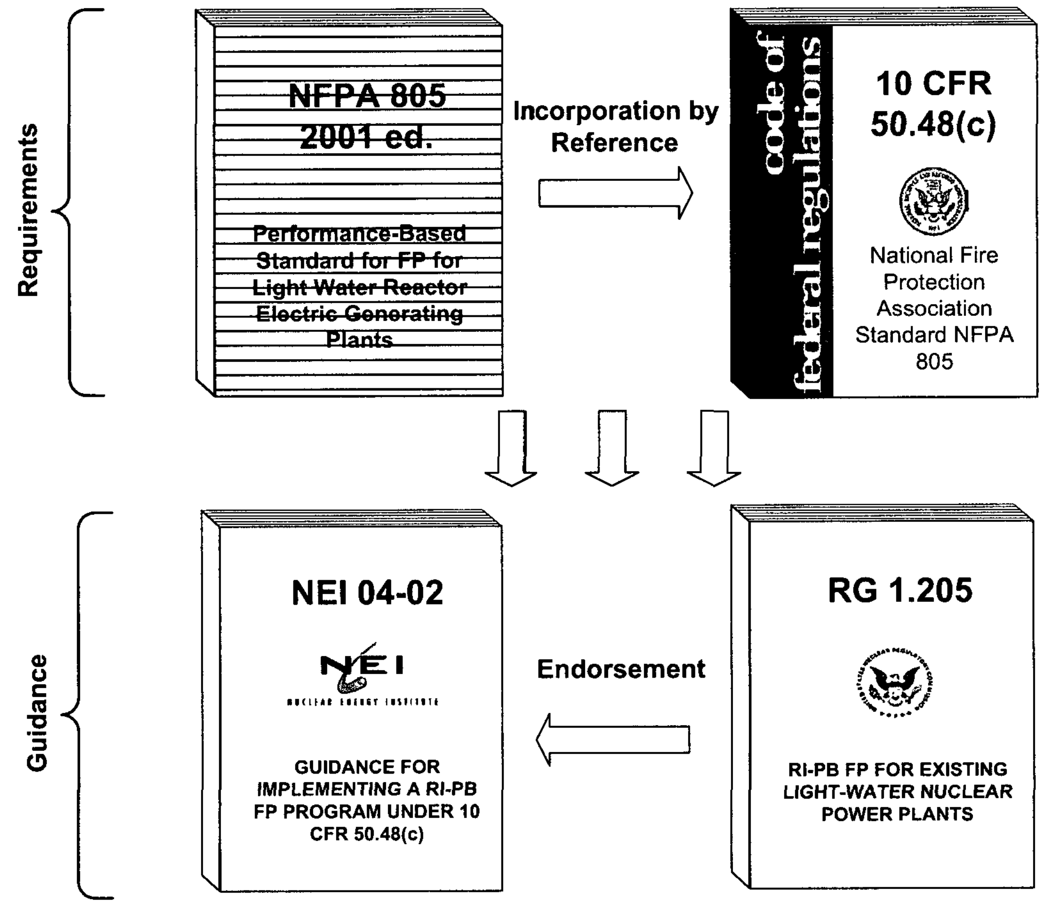

Figure 1-1 NFPA 805 Transition – Implementation Requirements/Guidance

1.1.2 Transition to 10 CFR 50.48(c)

1.1.2.1 Start of Transition

Constellation Generation Group submitted a letter of intent to the NRC on April 17, 2006, for NMP1 to adopt NFPA 805 in accordance with 10 CFR 50.48(c).

By letter dated May 31, 2006, the NRC granted a three year enforcement discretion period.

By letter dated January 16, 2009, NMPNS requested that the period of enforcement discretion be extended until six months after the approval of the second pilot transition request. By letter dated March 9, 2009, the NRC approved the enforcement discretion extension request.

In accordance with SECY-11-061, Constellation Energy Nuclear Group (CENG) letter dated June 20, 2011, requested that the period of enforcement discretion be extended to June 29, 2012. By letter dated July 28, 2011, the NRC approved the enforcement discretion extension request.

In accordance with NRC Enforcement Policy, the enforcement discretion period will continue until the NRC approval of the license amendment request (LAR) is completed.

1.1.2.2 Transition Process

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

- A new fire safe shutdown analysis
- 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; and
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

NMP1 was initially licensed to operate on August 22, 1969 (Provisional Operating License No. DPR-17 issued), with the Full-Term Operating License (DPR-63) issued on December 26, 1974. As a result, the station fire protection program is based on compliance with 10 CFR 50.48(a), 10 CFR 50.48(b), and the following license condition:

NMP1 Renewed Facility Operating License No. DPR-63, License Condition 2.D(7), states:

> *"2.D(7) Fire Protection*
>
> *Nine Mile Point Nuclear Station, LLC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report (Updated) for the facility and as approved in the Fire Protection Safety Evaluation Report dated July 26, 1979, and in the fire protection Exemption issued March 21, 1983, subject to the following provision:*
>
> *Nine Mile Point Nuclear Station, LLC may makes changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire."*

2.2 NRC Acceptance of the Fire Protection Licensing Basis

In response to the NRC's request, Niagara Mohawk Power Corporation (NMPC, the licensee at that time) performed a fire hazards analysis which analyzed the NMP1 fire protection program against the guidance of Appendix A to Branch Technical Position (BTP) APCSB 9.5-1. The results of the analysis, and the proposed modifications and additions to the fire protection system, were communicated to the NRC by letter dated February 28, 1977, and supplemented by letters dated April 10, 1978, October 6, 1978, November 17, 1978, January 2, 1979, and January 31, 1979. Those submittals served as the basis for License Amendment No. 33 and the fire protection safety evaluation report (FPSER), both issued by NRC letter dated July 26, 1979.

The July 26, 1979 FPSER identified several open items regarding plant modifications for which additional information was required by the NRC to assure that the design was acceptable prior to implementation. The five items concerned modifications relating to the following: (1) fire detection systems; (2) diesel generator building; (3) protection of structural steel; (4) sprinkler systems; and (5) diesel generator rooms. By letters dated July 22, 1980 and July 30, 1980, the NRC found acceptable the modifications proposed by NMPC to address the first three items, and stated their positions regarding the fourth and fifth items. NMPC subsequently addressed the fourth item (installation of sprinkler systems in the cable spreading room and the diesel generator rooms) and the fifth item (protection of diesel generator control cables) in a letter to the NRC dated August 11, 1980. Thereafter, NMPC letter dated November 3, 1989 identified several cases where the installed modification differed from the description contained in the FPSER. The NRC agreed with the NMPC evaluation that the differences did not result in a decrease in the effectiveness of the NMP1 fire protection program and that the requirements of

Appendix R to 10 CFR 50 were satisfied, as documented in the NRC letter dated December 15, 1989.

In response to NRC Generic Letter (GL) 81-12, dated February 20, 1981, NMPC submitted information addressing the requirements of 10 CFR 50, Appendix R, Section III.G.3, with respect to safe shutdown capability in the event of a fire. NMPC letters dated March 19, 1981, June 9, 1981, September 30, 1981, and September 2, 1982 submitted descriptions of modifications to the remote reactor shutdown system to address the requirements of Appendix R, Section III.G.3. In addition, on October 1, 1982, NMPC submitted the report entitled "Appendix R Review Safe Shutdown Analysis," addressing the provisions of Appendix R Sections III.G and III.L. Supplemental information addressing safe shutdown capability for postulated control room fire events was provided in the NMPC letter dated December 3, 1982. The NRC safety evaluation (SE) dated March 3, 1983 concluded that the modifications proposed by NMPC and the NMP1 alternate safe shutdown capability met the requirements of Appendix R Sections III.G.3 and III.L for fire areas within the control complex. Operability requirements for the remote shutdown panels were subsequently added to the NMP1 Technical Specifications (TS) by License Amendment No. 71, issued by NRC letter dated April 1, 1985.

As stated in NMP1 License Condition 2.D(7), by letter dated March 21, 1983, the NRC granted the following exemptions from the requirements of Section III.G of Appendix R:

- An exemption from the requirements of Section III.G.2 of Appendix R for the battery board rooms (FA 16A and FA 16B), since their boundary walls do not provide the required 3-hour rated barriers.
- An exemption from the requirements of Section III.G.2 of Appendix R for the battery rooms (FA 17A and FA 17B), since their boundary walls do not provide the required 3-hour rated barriers.
- An exemption from the requirements of Section III.G of Appendix R for the control room (FA 11), since the control room ceiling does not have a 3-hour rating from the control room side due to unprotected structural steel members.
- An exemption from the requirements of Section III.G.2 of Appendix R for the wall between the reactor building and the turbine building above elevation 340' (FA 1, FA 2, and FA 5), since the wall is not a 3-hour rated barrier.
- An exemption from the requirements of Section III.G.2 of Appendix R for the fire break zone separating FA 1 and FA 2 in the reactor building upper level (elevation 340'), since the wall is not a 3-hour rated barrier.

By letter dated May 11, 1984, NMPC requested NRC review and acceptance of the fire detection and suppression provided for the NMP1 reactor building, Fire Sub-Area 1 and Fire Sub-Area 2, at elevations 237', 261', 281', 298', and 318'. NRC GL 83-33, dated October 19, 1983, stated that to satisfy the requirements of Subsections III.G.2.b, III.G.2.c, and III.G.2.e of Appendix R to 10 CFR 50, fire detection and automatic suppression systems needed to be installed throughout the fire area. Evaluations performed by NMPC determined that the partial detection and suppression provided in the identified reactor building fire areas adequately protected against the area fire hazards. The NRC SE dated August 6, 1986 concluded that the lack of full coverage

detection and suppression systems in the subject reactor building fire areas was acceptable.

The NRC safety evaluations and approved exemptions described above provide the basis for the NRC approval of the NMP1 Fire Protection Program (FPP).

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, NMP1 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.

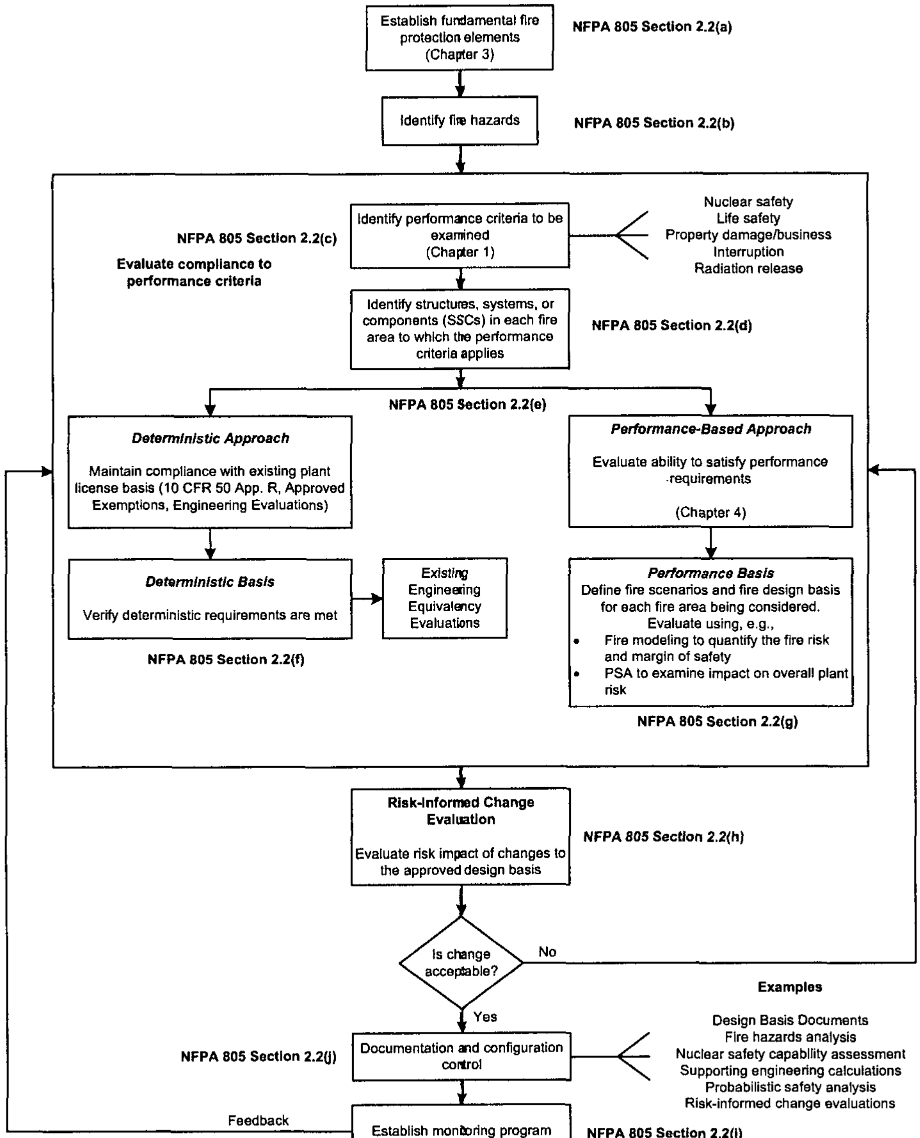

Figure 3-1 NFPA 805 Process [NEI 04-02 Figure 3-1 based on Figure 2-2 of NFPA 805][2]

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.

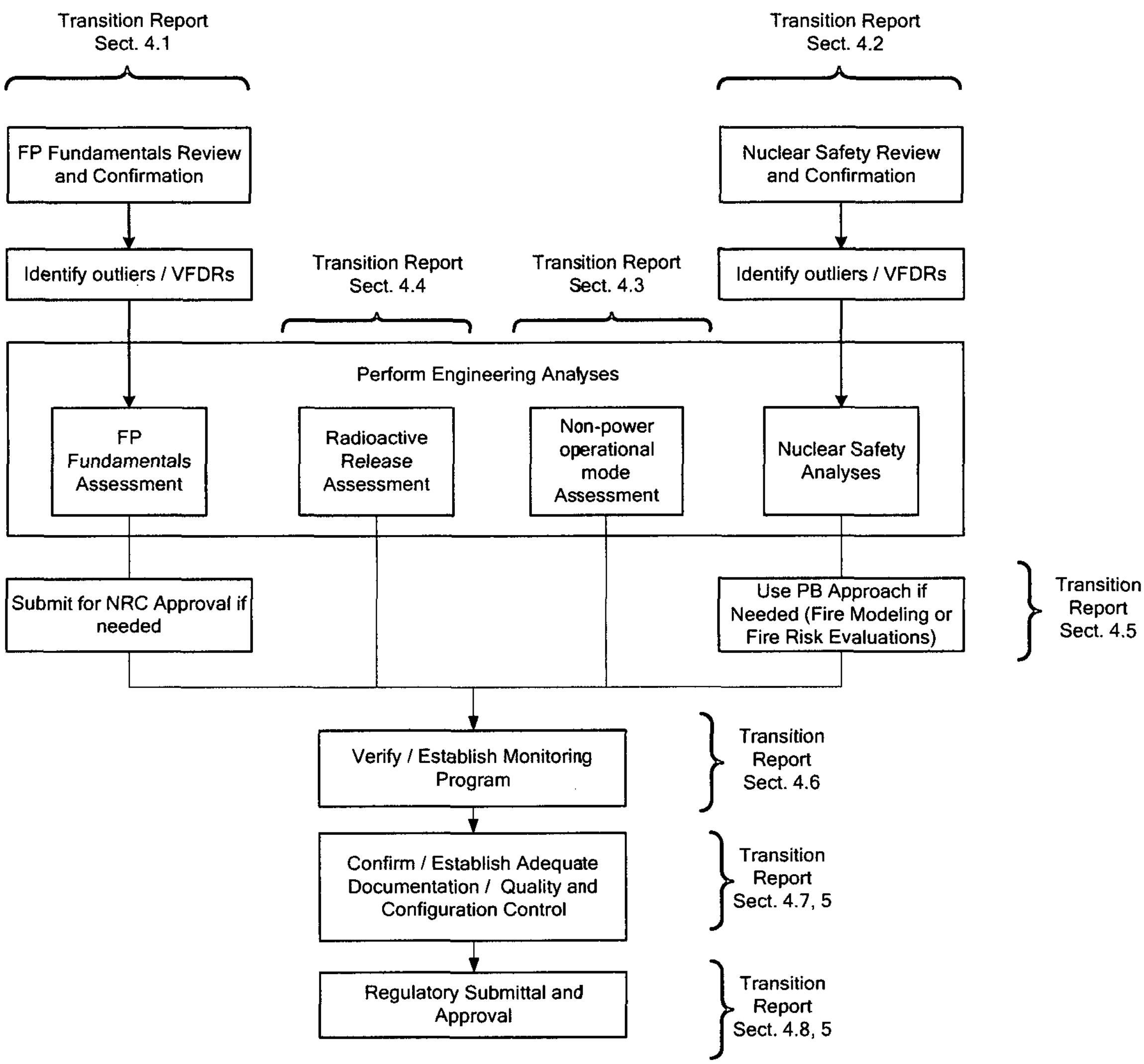

Figure 3-2 Transition Process (Simplified) [based on NEI 04-02 Figure 4-1]

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 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.

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 NMP1 Fire Protection Program to the requirements of NFPA 805 Chapter 3 was performed and documented in Engineering Information Record (EIR) 51-9077683, NFPA 805 Fundamental Fire Protection Program and Design Elements Transition Review. This document 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 item for implementation – For those sections/subsections determined to meet the requirements of NFPA 805 with an item for implementation or modification.
- 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.
- 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.

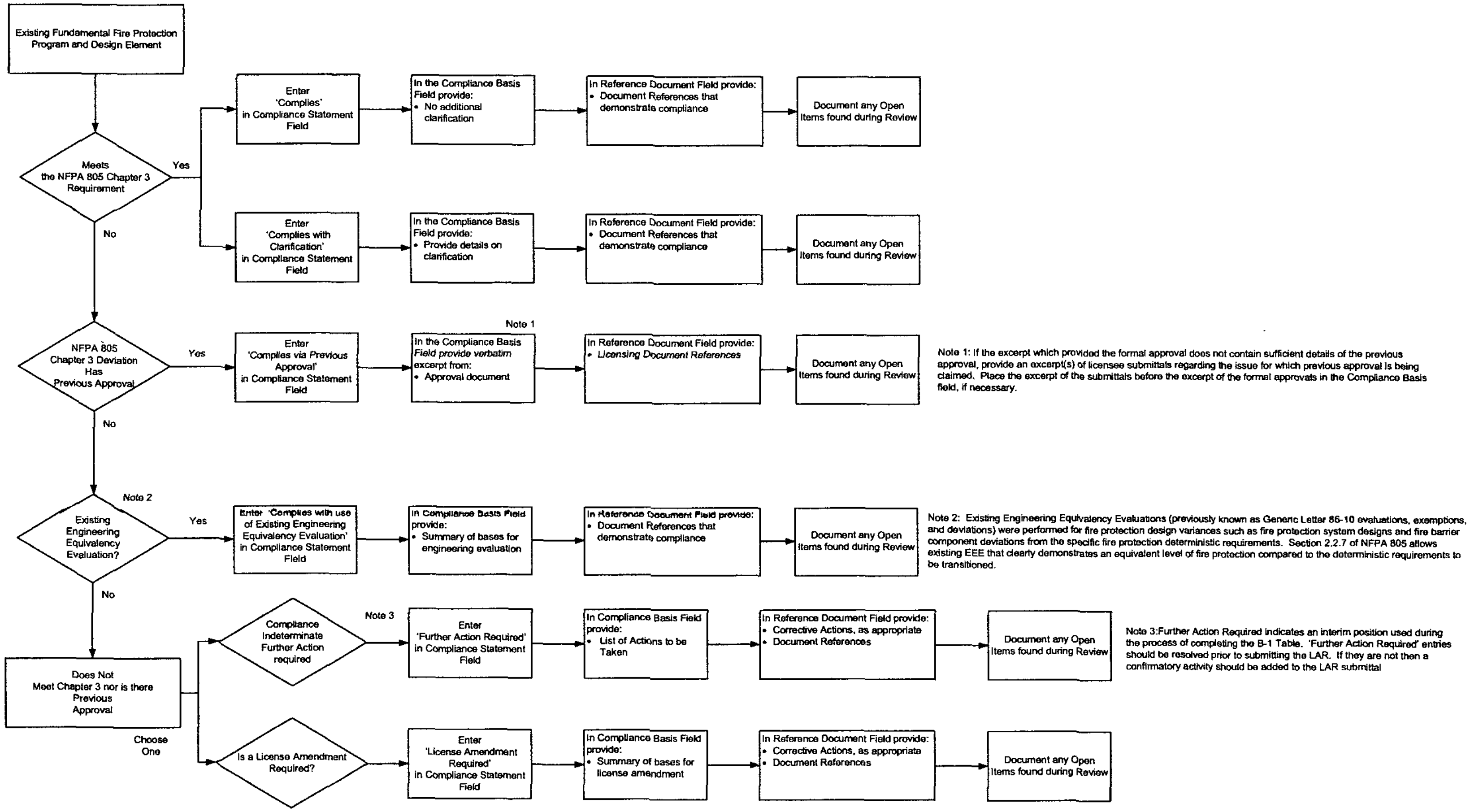

Figure 4-1 - Fundamental Fire Protection Program and Design Elements Transition Process [Based on NEI 04-02 Figure 4-2][3]

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[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 NMP1 either:

- Complies directly with the requirements of NFPA 805 Chapter 3,
- Complies with the requirements of NFPA 805 Chapter 3 with an item for implementation or modification,
- 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.

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

NPFA 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. NMPNS 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 Previously Approved by NRC

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

- 3.3.5.1 Approval is requested for Overhead wiring

The specific deviation and a discussion of how the alternative satisfies 10 CFR 50.48(c)(2)(vii) requirements are provided in Attachment L. NMPNS 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.

The process used to determine which NMP1 structures constitute the power block is consistent with that described in Section 1.6.46 of NFPA 805. Specifically, only

structures housing equipment required for nuclear plant operations are considered as "power block" structures.

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"*

The NSCA methodology review evaluated the existing post-fire safe shutdown analysis (SSA) methodology against the guidance provided in NEI 00-01, Revision 2, Chapter 3, "Deterministic Methodology," as discussed in Appendix B-2 of NEI 04-02. NMP1 used the guidance provided in NEI 00-01, Revision 2 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 subsection of NFPA 805 Section 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.

In addition, a review of NEI 00-01, Revision 3 was conducted against the guidance from NEI 00-01, Revision 2. There were no gaps relative to MSOs identified.
The comparison of the NMP1 existing post-fire SSA to NEI 00-01 Chapter 3 (NEI 04-02 Table B-2) was performed and documented in EIR 51-9133191, Nine Mile Point Unit 1 – Nuclear Safety Capability Assessment.

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, Revision 2, Chapter 3 directly or meets the intent of the endorsed guidance with adequate justification as documented in Attachment B. Referenced documents are planned as being retained as post-transition documents.

NEI 00-01, Revision 2, Chapter 3 contains guidance criteria concerning identifying required and important to SSD components. These specific guidance criteria are not applicable to plants transitioning to NFPA 805; therefore, they were not addressed for NMP1.

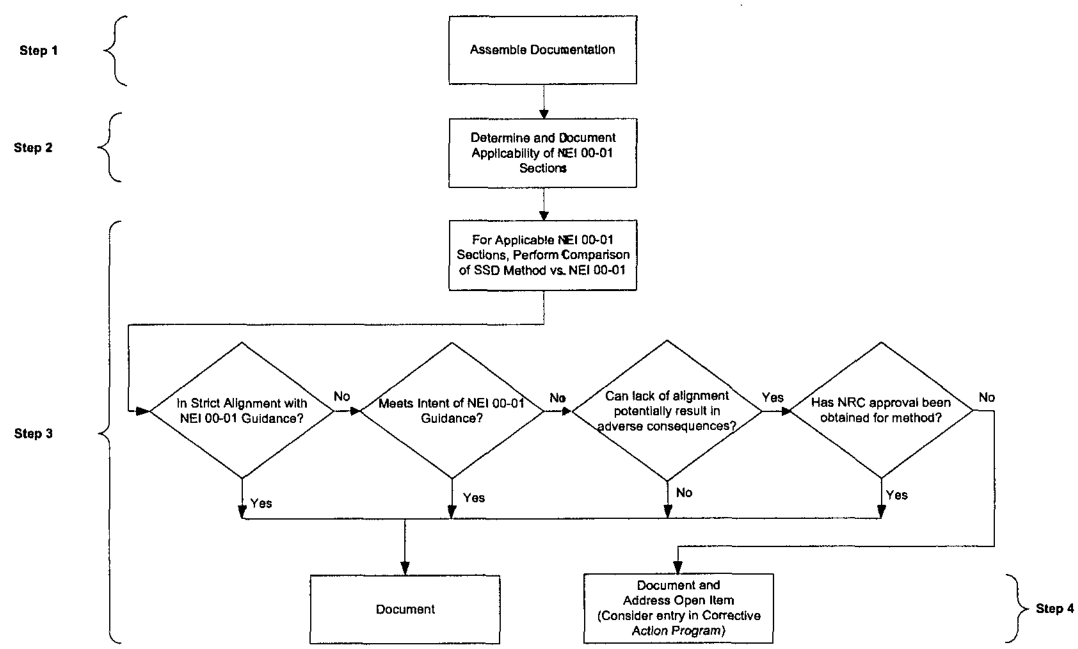

Figure 4-2 – Summary of Nuclear Safety Methodology Review Process (FAQ 07-0039)

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 K_{eff} <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 K_{eff} <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

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

- At-Power analysis, Power Operating Conditions including startup and run. This analysis is discussed in Section 4.2.4.
- Non-Power analysis, which includes Hot Shutdown and below. This analysis is discussed in Section 4.3.

Based on the EIR 51-9133191, Nine Mile Point Unit 1 – Nuclear Safety Capability Assessment, the NFPA 805 licensing basis for a safe and stable condition is defined as the ability to maintain $K_{eff} < 0.99$ with a reactor coolant temperature at or below the requirements for hot shutdown and then subsequently cool down and maintain NMP1 in a cold shutdown condition.

The primary means of achieving and maintaining Hot Shutdown (HSD) is via the Emergency Cooling system. Either of the two redundant Emergency Cooling decay heat removal loops can achieve and maintain HSD. The Emergency Cooling system operates by natural circulation where steam flows upward to the condenser(s) and returns as condensate to the Reactor Pressure Vessel (RPV). Decay heat is removed through the transfer of heat from the reactor coolant to the shell side water of the Emergency Cooling which vents the developed steam to atmosphere. Operation of either Emergency Cooling loop can sustain HSD conditions for 8 hours without the need for makeup from the Condensate Storage Tank (CST).

The Emergency Cooling system can be initiated either manually or automatically. The RPS instruments and logic that automatically initiate the Emergency Cooling system on high reactor pressure or low-low reactor level have been included in the analysis. Manual initiation of the Emergency Cooling system can be accomplished from either the Control Room or Remote Shutdown Panel (RSP) depending on the fire location. AC power is not required to manually initiate Decay Heat Removal (DHR) via the Emergency Cooling system.

In the event the primary HSD method is not available, the plant can be maintained in HSD by opening three Electromagnetic Relief Valves (ERVs) in the automatic depressurization system (ADS) and blowing steam to the Torus to reduce pressure. When reactor pressure reaches approximately 365 psig, Core Spray (CS) may be utilized to provide core cooling. AC and DC electrical power are required for this method.

The ADS system can be initiated either manually or automatically. The RPS instruments and logic automatically initiate the ADS system on a combination of low-low-low reactor level and high drywell pressure.

The CS system can be initiated either manually or automatically. The RPS instruments and logic automatically initiate the CS system on low-low reactor level or high drywell pressure.

The instrumentation and logic circuitry that automatically initiates ADS and CS have been included in the analysis. Manual initiation of the ERVs and CS system is accomplished from the Control Room. In the event spurious actuations of the ERVs

take place due to a Control Room fire, CS can be initiated manually from outside the Control Room.

The preferred method of achieving and maintaining Cold Shutdown (CSD) is via the Shutdown Cooling System (SDC). When reactor pressure is reduced to 120 psig and reactor temperature is reduced to 350°F, the plant can be transitioned to CSD by initiating SDC. The SDC system is supported by Reactor Building Closed Loop Cooling (RBCLC) and Emergency Service Water (ESW). One SDC pump in association with RBCLC and ESW is required to achieve and maintain CSD conditions. AC power is required to initiate SDC.

Under this scenario, reactor coolant makeup is required after 8 hours with an assumed Technical Specification leakage of 25 gpm. Makeup is provided via the Control Rod Drive (CRD) system using one of the CRD pumps drawing suction from the CST. AC power is required to operate a CRD pump. The diesel driven fire pump may also be aligned to provide makeup in the event no CRD pump is available.

In the event the primary CSD method is not available, the plant can be cooled down to CSD using the CS system. CS is a two loop system. Operation of one loop is adequate to achieve CSD. When utilizing CS, the reactor vessel floods to the point where the ERVs are passing fluid to the Torus rather than steam, in essence placing the RCS in recirculation through the Torus. During this process, decay heat is removed by operation of the Containment Spray (CTS) system in conjunction with the Containment Spray Raw Water (CTSRW) system. This method will bring the plant directly to CSD. Fully flooding the RPV negates the need for another system to provide inventory makeup. AC power is required to initiate this method.

Long Term Safe and Stable conditions will be maintained using either the preferred or alternate CSD method. With AC power available from either the station Emergency Diesel Generators (EDGs) or offsite power, CSD conditions can be maintained indefinitely.

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 2.21 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 EIR 51-9156521, Recovery Action Review for Nine Mile Point Nuclear Power Station Unit 1 Transition to NFPA 805. 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 NMP1 is summarized below.

As part of the NFPA 805 transition project, a review and evaluation of NMP1 susceptibility to fire-induced MSOs was performed. The process was conducted in accordance with NEI 04-02 and RG 1.205, as supplemented by FAQ 07-0038 Revision 3 (ML110140242). The BWR Generic MSO list in NEI 00-01, Revision 2, dated June 5, 2009 was utilized.

The approach outlined in Figure 4-3 (based on Revision 3 from 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 existing post-fire SSA 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).

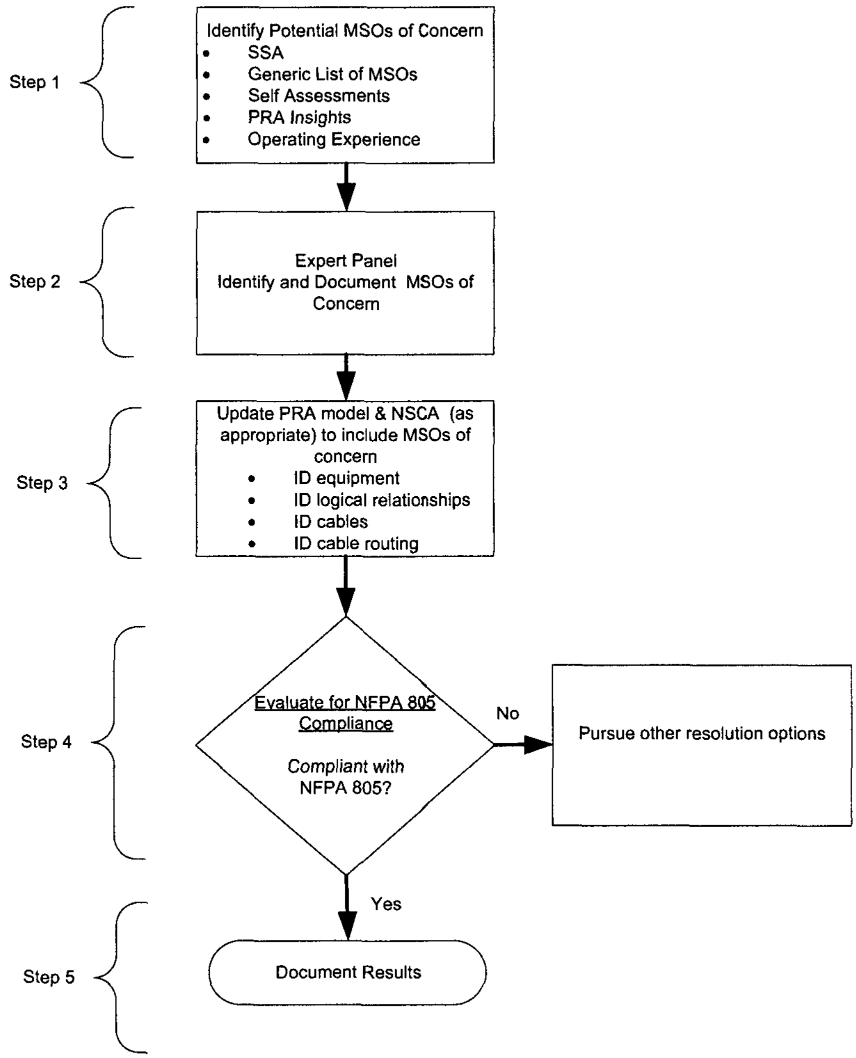

***Figure 4-3 – Multiple Spurious Operations – Transition Resolution Process* (Based on FAQ 07-0038)**

Results

Refer to Attachment F for the process used for NMP1 and the results from the process.

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 includes 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 07-0054, 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 the EEEE was referenced where required to demonstrate compliance and was included in Attachment L for NRC review and approval.

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

Results

The review results for EEEEs are documented in EIR 51-9077683, NFPA 805 Fundamental Fire Protection Program and Design Elements Transition Review.

In accordance with the guidance provided in RG 1.205, Regulatory Position 2.3.2, and NEI 04-02, as clarified by FAQ 07-0054, EEEEs used to demonstrate compliance with Chapters 3 and 4 of NFPA 805 are referenced in 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 / 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 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 requirements of Section III.G.2 of Appendix R for the battery board rooms (FA 16A and FA 16B), since their boundary walls do not provide the required 3-hour rated barriers.
- An exemption from the requirements of Section III.G.2 of Appendix R for the battery rooms (FA 17A and FA 17B), since their boundary walls do not provide the required 3-hour rated barriers.
- An exemption from the requirements of Section III.G of Appendix R for the control room (FA 11), since the control room ceiling does not have a 3-hour rating from the control room side due to unprotected structural steel members.
- An exemption from the requirements of Section III.G.2 of Appendix R for the wall between the reactor building and the turbine building above elevation 340' (FA 1, FA 2, and FA 5), since the wall is not a 3-hour rated barrier.
- An exemption from the requirements of Section III.G.2 of Appendix R for the fire break zone separating FA 1 and FA 2 in the reactor building upper level (elevation 340'), since the wall is not a 3-hour rated barrier.

These exemptions are no longer required because the subject boundaries have been demonstrated adequate for the hazard in an EEEE.

Since the exemptions are either compliant with 10 CFR 50.48(c) or no longer necessary, in accordance with the requirements of 10 CFR 50.48(c)(3)(i), NMPNS requests that the exemptions listed in Attachment K be rescinded as part of the LAR process. 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 07-0054. 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 documented 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.
- 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, 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). 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 were 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.

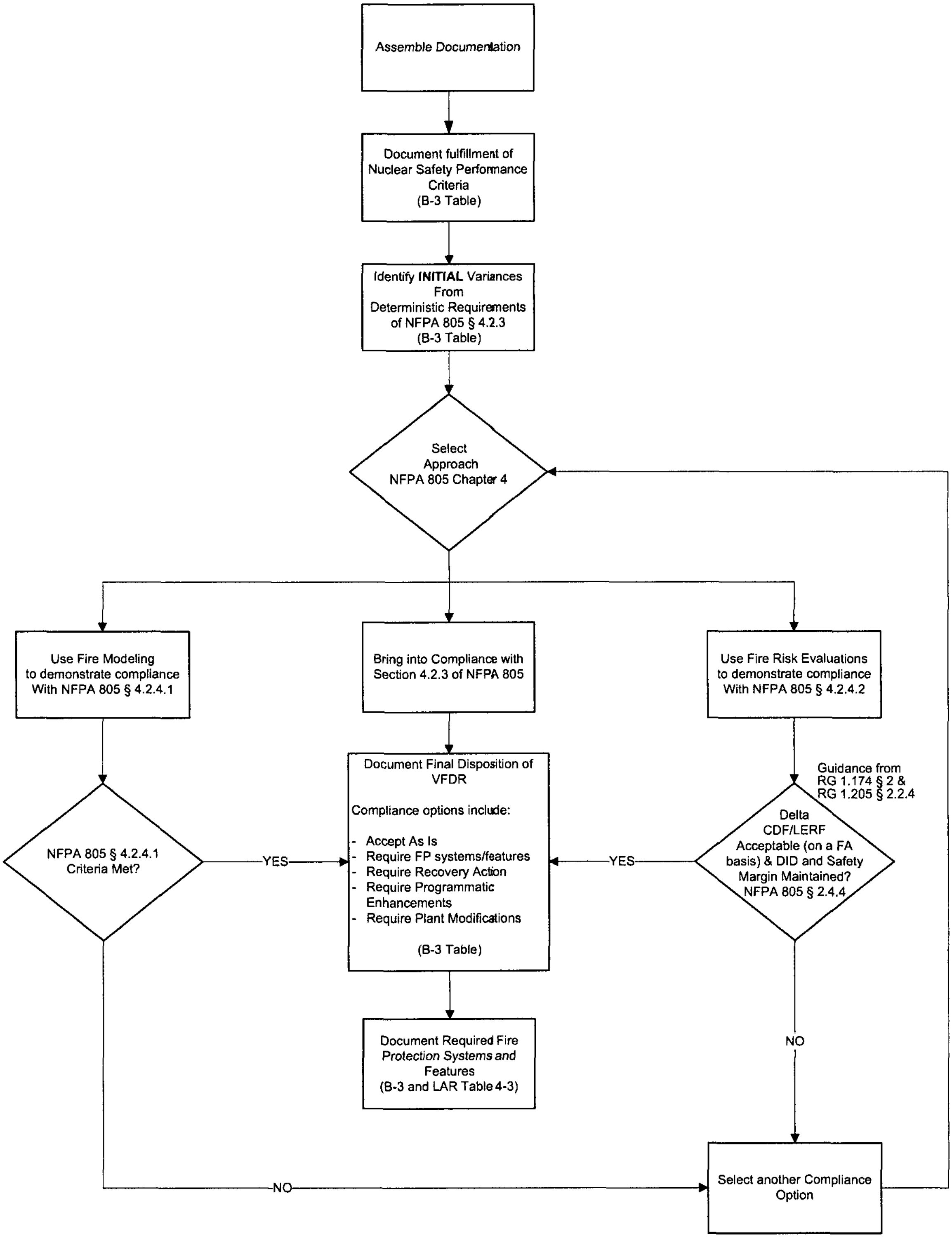

Figure 4-4 – Summary of Fire Area Review [Based on FAQ 07-0054 Revision 1]

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.
- Performance Goal Summary – An overview of the method of accomplishment of each of the performance criteria in NFPA 805 Section 1.5.
- Reference Documents – Specific references to Nuclear Safety Capability Assessment Documents.
- Licensing Actions – Specific references to 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 are provided. In addition, summaries of Fire Risk Evaluations performed for variances from the deterministic requirements are also 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 are 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

NMP1 implemented the process outlined in NEI 04-02 and FAQ 07-0040, "Non-Power Operations Clarifications." 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:

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

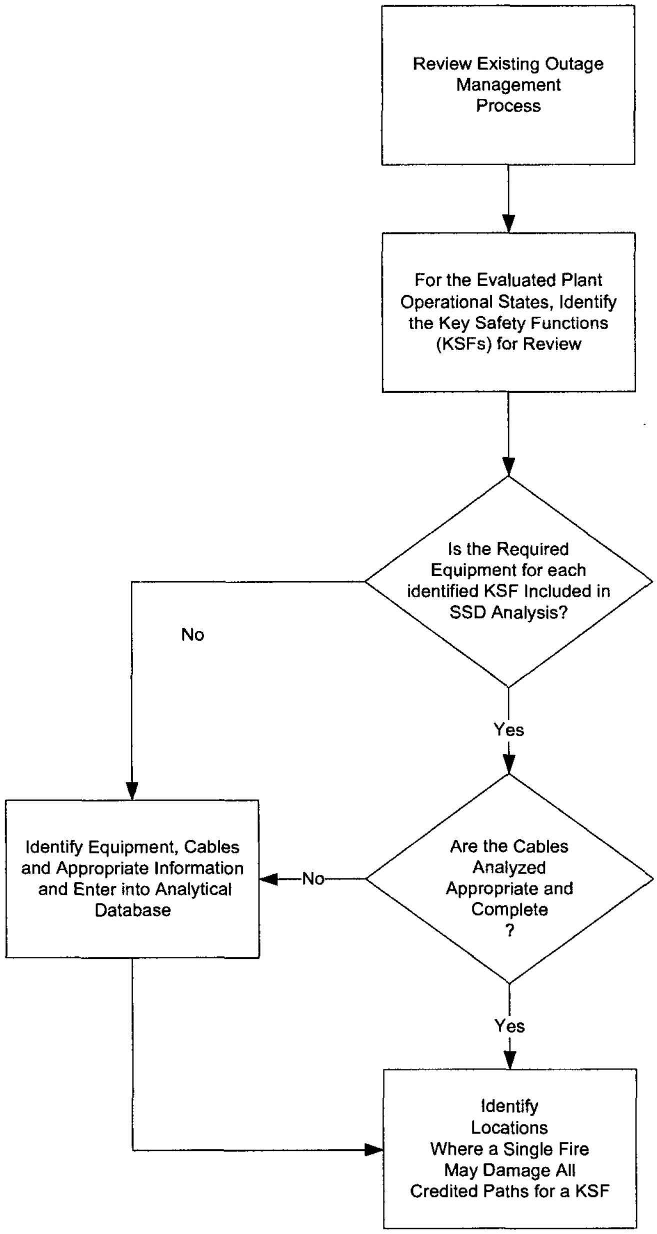

Figure 4-5 Review POSs, KSFs, Equipment, and Cables, and Identify Pinch Points

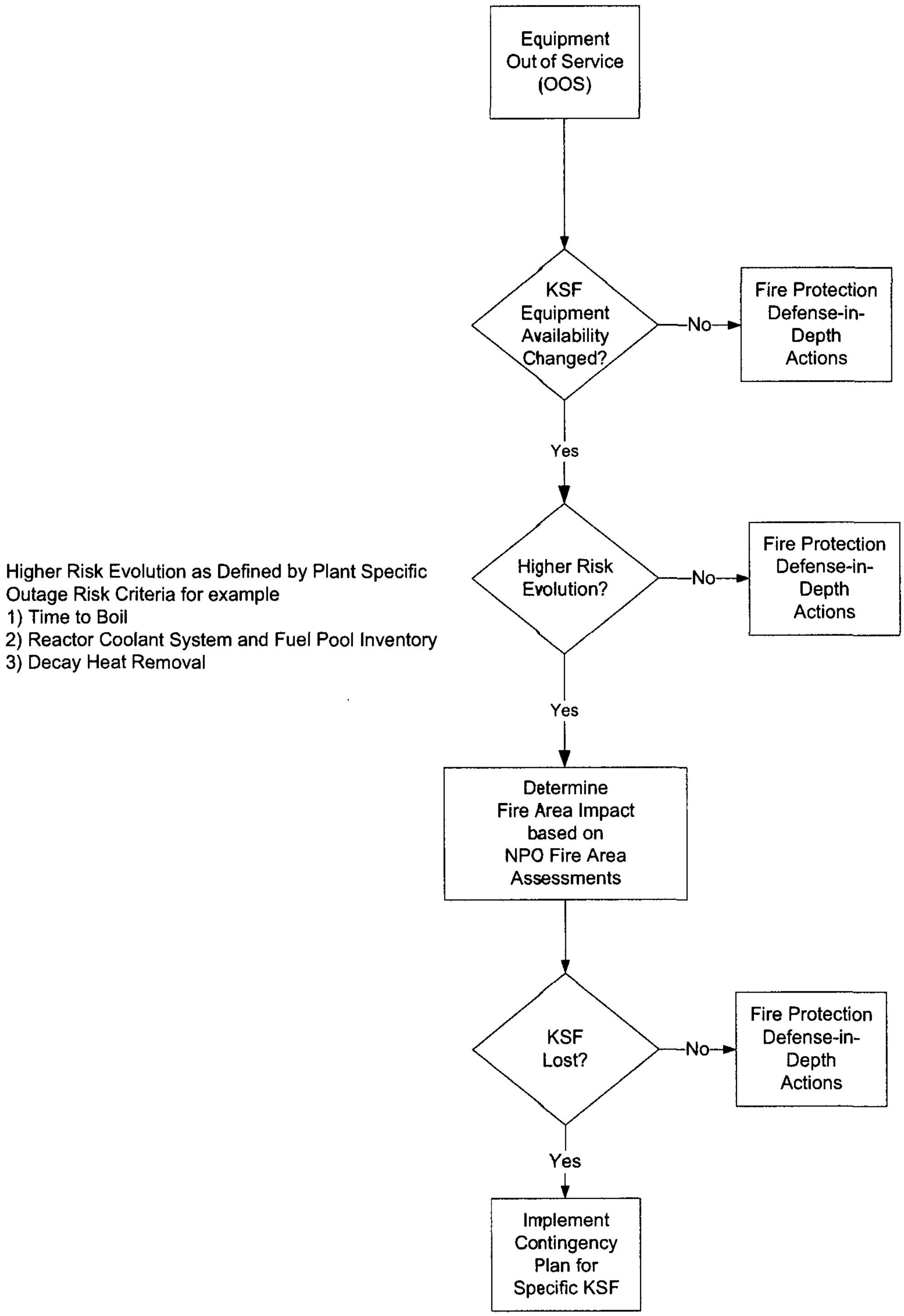

Figure 4-6 Manage Pinch Points

4.3.2 Results of the Evaluation Process

Based on FAQ 07-0040, the Plant Operating States (POSs) considered for equipment and cable selection are defined in EIR 51-9171174, Nine Mile Point 1, Non-Power Operations KSF Equipment List. The methodology for determination of KSFs, success paths, components required to achieve the success paths and their associated cabling were defined in EIR 51-9137629, Nine Mile Point 1 – Nuclear Power Station – NFPA 805 Transition – Non Power Operations Component Pinch Point Analysis. These documents provide the component selection information for HSD, CSD, Refueling and Defueled conditions.

KSFs are evaluated in each fire zone: DHR for both the Reactor Vessel and the Spent Fuel Pool (SFP), Inventory Control for both the Reactor Vessel and the SFP, and Power availability. Each may have one or more KSF paths that satisfy that specific KSF.

No effort was made to eliminate or reduce fire impact by circuit analysis; therefore, a conservative estimate of damage is provided, including spurious operation of equipment. By assuming that a single fire impacts any and all components in a fire zone, (whether the individual component or its associated cables are physically located within the fire zone), the assumption is made that the entire contents of the fire zone are lost.

If a component that is part of a particular KSF flow path is impacted, it is assumed that the KSF path is lost. However, there may be one or more other flow paths within the particular KSF that are not impacted; therefore, the KSF is not considered lost and does not constitute a pinch point. Only when all paths for a particular KSF are impacted, is the KSF itself considered lost and is a pinch point.

The results categorize each KSF (in each fire zone), as either "I", "L" or "N" as follows:

- "I" (Impacted): At least one of the KSF paths associated with the KSF is affected, i.e., a component of the KSF path or any of its associated cables within the fire zone are impacted whereby that path can no longer be assured of being functional. However, at least <u>one</u> other KSF path within that KSF is still available.
- "L" (Lost): <u>All</u> available success paths for a given KSF are impacted.
- "N" (None): <u>No impacts</u> to the KSF are identified.

"Pinch Points" were then identified (on a fire zone basis), based on the loss of a KSF. An "N" in the pinch point column indicates that no KSFs were lost in this fire zone. A "Y" in this column indicates that one or more KSFs were lost in this fire zone and therefore, constitutes a pinch point.

NUMARC 91-06 discusses the development of outage plans and schedules. A key element of that process is to ensure the KSFs perform as needed during the various outage evolutions. During outage planning, the NPO Fire Zone Assessment is reviewed to identify areas of single-point KSF vulnerability during HRE to develop needed contingency plans/actions. Depending upon the significance of the damage for those areas, combinations of the following options to reduce fire risk are considered at a minimum:

- Prohibition or limitation of hot work in fire areas during periods of increased vulnerability;
- Verification of fire 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 its desired position);
- Provision of additional fire patrols at periodic intervals or other appropriate compensatory measures (such as surveillance cameras) during increased vulnerability;
- Use of recovery actions to mitigate losses of Key Safety Functions;
- Reschedule the work to a period with lower risk;
- Identification and monitoring in-situ ignition sources for fire precursors.

In addition, KSF equipment removed from service during the HREs is evaluated. The evaluation is based on KSF equipment availability and NPO Fire Zone Assessment for any necessary contingency plans/actions. Note that recovery actions were not used to mitigate/eliminate fire-induced failures.

See Attachment D for more complete details. Based on incorporation of the recommendations from the KSF pinch point evaluations into appropriate plant procedures prior to implementation of the NFPA 805 fire protection program, the performance goals for NPO modes are fulfilled and the requirements of NFPA 805 will be met. See Implementation Items in Attachment S.

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 EIR 51-9085686, NMP-1 NFPA 805 Radiological Release Transition Review. The methodology consists of the following:

- Screened the fire zones based on the potential for the presence of contaminated materials during all plant operating modes, including full power and non-power conditions. The screening process considered input from radiation protection personnel and review of the NMP1 fire pre-plans. The evaluation focused on radioactive release to any unrestricted area due to firefighting activities.
- Reviewed 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.
- Reviewed 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 NMP1 Technical Specifications are met.

4.4.2 Results of the Evaluation Process

The method utilized to identify the systems, components, and flow paths that are used to meet the radiological release performance criteria consists of review of the plant pre-fire plans, fire brigade training materials, gaseous effluent (building ventilation system) design documents, and liquid effluent (building drainage system) design documents.

Fire protection program elements, including measures, systems, procedural control actions, and flow paths, are credited to meet the radiological release performance criteria. These elements provide reasonable assurance that the radiological release performance criteria will be met for both smoke and fire suppression agents, on a fire zone-by-fire zone basis, during all modes of operation, and minimize the potential for cross-contamination (water runoff and smoke from a contaminated area being directed through an uncontaminated area).

No instances were identified where the existing engineering controls are not adequate to ensure containment of potentially contaminated gaseous or liquid effluents. Therefore, no need for a bounding analysis, quantitative analysis, or other analysis was identified to demonstrate that the limitations for instantaneous release of gaseous or liquid radioactive effluents specified in the Technical Specifications are met. If engineering controls were to be inoperable due to a fire (e.g., sump pump, filter, ventilation flow path), administrative controls would ensure effluents can be contained within the structure boundaries.

The results of the fire zone-by-fire zone review are included in Attachment E, with a summary of how the radioactive release goals, objectives and performance criteria are met for each zone.

The radioactive release review determined the fire protection program is compliant with the requirements of NFPA 805 and the guidance in NEI 04-02 and RG 1.205.

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 NMP1 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). NMPNS conducted a peer review during the period of October 24-28, 2011, using

independent industry analysts in accordance with RG 1.200 prior to submitting this risk-informed submittal.

The resulting fire PRA model is being 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 of the Fire PRA peer review. Section 4.5.1.4 describes insights gained from the Fire PRA.

4.5.1.1 Internal Events PRA

The NMP1 base internal events PRA was the starting point for the Fire PRA. The internal events PRA model was modified to capture the effects of fire both as an initiator of an event and the subsequent potential failure modes for affected circuits or individual targets.

In 2008, the Boiling Water Reactor Owners Group (BWROG) performed a peer review of the NMP1 internal events PRA against the requirements of ASME/ANS RA-Sb-2005 and RG 1.200, Revision 1. In that assessment, of the 309 Supporting Requirements (SRs) applicable to NMP1, 90% were assessed to be met at Capability Category II or better. The review identified 26 findings, 2 best practices and 52 suggested enhancements. These resultant facts and observations (F&Os) were reviewed and entered into the NMPNS PRA Configuration Control database. Complete results of the peer review, including discussion of how any open F&Os have no significant impact on use of the internal events PRA model to support the NFPA 805 transition process, are presented in Attachment U.

Note that NMPNS has performed a gap assessment against RG 1.200, Revision 2. A business case has been drafted to evaluate gap closure activities.

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 the subsequent potential failure modes for affected circuits or individual targets. The process for creation of the Fire PRA model and quantification of that model use a methodology consistent with the guidance provided in NUREG/CR-6850/EPRI TR 1011989 and subsequent clarifications documented in responses to NFPA 805 FAQs. No unreviewed methods or deviations from NUREG/CR-6850 were utilized in the Fire PRA model development. The Fire PRA model is created in CAFTA and is quantified using the EPRI FRANX software. NMP1 is a single independent unit, so no consideration was made for any dual-unit features or shared fire zones between two units.

Fire Model Utilization in the Application

The applicability of a fire model and thus its acceptability to the AHJ is expected to vary with the particular application considered. It is not possible for a fire model to be acceptable without providing substantiation which includes V&V documentation for the model in general and documentation that the particular application is within acceptable limits for the way in which it is applied.

When performance-based fire modeling per NFPA 805 Section 4.2.4.1 is used, the V&V of those fire models is required to be directly documented in the LAR for NRC review. However, such fire modeling was not used as part of NMP1 NFPA 805 transition. That is, VFDRs were not dispositioned at a fire area level through the use of deterministic fire modeling and therefore maximum expected fire scenario (MEFS) and limiting fire scenario (LFS) were not each analyzed and compared.

Fire modeling was performed as part of the Fire PRA development (NFPA 805, Section 4.2.4.2). The determination of the acceptability of methods used in the Fire PRA applied to NFPA 805, per RG 1.205, is made through an independent peer review of the Fire PRA against RG 1.200 and the referenced ASME/ANS PRA Standard RA-Sa-2009. Therefore, for NMP1 the acceptability (V&V) of the fire modeling tools used was assessed during the peer review and determined to meet ASME/ANS RA-Sa-2009. This review found that most of the fire modeling SRs were met at Capability Category II or higher and issued findings for those SRs that were not. The findings were addressed, as documented in Attachment V, and NMPNS believes that the fire modeling meets Capability Category II for those SRs based on the information provided in Attachment V and the fire modeling PRA notebooks. This is sufficient for a Fire PRA being applied to NFPA 805. The acceptability of the use of fire modeling is included in Attachment J, which also summarizes the fire modeling approach implemented within the Fire PRA to ensure that the limitations of the available V&V studies (i.e., NUREG 1924) do not impact the development of the fire risk assessment and do not result in the screening of potentially risk contributing scenarios.

4.5.1.3 Results of Fire PRA Peer Review

The NMP1 Fire PRA was peer reviewed against the requirements of ASME/ANS RA-Sa-2009, Part 4 during the period of October 24-28, 2011. The summary results of the review include 237 supporting requirements met at Capability Category II or better, 5 supporting requirements were found to be met at Capability Category 1 and 37 supporting requirements were determined to be not met. The results of the review are summarized in the Peer Review report.

The Peer Review noted a number of F&Os, 128 in total. Of these, 65 were designated as findings that need to be addressed to meet the SRs or to achieve Capability Category II. In addition, 59 F&Os were classified as suggestions and the remaining 4 were classified as “best-practice.” No SRs were classified as “not reviewed” or “un-reviewed analysis method.”

The F&Os and the disposition of the F&Os, including the suggestion level ones, are provided in Table V-1 of Attachment V. All of the 65 SRs designated as findings were resolved through changes to the Fire PRA itself and/or the corresponding documentation.

The Fire PRA meets Capability Category II in most, but not all cases. A limited number of ASME/ANS areas were identified by the peer review team as either meeting Category I only requirements or as not met. These are provided in Table V-2 of Attachment V. The capability categories are defined in ASME/ANS RA-Sa-2009. These classifications were associated with F&Os that were resolved to meet the supporting requirement at Capability Category II.

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 collectively represent 95% of the calculated fire risk as well as the corresponding insights are included as 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 NMP1 NFPA 805 transition. Fire modeling tools were utilized within the Fire PRA only. The use of fire modeling tools within the Fire PRA is discussed in Attachment J.

4.5.2.2 Fire Risk Approach

Overview of Evaluation Process

The Fire Risk Evaluations were completed as part of the NMP1 NFPA 805 transition. These Fire Risk Evaluations were developed using the methodology described in Technical Report NMP1-FRE-F001. 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, as depicted in Figure 4-7. This is generally based on FAQ 07-0054 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 in 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. The defense-in-depth review 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 the safety margin 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 sufficient margin exists 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.

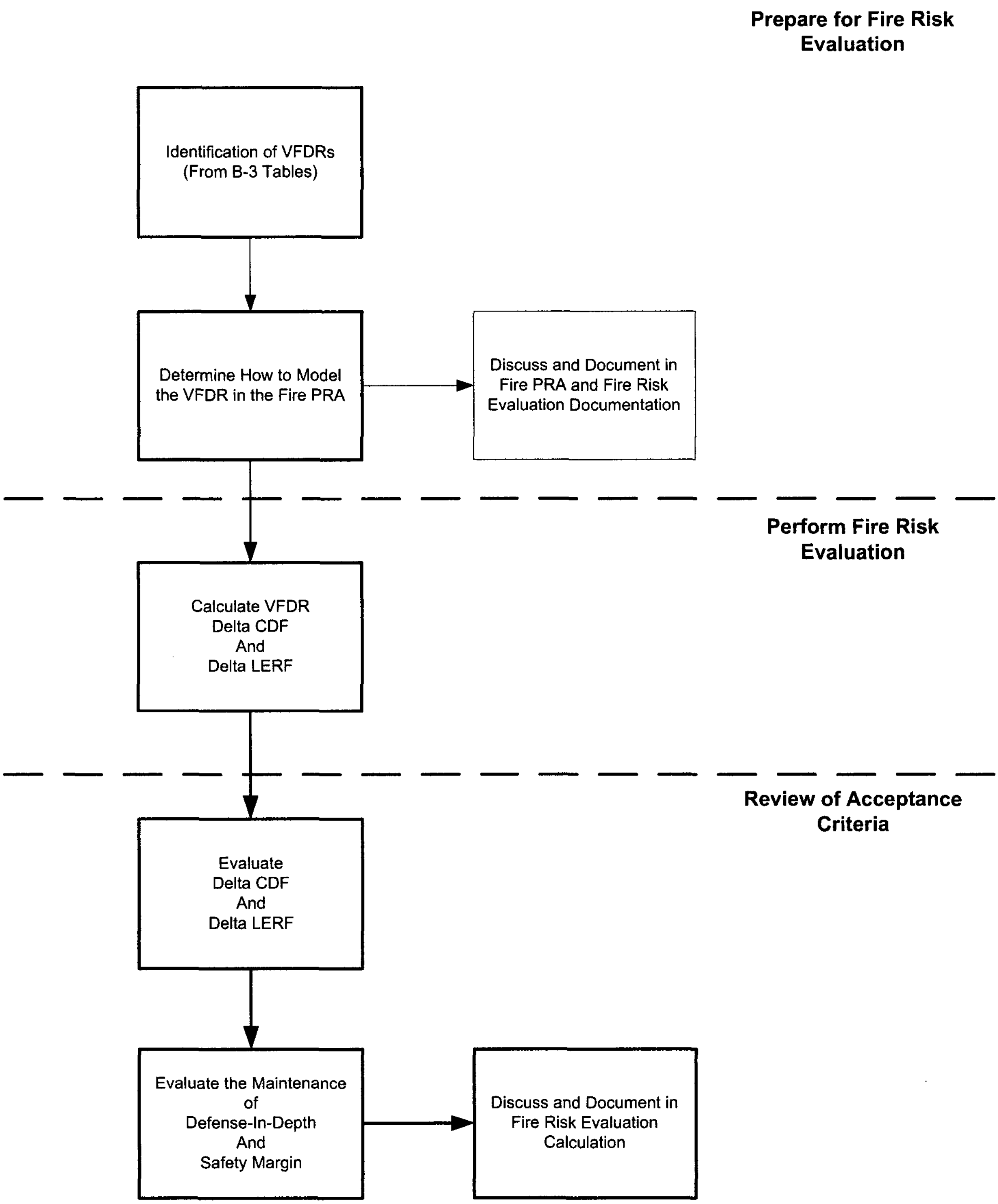

Figure 4-7 – Fire Risk Evaluation Process (NFPA 805 Transition) [Based on FAQ 07-0054 Revision 1]

Results of Evaluation Process

Disposition of VFDRs

The NMP1 existing post-fire SSA and the NFPA 805 transition project activities have identified variances from deterministic requirements of NFPA 805, Section 4.2.3. These variances were resolved using the Fire Risk Evaluation Process. Each variance resolved 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 NEI 04-02, Section 5.3.5 and RG 1.205. The results of these assessments 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 are provided. Note that the risk increase due to the use of any applicable recovery actions was included in the risk change for transition of each fire area.

This is consistent with RG 1.205, Section C.2.2.4.2 which 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 for the NFPA 805 Monitoring Program

Section 2.6 of NFPA 805 states:

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

The intent of the monitoring review is to confirm the adequacy of the existing surveillance, inspection, testing, compensatory measures, and oversight processes for transition to NFPA 805. This review considers the following:

- The adequacy of the scope of structures, systems and components within existing plant programs,
- The performance criteria for the availability and reliability of the required structures, systems and components, and
- The adequacy of the plant corrective action program in determining causes of equipment and programmatic failures and in minimizing their recurrence.

4.6.2 Overview of Post-Transition NFPA 805 Monitoring Program

This section provides an overview of the post-transition NFPA 805 Monitoring Program process. The Monitoring Program will be implemented after the safety evaluation issuance as part of the fire protection program transition to NFPA 805. The Monitoring Program described in this section is based on FAQ 10-0059, Revision 5. NMP1 will implement a Monitoring Program during implementation. See Implementation Item in Attachment S. 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 NMP1 NFPA 805 monitoring program 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:

- SSCs 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
  - SSCs 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 will 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 program activities and will be documented in the NFPA 805 Monitoring Program Engineering Evaluation.

1. Fire Protection Systems and Features

Those fire protection systems and features identified in Phase 1 would be 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 sufficient basis is documented.

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 protection systems and features that meet or exceed the criteria identified above will be included in the monitoring program contained in the site Maintenance Rule Program described in CNG-AM-1.01-1023, "Maintenance Rule Program." The remaining required fire protection systems and features will be monitored via the existing inspection and test program and in the existing system / program health program as described in CNG-AM-1.01, "Equipment Reliability," and CNG-AM-1.01-1000, "Equipment Reliability Process."

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 high safety significant (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 Program will be included in the Maintenance Rule Program. All NSCA equipment that are not HSS are considered LSS and need not be included in the monitoring program.

For fires originating during 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 programs 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 for radiological effluents (which are lower than releases due to core damage and containment breach) 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 programs is not considered necessary.

4. Monitoring of 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
- Fire Watch Programs; 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, and 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” will 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 NFPA 805 Monitoring Program Engineering Evaluation.

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 CNG-CA-1.01-1000, "Corrective Action Program" will be initiated to identify the negative trend. A corrective action plan will then be developed to ensure that 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 other established assessment activities. 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?

See Implementation Item in Attachment S.

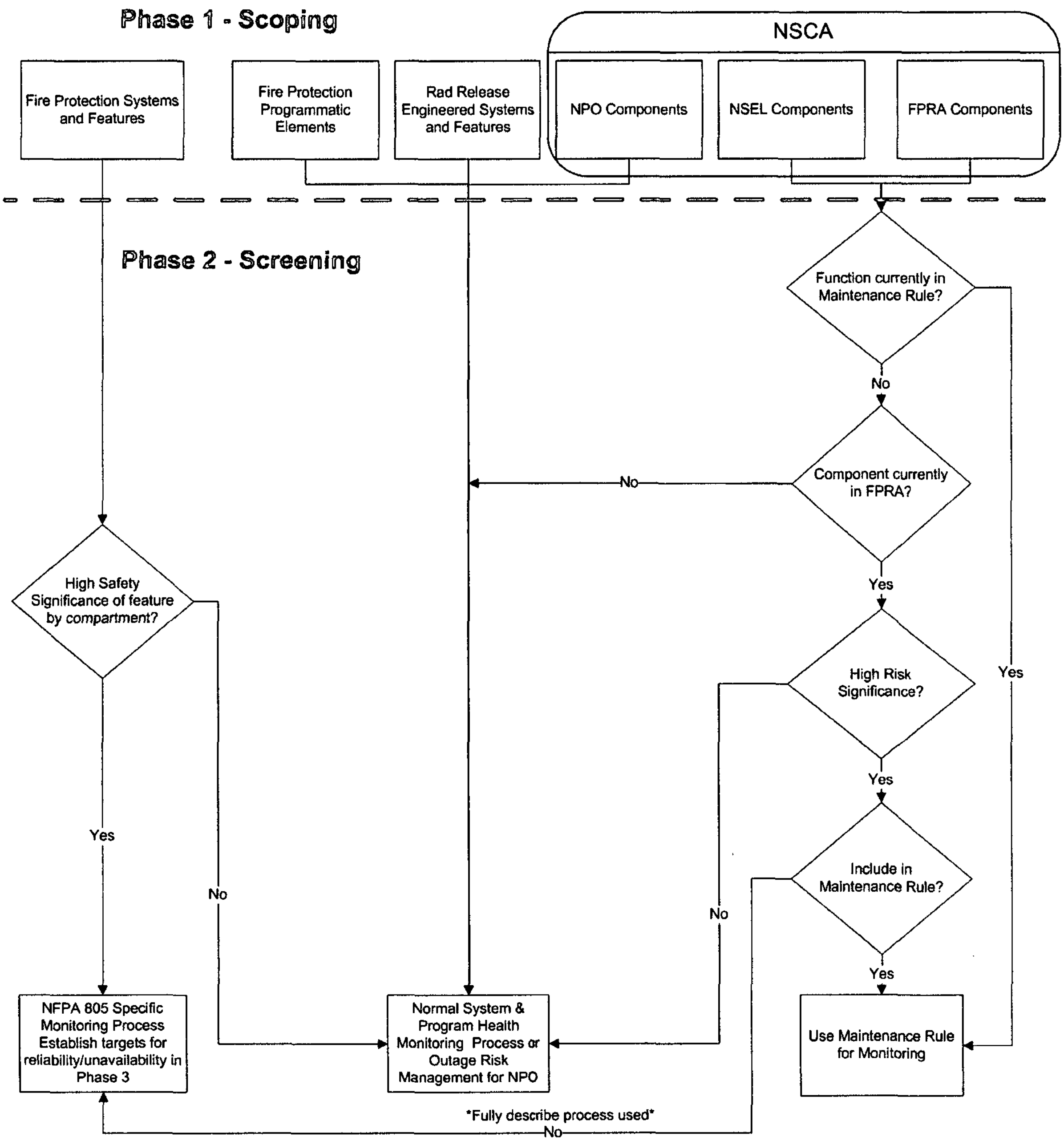

Figure 4-8 – NFPA 805 Monitoring – Scoping and Screening

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, NMP1 has documented analyses to support compliance with 10 CFR 50.48(c). The analyses are being performed in accordance with NMPNS 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 that 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. Appropriate cross references will be established to supporting documents as required by NMPNS processes. Figure 4-9 depicts the planned post-transition documentation and relationships.

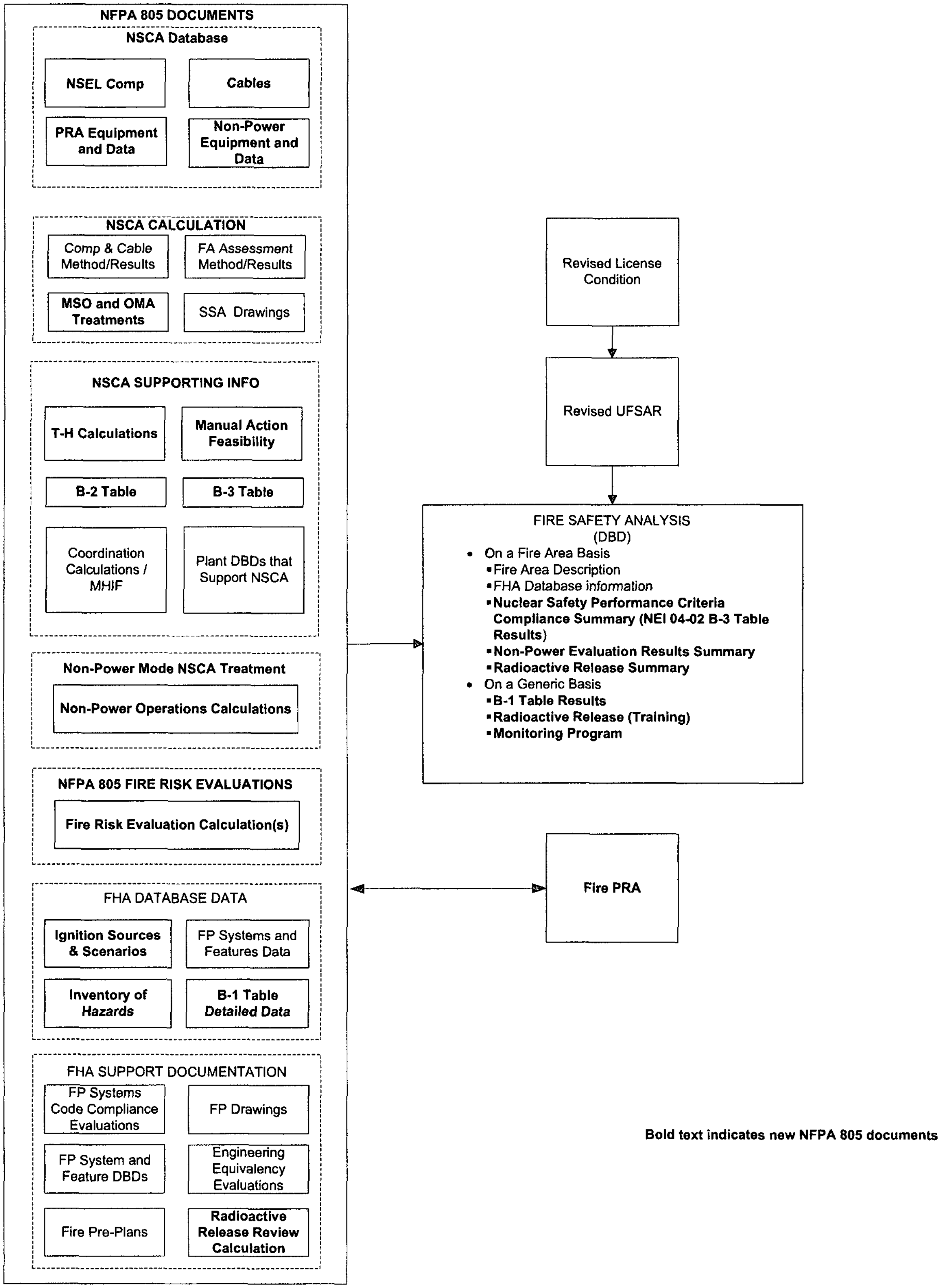

Figure 4-9 – NFPA 805 Planned Post-Transition Documents and Relationships

4.7.2 Compliance with Configuration Control Requirements in Sections 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 NMPNS 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, 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 that they are consistent with the defense-in-depth philosophy and that sufficient safety margins were maintained.

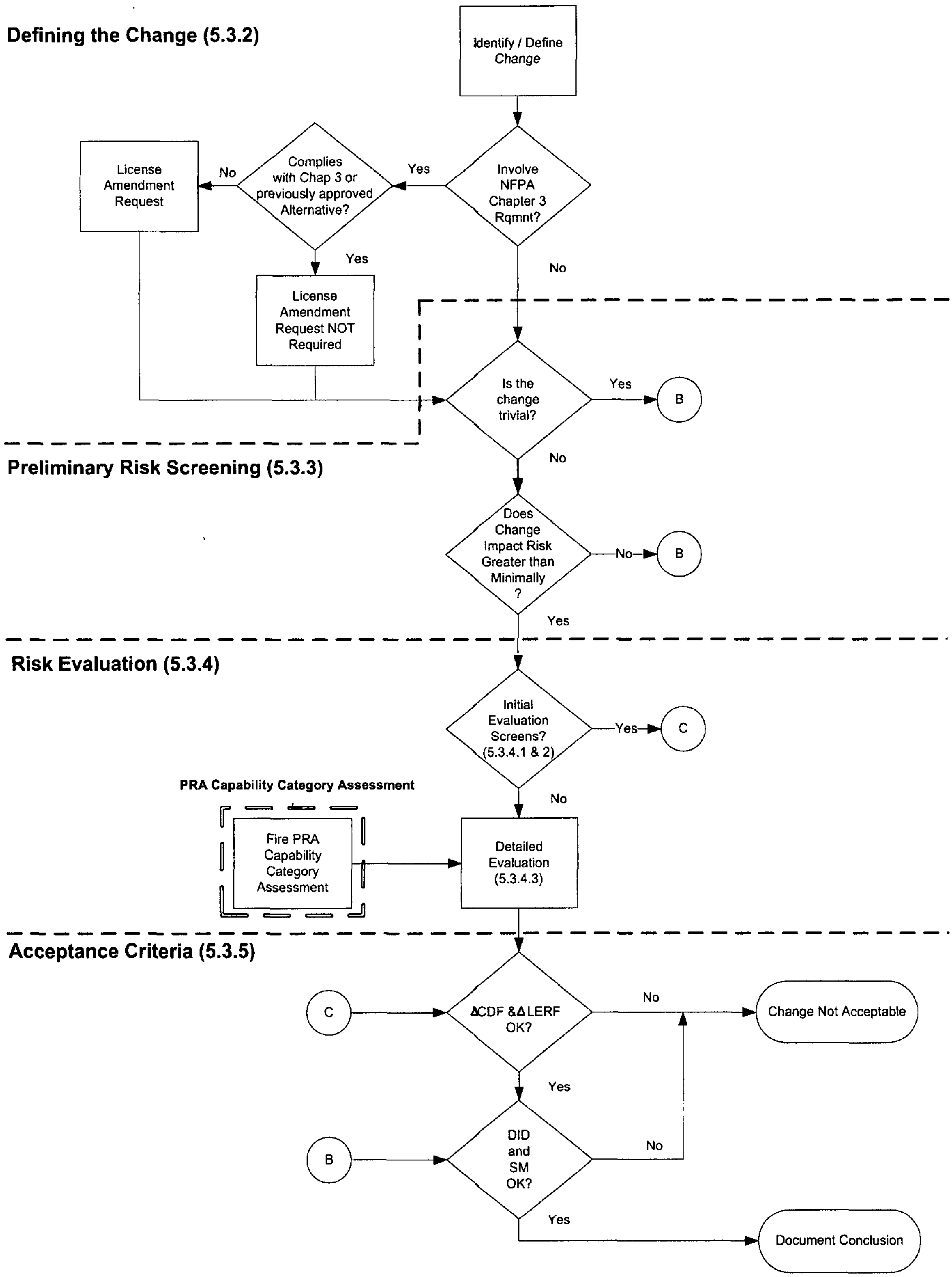

Figure 4-10 Plant Change Evaluation [NEI 04-02 Figure 5-1]
Note: References in Figure refer to NEI 04-02 Sections

The NMP1 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 NMPNS 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 Operational 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. See Implementation Items in Attachment S.

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 Section 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 require the use of qualified individuals, procedures that require independent review and verification of calculations, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when discrepancies are discovered.

Specifically, NMPNS evaluated potential impact on the FPP during transition to NFPA 805 through the implementation of CNG-CM-1.01-1003, "Design Engineering and Configuration Control." This document controls permanent and temporary design changes to the facility. It contains evaluation criteria that must be reviewed to determine if the design change can have potential FPP impact.

Changes were incorporated into CNG-CM-1.01-1003, as well as CNG-FES-007, "Preparation of Design Inputs and Change Impact Screen," to include questions specifically related to attributes that may impact NFPA 805. The evaluation criteria are reviewed during the development of the plant design change. If this evaluation identifies potential impact, then a more detailed review is performed by qualified fire protection, safe shutdown, and PRA personnel who are involved with the ongoing NFPA 805 transition activities.

The NMP1 NFPA 805 NSCA EIR was generated as part of the transition to NFPA 805. A final update and evaluation of plant changes is planned during the NFPA 805 implementation period and, in conjunction with CNG-CM-1.01-1003 and CNG-FES-007 change evaluation criteria, provides assurance that plant changes will be properly integrated into the NFPA 805 transition.

A review of implemented plant changes is also performed by the PRA organization to determine potential model impacts. A review of plant changes will be performed during the NFPA 805 implementation period to ensure that changes are appropriately evaluated for potential impact on the PRA model. This is done per CNG-CM-1.01-3004, "PRA Process for Internal Evaluations," and CNG-CM-1.01-1003, "Design Engineering and Configuration Control."

The Fire Protection Program licensing basis is also maintained through fire protection program reviews. NEP-FPP-01, "Appendix R Review," contains evaluation screening questions which, if impacted, requires a fire protection program evaluation by qualified fire protection or safe shutdown engineers. NEP-FPP-01 will be updated as part of the implementation period to address the NFPA 805 Change Evaluation Process. See Implementation Items in Attachment S.

In conclusion, the plant processes described above have been in place during the NFPA 805 transition to identify changes that may impact the fire protection program. Additionally, a comprehensive update of the NFPA 805 analyses is planned as part of the NFPA 805 implementation period to reflect the current plant configurations. The update will include review of plant configuration changes along with changes that may have occurred from RAI responses, updates from industry groups for MSO configurations, new or revised FAQs, and development of modifications. This final review will ensure current plant configurations are appropriately reflected and evaluated in the NFPA 805 documentation prior to full implementation of NFPA 805.

4.7.3 Compliance with Quality Requirements in Section 2.7.3 of NFPA 805

Fire Protection Program Quality

Current QA Program

The existing NMP1 Fire Protection QA program requirements are contained in the following documents:

- UFSAR Section XIII-A.3.0
- Quality Assurance Program Topical Report
- NDD-FPP, Nuclear Division Directive, Fire Protection Program

- UFSAR Appendix 10A, Section 2.3

QA Program Utilized During Transition

During the transition to 10 CFR 50.48(c), NMP1 performed work in accordance with the quality requirements of Section 2.7.3 of NFPA 805 and the existing Fire Protection QA Program described above. This included requirements that each analysis, calculation, or evaluation performed to support compliance with 10 CFR 50.48(c) be independently reviewed.

Post Transition QA Program

The existing NMP1 Fire Protection QA program will be utilized with the following changes:

- In addition to editorial and administrative changes (i.e. replacing references to previous NRC guidelines with those associated with the NFPA 805 transition and ensuring the features required for a performance based program under NFPA 805 are addressed), the components and systems currently considered within the scope of the NMP1 Fire Protection QA Program will be expanded to include those components and systems that are in the power block and are required by Chapter 4 of NFPA 805. This means that certain Fire Protection systems and features in some buildings not currently considered under the Fire Protection QA Program that are required by NFPA 805 Chapter 4 will now fall under the Fire Protection QA program. As such, any future modifications to these systems will be conducted under the design controls required by the Fire Protection QA program.
- The audit requirements will be revised to include the periodic review of the Monitoring Program.

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 5 of the ASME Standard for PRA Quality and ensures that NMP1 maintains an as-built, as-operated PRA model of the plant. The process has been peer reviewed. Fire PRA quality 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 independent review and verification of calculations, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when discrepancies 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 compliance with 10 CFR 50 Appendix B is applied to the PRA model calculations, as well.

With respect to QA Program requirements for independent reviews of calculations and evaluations, those existing requirements for Fire Protection Program documents will

remain unchanged. NMPNS 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 NMPNS 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. An uncertainty and sensitivity matrix was developed and included with the Uncertainty and Sensitivity Notebook. In addition, sensitivity to uncertainty associated with specific Fire PRA parameters was quantitatively addressed in the same notebook.

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 NMPNS 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

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 NMPNS 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) are used and were used appropriately as required by Section 2.7.3.3 of NFPA 805.

NFPA 805 Section 2.7.3.4 – Qualification of Users

Cognizant personnel who 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.

For personnel performing fire modeling or Fire PRA development and evaluation, NMPNS develops and maintains qualification requirements for individuals assigned various tasks. Position specific guides were 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.

NFPA 805 Section 2.7.3.5 – Uncertainty Analysis

Uncertainty analyses were performed as required by Section 2.7.3.5 of NFPA 805 and the results were considered in the context of the application. This is of particular interest in fire modeling and Fire PRA development.

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 Table 4-3. 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, 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
    - E – EEEE/LA Criteria: Systems/Features required for acceptability of Existing Engineering Equivalency Evaluations / NRC approved Licensing Action (i.e., Exemptions/Deviations/Safety Evaluations) (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.

In Attachment S, two tables are listed. Table S-1 identifies Plant Modifications required to be completed. Table S-2 identifies training, programs, personnel equipment, and document changes and upgrades required to be completed.

The Fire PRA model will represent the as-built, as-operated and maintained plant following completion of the risk related modifications identified in Attachment S. In the event the PRA model requires revision following completion of the modifications and implementation items listed in Attachment S, the changes will be controlled through normal NMPNS processes. 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.

4.8.3 Supplemental Information –Other Licensee Specific Issues

None

Table 4-3 Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features

Fire Area	Fire Zone	Description	NFPA 805 Regulatory Basis	Required Suppression System (S, E, R, D)[2]	Required Detection System (S, E, R, D)[2]	Required Fire Protection Feature (S, E, R, D)[2]	Required Fire Protection Feature and System Details[1]
1		**Reactor Building East EL 198-0 thru EL 340-0**	**4.2.4.2[3]**				
1	FBZR237N	Reactor Building EL 237-0 COL N-Q, ROW 8-9		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
1	FBZR261N	Reactor Building EL 261-0 COL N-Q, ROW 8-9		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
1	FBZR281N	Reactor Building EL 281-0 COL M-Q, ROW 6-7		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
1	FBZR281S	Reactor Building EL 281-0 COL K-L, ROW 7-8		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
1	FBZR298N	Reactor Building EL 298-0 COL N-Q, ROW 7.5-8.5		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
1	FBZR298S	Reactor Building EL 298-0 COL K-L, ROW 7-8		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
1	FBZR318N	Reactor Building EL 318-0 COL M-Q, ROW 6-7		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
1	FBZR318S	Reactor Building EL 318-0 COL K-M, ROW 6-7		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
1	FBZR340N	Reactor Building EL 340-0 COL M-Q, ROW 6-7		None	D	None	Detection
1	FBZR340S	Reactor Building EL 340-0 COL L-N, ROW 7-8		None	D	None	Detection
1	R1A	CTS Pump Room And General Floor Area East EL 198-0 & 237-0		D	R	None	Water Pre-Action Sprinkler, Detection
1	R1C	Access Stairwell Southeast EL 237-0 & 261-0		None	D	None	Detection
1	R1D	CS Pump Room And Protective Clothing Change Area EL 198-0 & 237-0		D	D	None	Water Pre-Action Sprinkler, Detection
1	R2A	General Floor Area East EL 261-0		D	R	None	Water Pre-Action Sprinkler, Detection
1	R3A	General Floor Area East EL 281-0		None	R	None	Detection
1	R4A	General Floor Area East EL 298-0		None	D	None	Detection
1	R4C	Emergency Condenser Isolation Valve Room EL 298-0		D	D	None	Halon Suppression System, Detection
1	R5A	General Floor Area East EL 318-0		None	D	None	Detection
1	R6A	General Floor Area East EL 340-0		None	D	None	Detection
2		**Reactor Building West EL 198-0 thru EL 340-0**	**4.2.4.2[3]**				
2	FBZR237N	Reactor Building EL 237-0 COL N-Q, ROW 8-9		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
2	FBZR261N	Reactor Building EL 261-0 COL N-Q, ROW 8-9		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
2	FBZR281N	Reactor Building EL 281-0 COL M-Q, ROW 6-7		S,R	S,R	None	Water Pre-Action Sprinkler, Detection

Table 4-3 Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features

Fire Area	Fire Zone	Description	NFPA 805 Regulatory Basis	Required Suppression System (S, E, R, D)[2]	Required Detection System (S, E, R, D)[2]	Required Fire Protection Feature (S, E, R, D)[2]	Required Fire Protection Feature and System Details[1]
2	FBZR281S	Reactor Building EL 281-0 COL K-L, ROW 7-8		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
2	FBZR298N	Reactor Building EL 298-0 COL N-Q, ROW 7.5-8.5		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
2	FBZR298S	Reactor Building EL 298-0 COL K-L, ROW 7-8		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
2	FBZR318N	Reactor Building EL 318-0 COL M-Q, ROW 6-7		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
2	FBZR318S	Reactor Building EL 318-0 COL K-M, ROW 6-7		S,R	S,R	None	Water Pre-Action Sprinkler, Detection
2	FBZR340N	Reactor Building EL 340-0 COL M-Q, ROW 6-7		None	D	None	Detection
2	FBZR340S	Reactor Building EL 340-0 COL L-N, ROW 7-8		None	D	None	Detection
2	R1B	CTS Pump Room, CS Pump Room, General Floor Area West EL 198-0 & 237-0		None	R	None	Detection
2	R2B	General Floor Area West EL 261-0		D	R	None	Water Pre-Action Sprinkler, Detection
2	R2C	Shutdown Cooling Room EL 261-0		None	D	None	Detection
2	R2D	Reactor Building Track Bay EL 261-0		D	None	None	Dry Pipe System
2	R3B	General Floor Area West EL 281-0		None	E,R	None	Detection
2	R4B	General Floor Area West EL 298-0		None	D	None	Detection
2	R4C	Emergency Condenser Isolation Valve Room EL 298-0		D	D	None	Halon Suppression System, Detection
2	R5B	General Floor Area West EL 318-0		D	D	None	Water Pre-Action Sprinkler, Detection
2	R6B	General Floor Area West EL 340-0		None	D	None	Detection
3		**Drywell EL 237-0 thru 318-0**	**4.2.3.1**				
3	R1	Drywell EL 237 – 318		None	None	None	
4		**Foam Room EL 261-0**	**4.2.4.2[3]**				
4	AB1F	Foam Room EL 261-0		None	R	None	Detection
5		**Turbine Building EL 240-0 thru 369-0**	**4.2.4.2[3]**				
5	FBZT261N	Turbine Building Fire Break Zone North EL 261-0		E,R	E,R	None	Water Pre-Action Sprinkler, Detection
5	FBZT261S	Turbine Building Fire Break Zone South EL 261-0		E,R	E,R	None	Water Pre-Action Sprinkler, Detection
5	OG1	General Floor Area EL 232-0		None	E,D	None	Detection

Table 4-3 Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features

Fire Area	Fire Zone	Description	NFPA 805 Regulatory Basis	Required Suppression System (S, E, R, D)[2]	Required Detection System (S, E, R, D)[2]	Required Fire Protection Feature (S, E, R, D)[2]	Required Fire Protection Feature and System Details[1]
5	OG2	General Floor Area EL 247-0		None	E,D	None	Detection
5	OG3	General Floor Area EL 261-0		E,D	E,R	None	Wet Pipe System, Detection
5	T1	Turbine Condenser/Heater Bay Area EL 250-0		E,R	E,R	None	Deluge System, CO_2 required for risk, Detection
5	T1A	Turbine Building EL 240-261 MSIV Room & Steam Tunnel		None	None	None	
5	T3A	General Floor Area East of MSIV Room and Fire Zone T1 EL 261-318		E,R,D	E,R	None	Water Pre-Action Sprinkler, Wet and Dry Pipe System, CO_2 required for DID, Detection
5	T3B	General Floor Area West of MSIV Room; Also South And West Of Fire Zone 1 EL 237-0 & 261-0		E,R,D	E,R	None	Water Pre-Action Sprinkler, CO_2 required for DID, Detection
5	T4A	General Floor Area East Of Fire Zone T1 EL 277-0		E,R	E,R	None	Water Pre-Action Sprinkler, Wet Pipe and Deluge System, Detection
5	T4B	General Floor Area West Of Fire Zone T1 EL 277-0		E,D	E,R	None	Water Pre-Action Sprinkler, Wet Pipe System, Detection
5	T4C	Hydrogen Seal Oil Unit Room EL 277-0		E,R,D	E,R	None	Deluge System, CO_2 required for DID, Detection
5	T4D	Battery Room EL 277		None	E,D	None	Detection
5	T5A	General Floor Area North EL 291-0		E,D	E,D	None	Wet Pipe System, Detection
5	T6A	General Floor Area North EL 305-6		E,D	E,R	None	Wet Pipe System, CO_2 required for DID, Detection
5	T6B	Turbine Laydown Area East EL 300-0		None	E,D	None	Detection
5	T6C	General Floor Area South EL 300-0		E,D	E,D	None	Deluge System, Detection
5	T6D	Mechanical Storage Area EL 300-0		E,D	E,D	None	Water Pre-Action Sprinkler, Detection
5	T7A	General Floor Area South EL 320-0		None	E,D	None	Detection
5	T8A	General Floor Area North EL 333-0, General Floor Area North EL 351-0, General Floor Area East EL 369		E,D	E,D	None	Wet Pipe System, Detection
5	T8B	General Floor Area West EL 369-0		None	E,D	None	Detection
6		**Turbine Building North EL 250-0**	**4.2.4.2[3]**				
6	T2A	Turbine Building EL 250-0		R	E,R	None	Water Pre-Action Sprinkler, Detection
7		**Turbine Building South & West EL 250-0**	**4.2.4.2[3]**				

Table 4-3 Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features

Fire Area	Fire Zone	Description	NFPA 805 Regulatory Basis	Required Suppression System (S, E, R, D)[2]	Required Detection System (S, E, R, D)[2]	Required Fire Protection Feature (S, E, R, D)[2]	Required Fire Protection Feature and System Details[1]
7	T2B	Turbine Building South And West EL 250-0		R	R	None	Water Pre-Action Sprinkler, Detection
7	T2E	UPS Battery Room EL 250		None	D	None	Detection
9		**Turbine Building East EL 250-0**	**4.2.4.2[3]**				
9	T2C	Turbine Building Offgas Tunnel EL 250-0		None	D	None	Detection
9	T2D	Turbine Building General Area East EL 250-0		R	R	None	Water Pre-Action Sprinkler, Detection
10		**Cable Spreading Room EL 250-0**	**4.2.4.2[3]**				
10	C1	Cable Spreading Room EL 250-0		E,R,D	E,R	None	Water Pre-Action Sprinkler, CO_2 required for DID, Detection
11		**Control Complex EL 261-0 And EL 277-0**	**4.2.4.2[3]**				
11	C2	Auxiliary Control Room, Computer Room 261-0		R,D	E,R	None	Halon Suppression System, CO_2 required for DID, Detection
11	C3	Control Room EL 277-0		None	E,R	None	Detection
12		**Administration Building EL 250-0**	**4.2.4.2[3]**				
12	AB1A	Records Storage Area EL 250-0		None	D	None	Detection
12	AB1B	SAS Equipment Area EL 252-0		D	D	None	Halon Suppression System, Detection
12	AB1C	CPU Equipment Area EL 252-0		D	D	None	Halon Suppression System, Detection
12	AB1D	General Area EL 250-0		D	D	None	Wet Pipe System, Detection
12	AB1E	Locker Area, Lunch Room, Offices EL 261-0		D	D	None	Wet Pipe System, Detection
12	AB2A	Access Passageway EL 248-0		D	None	None	Wet Pipe System
12	AB2B	Technical Support Area EL 248-0		D	D	None	Wet Pipe System, Detection
12	AB2C	Radiation Records Area EL 248-0		D	D	None	Wet Pipe System, Detection
12	AB2D	Warehouse Area EL 248-0		D	None	None	Wet Pipe System
12	AB3A	Warehouse Area EL 261-0		D	None	None	Wet Pipe System
12	AB3B	Oil Storage Room EL 261-0		D	None	None	Wet Pipe System
12	AB3C	Storeroom Truck Dock EL 261-0		D	None	None	Dry Pipe System

Table 4-3 Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features

Fire Area	Fire Zone	Description	NFPA 805 Regulatory Basis	Required Suppression System (S, E, R, D)[2]	Required Detection System (S, E, R, D)[2]	Required Fire Protection Feature (S, E, R, D)[2]	Required Fire Protection Feature and System Details[1]
12	AB3D	Electrical/Mechanical Shop Area, Office Areas, Locker Rooms EL 261-0		D	D	None	Wet Pipe System, Detection
12	AB3E	Telephone Room 1 EL 261-0		D	D	None	Halon Suppression System, Detection
12	AB3F	Telephone Room 2 EL 261-0		D	D	None	Halon Suppression System, Detection
12	AB4A	General Office Area EL 277-0		D	D	None	Wet Pipe System, Detection
12	AB4B	File Room EL 277-0		D	D	None	Water Pre-Action Sprinkler, Detection
12	AB4C	Records Processing Area EL 277-0		R	R	None	Water Pre-Action Sprinkler, Detection
12	AB4D	General Office Area EL 277-0		R	R	None	Wet Pipe System, Detection
12	AB5	Penthouse Ventilation Room EL 290-0		D	D	None	Deluge System, Detection
13		**Screenhouse**	**4.2.4.2[3]**				
13	S1	Screenhouse EL 225-0 – 256-0		S,D	R	None	Dry Pipe System, Detection
14		**Diesel Fire Pump Room EL 261-0**	**4.2.4.2[3]**				
14	S2	Diesel Fire Pump Room EL 256-0		S,D	S,D	None	Dry Pipe System, Detection
15		**Radwaste And Waste Disposal Buildings EL 252-0 thru 292-0**	**4.2.4.2[3]**				
15	RS1A	Drum Waste Storage Vaults EL 252-0		None	None	None	
15	RS1B	Electrical Equipment Room EL 252-0		D	D	None	Halon Suppression System, Detection
15	RS1C	General Floor Area South, Drum Storage Room EL 252-0		None	D	None	Detection
15	RS2A	Truck Loading Area, North EL 261-0		D	None	None	Dry Pipe System
15	RS2B	Truck Loading Area, West EL 261-0		D	None	None	Dry Pipe System
15	RS2C	General Floor Area EL 261-0		D	D	None	Dry-pipe System, Detection
15	RS2D	Radwaste Control Room, West EL 261-0		D	D	None	Halon Suppression System, Detection
15	RS2E	General Floor Area, South EL 261-0		None	None	None	
15	RS3A	General Floor Area, West EL 281-0		None	D	None	Detection

Table 4-3 Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features

Fire Area	Fire Zone	Description	NFPA 805 Regulatory Basis	Required Suppression System (S, E, R, D)[2]	Required Detection System (S, E, R, D)[2]	Required Fire Protection Feature (S, E, R, D)[2]	Required Fire Protection Feature and System Details[1]
15	RS4A	General Floor Area, Northwest EL 292-0		D	D	None	Deluge System, Detection
15	RS5B	General Floor Area, Southwest EL 292-0		None	D	None	Detection
15	WD1	General Area EL 225-0 & 229-0		None	D	None	Detection
15	WD2	General Area EL 247-0		None	D	None	Detection
15	WD3A	General Area EL 261-0		D	D	None	Water Pre-Action Sprinkler, Detection
15	WD3B	Radwaste Control Room EL 261-0		None	D	None	Detection
15	WD3C	Baler Room EL 261-0		D	D	None	Dry Pipe System, Detection
15	WD3D	Dow Solidification Area EL 261-0		D	D	None	Dry Pipe System, Detection
15	WD3E	Truck Bay EL 261-0		E,D	E,D	None	Dry Pipe System, Detection
15	WD4	Waste Building Ventilation Area EL 277-0		None	D	None	Detection
16A		**Battery Board Room 12 EL 261-0**	**4.2.4.2[3]**				
16A	B1A	Battery Board Room 12 EL 261-0		None	E,D	None	Detection
16B		**Battery Board Room 11 EL 261-0**	**4.2.4.2[3]**				
16B	B1B	Battery Board Room 11 EL 261-0		None	E,D	None	Detection
17A		**Battery Room 12 EL 277-291**	**4.2.4.2[3]**				
17A	B2A	Battery Room 12 EL 277-0		None	E,D	None	Detection
17B		**Battery Room 11 EL 277-291**	**4.2.4.2[3]**				
17B	B2B	Battery Room 11 EL 277-0		None	E,D	None	Detection
18		**Emergency Diesel Generator 102 Missile Enclosure EL 271**	**4.2.4.2[3]**				
18	D3	EDG 102 Control Cable Missile Enclosure EL 271-0		None	E,D	None	Detection
19		**Emergency Diesel Generator 103 Foundation Room EL 250-0 and Diesel Generator Room El 261-0**	**4.2.4.2[3]**				
19	D1A	EDG 103 Foundation Room EL 250-0		E,D	E,D	None	Water Pre-Action Sprinkler, Detection
19	D2A	EDG 103 Room EL 261-0		R	E,R	None	CO_2 System, Detection
20		**Diesel Generator Enclosed Cableway EL 250-0**	**4.2.4.2[3]**				

Table 4-3 Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features

Fire Area	Fire Zone	Description	NFPA 805 Regulatory Basis	Required Suppression System (S, E, R, D)[2]	Required Detection System (S, E, R, D)[2]	Required Fire Protection Feature (S, E, R, D)[2]	Required Fire Protection Feature and System Details[1]
20	D1C	EDG 103 Cable Routing Area EL 250-0		D	D	None	Water Pre-Action Sprinkler, Detection
21		**Below Powerboards 102 & 103 EL 250-0**	**4.2.4.2[3]**				
21	D1D	Room Below PB's 102 & 103 EL 250-0		D	D	None	Water Pre-Action Sprinkler, Detection
22		**Emergency Diesel Generator 102 Foundation Room EL 250-0 and Diesel Generator Room EL 261-0**	**4.2.4.2[3]**				
22	D1B	EDG 102 Foundation Room EL 250-0		E,D	E,D	None	Water Pre-Action Sprinkler, Detection
22	D2B	EDG 102 Room EL 261-0		R	E,R	None	CO_2 System, Detection
23		**Power Board 102 Room EL 261-0**	**4.2.4.2[3]**				
23	D2C	Power Board 102 Room EL 261-0		D	E,R	None	CO_2 System, Detection
24		**Power Board 103 Room EL 261**	**4.2.4.2[3]**				
24	D2D	Power Board 103 Room EL 261-0		D	E,R	None	CO_2 System, Detection
EXT		**External to Plant**	**4.2.3.1[3]**				
EXT	EXT	External to Plant		E	E	None	Deluge System, Detection

Notes:
1. Refer to Attachment C for each area and additional information
2. NR – Not Required; S – Required for Separation; E – Required for Engineering Evaluation; R – Required for Risk; D – Required for Defense-in-Depth
3. Compliance includes reliance on simplifying deterministic assumptions

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 NMP1 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 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."*

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 FAQ 07-0032, Clarification of 10 CFR 50.48(c), 10 CFR 50.48(a) and GDC 3 (ML081400292). The following tables provide a cross reference of fire protection regulations associated with the post-transition NMP1 fire protection program and applicable industry and NMP1 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.* CNG-PR-3.01-1000, "Record Management"

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. NMP1 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.
	(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.	Not applicable to NMP1 (BWR).
	(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 NEI 04-02 Table B-1.
	(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.	NMP1 complies by previous NRC approval. See NEI 04-02 Table B-1.

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

	10 CFR 50.48(c) Section(s)	Applicability/Compliance Reference
	(vii) Performance-based methods. Notwithstanding the prohibition in Section 3.1 against the use of performance-based methods, the fire protection program elements and minimum design requirements of Chapter 3 may be subject to the performance-based methods permitted elsewhere in the standard. Licensees who wish to use performance-based methods for these fire protection program elements and minimum design requirements shall submit a request in the form of an application for license amendment under § 50.90. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the performance-based approach; (A) Satisfies the performance goals, performance objectives, and *performance criteria specified in* NFPA 805 *related to nuclear safety and* radiological release; (B) Maintains safety margins; and (C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).	The use of performance-based methods for NFPA 805 Chapter 3 is requested. See Attachment L.
(3)	*Compliance with NFPA 805.*	See below
	(i) A licensee may maintain a fire protection program that complies with NFPA 805 as an alternative to complying with paragraph (b) of this section for plants licensed to operate before January 1, 1979, or the fire protection license conditions for plants licensed to operate after January 1, 1979. The licensee shall submit a request to comply with NFPA 805 in the form of an application for license amendment under § 50.90. The application must identify any orders and license conditions that must be revised or superseded, and contain any necessary revisions to the plant's technical specifications and the bases thereof. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the licensee has identified orders, license conditions, and the technical specifications that must be revised or superseded, and that any necessary revisions are adequate. Any approval by the Director or the designee must be in the form of a license amendment approving the use of NFPA 805 together with any necessary revisions to the technical specifications.	The 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 NMP1 fire protection License Condition 2.D(7) 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

NMPNS conducted a review of the Technical Specifications to determine which Technical Specifications are required to be revised, deleted, or superseded. NMPNS determined that the changes to the Technical Specifications and applicable justification listed in Attachment N are adequate for the NMP1 adoption of the new fire protection licensing basis.

5.2.3 Orders and Exemptions

A review was conducted of the NMP1 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. NMPNS 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 NMP1 UFSAR

After the approval of the LAR, in accordance with 10 CFR 50.71(e), the NMP1 UFSAR will be revised. The format and content will be consistent with FAQ 12-0062.

5.5 Transition Implementation Schedule

The following schedule for transitioning NMP1 to the new fire protection licensing basis requires NRC approval of the LAR in accordance with the following schedule:

- Implementation of the new NFPA 805 fire protection program to include procedure changes, process updates, and training to affected plant personnel. This will occur 180 days after issuance of the license amendment unless that date falls within a scheduled refueling outage. Then, implementation will occur 60 days after startup from that scheduled refueling outage. See Attachment S, Table S-2.
- Attachment S provides a listing of plant modifications associated with the transition to NFPA 805. NMPNS will complete implementation of these modifications no later than the end of the first NMP1 refueling outage following issuance of the license amendment. No compensatory measures are required relative to the modifications listed in Attachment S, Table S-1.

6.0 REFERENCES

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

6.1 NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition"

6.2 NEI 04-02, Revision 2, "Guidance for Implementing A Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)"

6.3 Regulatory Guide 1.205, Revision 1, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants"

6.4 CGG letter to NRC, "Letter of Intent to Adopt NFPA 805 – Performance-Based Standard for Fire Protection for Light Water Reactor Generation Plants, 2001 Edition," April 17, 2006 (ML061150157)

6.5 NRC letter to CGG, "Response to Letter of Intent to Adopt National Fire Protection Association (NFPA) Standard 805, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, and Nine Mile Point Nuclear Station, Unit Nos. 1 and 2," May 31, 2006 (ML061350098)

6.6 NMPNS letter to NRC, "Request for Extension of Enforcement Discretion for a Revised Date of 10 CFR 50.48(c) License Amendment Request Submittal," January 16, 2009 (ML090230353)

6.7 NRC letter to NMPNS, "Nine Mile Point Nuclear Station, Unit No. 1 – Evaluation of the Request for an Extension of Enforcement Discretion in Accordance with the Interim Enforcement Policy for Fire Protection Issues During Transition to National Fire Protection Standard NFPA 805," March 9, 2009 (ML090620089)

6.8 CENG letter to NRC, "Request for Extension of Fire Protection Enforcement Discretion," June 20, 2011 (ML11179A032)

6.9 NRC letter to CENG, "Commitment to Submit a License Amendment Request to Transition to 10 CFR 50.48(c), National Fire Protection Association Standard NFPA 805, and Request to Extend Enforcement Discretion – Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Nine Mile Point Nuclear Station, Unit No. 1, and R.E. Ginna Nuclear Power Plant," July 28, 2011 (ML112000464)

6.10 NRC letter to NMPC, "Exemption Request, Nine Mile Point Nuclear Station, Unit No. 1, Docket No. 50-220," March 21, 1983 (ML010990035)

6.11 NRC letter to NMPC, "Fire Protection Safety Evaluation Report, Nine Mile Point – Unit 1, Docket No. 50-220," July 26, 1979 (ML010990290)

6.12 NRC letter to NMPC, "Amendment 71 Addition of Technical Specifications for Remote Shutdown Panels," April 1, 1985 (ML010990284)

6.13 NRC letter to NMPC, "Safety Evaluation Report (SER), Nine Mile Point Nuclear Station, Unit No. 1, Docket No. 50-220," August 6, 1986

6.14 NRC letter to NEI, "Process for Frequently Asked Questions for Title 10 of the Code of Federal Regulations, Part 50.48(c) Transitions," July 12, 2006 (ML061660105)

6.15 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)

6.16 EIR 51-9077683-001, "NFPA 805 Fundamental FPP and Design Elements Transition Review"

6.17 NFPA 10, Portable Fire Extinguishers, 1998 Edition

6.18 NFPA 12, Carbon Dioxide Extinguishing Systems, 1968, 1977, 1980, 2005, and 2008 Editions

6.19 NFPA 12A, Halon 1301 Fire Extinguishing Systems, 1980 Edition

6.20 NFPA 13, Automatic Sprinkler Systems, 1969, 1980, and 1996 Editions

6.21 NFPA 14, Standpipe and Hose Systems, 1963 Edition

6.22 NFPA 15, Water Spray Fixed Systems, 1969 Edition

6.23 NFPA 20, Centrifugal Fire Pumps, 1965 Edition

6.24 NFPA 24, Private Fire Mains, 1973 Edition

6.25 NFPA 30, Flammable and Combustible Liquids Code, 2000 Edition

6.26 NFPA 50A, Hydrogen Systems at Consumer Sites, 1999 Edition

6.27 NFPA 72A, Local Protective Signaling Systems, 1964 Edition

6.28 NFPA 72E, Automatic Fire Detectors, 1974 Edition

6.29 NFPA 600, Industrial Fire Brigades, 2000 Edition

6.30 NEI 00-01, Revision 2, "Guidance for Post-Fire Safe Shutdown Circuit Analysis"

6.31 EIR 51-9133191, "Nine Mile Point Unit 1 – Nuclear Safety Capability Assessment"

6.32 EIR 51-9156521-000, "Recovery Action Review for Nine Mile Point Nuclear Power Station Unit 1 Transition to NFPA 805"

6.33 Technical Report on Identification and Classification of the NMP1 MSO Scenarios Using an Expert Panel – Review of New Generic Scenarios, 2008

6.34 Technical Report on Identification and Classification of the NMP1 MSO Scenarios Using an Expert Panel – Review of New Generic Scenarios, 2010

6.35 Technical Report on Identification and Classification of the NMP1 MSO Scenarios Using an Expert Panel – Review of New Generic Scenarios, May, 2012

6.36 EIR 51-9172752-001, "Nine Mile Point Unit 1 NFPA 805 Transition B-3 Table/Report"

6.37 EIR 51-9137629-000, "Nine Mile Point 1, Non-Power Operations KSF Equipment List"

6.38 EIR 51-9171174-000, "NMP1 – Nuclear Power Station – NFPA 805 Transition – Non-Power Operation Component Pinch Point Analysis"

6.39 EIR 51-9085686-002, "NMP-1 NFPA 805 Radiological Release Transition Review"

6.40 NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities"

6.41 Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"

6.42 NUREG-1824, Volume 3, "Verification & Validation of Selected Fire Models for Nuclear Power Plant Applications, Volume 3: Fire Dynamics Tools (FDTS)," (EPRI 1011999), Salley, M. H. and Kassawara, R. P., Final Report, May, 2007

6.43 NUREG-1824, Volume 5, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications Volume 5: Consolidated Fire Growth and Transport Model," (EPRI 1011999), Salley, M. H. and Kassawara, R. P., Final Report, May, 2007

6.44 NUREG -1934, "Nuclear Power Plant Fire Modeling Applications Guide (NPP FIRE MAG)," Draft Report for Public Comment, (EPRI 1023259), Nuclear Regulatory Commission, Rockville, MD, 2011

6.45 Volume 2 Detailed Methodology, (EPRI 1008239), Final Report, September, 2005 NUMARC 93-01, Revision. 3, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

6.46 NEI 07-12, Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines, NEI, Rev. 0, November 2008.

6.47 NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management"

6.48 ASME/ANS Ra-Sa-2009, Addenda to ASME/ANS Ra-Sa-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, American Society of Mechanical Engineers/American Nuclear Society, New York, NY

6.49 EPRI Fire Protection Equipment Surveillance Optimization and Maintenance Guide 1006756, Final Report July 2003

6.50 PRA Notebook, N1-PP-F001, PP - Plant Partitioning

6.51 PRA Notebook, N1-ES-F001, ES – Equipment Selection

6.52 PRA Notebook, N1-CS-F001, CS – Cable Selection, Detailed Circuit Analysis and Route Location (CS)

6.53 PRA Notebook, N1-QLS-F001, QLS – Qualitative Screening

6.54 PRA Notebook, N1-PRM-F001, PRM – Plant Response Modeling

6.55 PRA Notebook, N1-IGN-F001, IGN – Fire Ignition Frequencies

6.56 PRA Notebook, N1-QNS-F001, QNS – Quantitative Screening

6.57 PRA Notebook, N1-FRE-F001, Fire Risk Evaluation

6.58 PRA Notebook, N1-CF-F001, CF – Circuit Failure Probability

6.59 PRA Notebook, N1-FSS-F001, FSS – Detailed Fire Modeling

6.60 PRA Notebook, N1-FSS-F002, FSS – Fire Scenario Selection – Structure Steel

6.61 PRA Notebook, N1-FSS-F003, FSS – Fire Scenario Selection – Multi-compartment

6.62 PRA Notebook, N1-FSS-F004, FSS – Fire Scenario Selection – MCR

6.63 PRA Notebook, N1-FSS-F005, FSS –Control Room Abandonment

6.64 PRA Notebook, N1-HRA-F001 – HRA – Post-fire HRA

6.65 PRA Notebook, N1-SF-F001 – FS - Seismic Fire

6.66 PRA Notebook, N1-FQ-F001 – FQ – Fire Quantification

6.67 PRA Notebook, N1-UNC-F001 – UNC – Uncertainty & Sensitivity

ATTACHMENTS

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

66 Pages Attached

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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. See sub-sections for specific compliance statements and references.	
3.2 Fire Protection Plan	N/A	N/A - Section Title, no technical requirements. See sub-sections for specific compliance statements and references.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	The Nuclear Division Directive (NDD) states its purpose is, "to establish requirements for development and execution of the NMP FPP, and to assign departmental responsibility for implementation of those requirements." It also states, "the Fire Protection Program implements requirements specified in documents referenced in this Directive." The FPP is a program to implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR, and as approved in the FPSER dated July 26, 1979, and in the fire protection Exemption issued March 21, 1983. Noncompliance's with the above described FPP that adversely affect the ability to achieve and maintain safe shutdown in the event of a fire shall be reported in accordance with the requirements of 10 CFR 50.72 and 10 CFR 50.73. In addition the FPP key elements are: Implementing Procedures, Administrative Controls, Fire Protection System Drawings and Calculations, Fire Protection Engineering Evaluations, (FPEEs), Monitoring and Evaluation Program, Quality Assurance Program, Fire Brigade Manning, Training, Drills, and Responsibilities, and Surveillance and Tests.	NDD-FPP, "Fire Protection Program," Rev.01400, Sec. 1.0, 3.2.2 UFSAR Sec. X.N, Appendix 10A, "Fire Hazards Analysis," Rev. 21, Sec. 2.1, pgs. 10A-8 – 10A-13 UFSAR Sec. X.K, "Fire Protection Program," Rev. 21, Sec. 2.0, pgs. X-46 – X-48
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	NDD-FPP and NIP-FPP-01 serve as the policy documents that establish and delineate responsibilities and authorities for implementation and administration of the FPP.	NDD-FPP, "Fire Protection Program," Rev. 01400, Sec. 4.0 NIP-FPP-01, "Fire Protection Program," Rev. 01700, Sec. 2.0

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.2.2.1 The policy document shall designate the senior management position with immediate authority and responsibility for the fire protection program.	Complies	The FHA serves as the policy document that designates the senior management position with immediate authority and responsibility for the FPP and states: "The Site Vice President has the overall responsibility for the fire protection program at Nine Mile Point site."	UFSAR, Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.1.1.1, pg. 10A-8
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	NIP-FPP-01 and the FHA serve as the policy documents that assign daily administration, coordination, and implementation responsibilities to the Fire Protection Program Manager.	NIP-FPP-01, "Fire Protection Program," Rev. 01700, Sec. 2.5 UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.1.1.1, pg. 10A-11
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	NIP-FPP-01 serves as the policy document that defines the fire protection interfaces with other organizations and assigns responsibilities for coordination of activities. Additionally, NIP-FPP-01 identifies the plant positions having authority for implementing the various areas of the FPP.	NIP-FPP-01, "Fire Protection Program," Rev. 01700, Sec. 2.0
3.2.2.4 The policy document shall identify the appropriate AHJ for the various areas of the fire protection program.	Complies with item for implementation	Implementation Item: Revise the fire protection policy document to identify the appropriate AHJ for the various areas of the program. See Implementation Item in Attachment S.	
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	Procedures are established for the implementation of the fire protection program as described in the FPP. *See sub-paragraphs for specific compliance* statements and references for the elements below.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.2.3 Procedures (1)* Inspection, testing, and maintenance for fire protection systems and features credited by the fire protection program	Complies with item for implementation	NDD-FPP states, "Site personnel are responsible for ensuring, through planned system surveillance and preventive maintenance, that the Engineering-defined systems and components specific to the Fire Protection Program are maintained." The Fire Preventative Maintenance and Fire Surveillance Test Procedures implement required inspection, testing, and maintenance activities. Surveillance frequencies are outlined in the Fire Plan and may be modified in accordance with the methodology in EPRI Report TR1006756, Fire Protection Equipment Surveillance Optimization and Maintenance Guide. Implementation item: Performance-based surveillance frequencies will be updated as described in Electric Power Research Institute (EPRI) Technical Report TR-1006756, "Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features." See Implementation Item in Attachment S.	NDD-FPP, Rev. 01400, Sec. 4.2.1 Fire Preventative Maintenance (FPM) and Fire Surveillance Test (FST) series procedures.
3.2.3 Procedures (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	S-SAD-FPP-0105 states that the purpose of the procedure is, "To establish the compensatory measures to be taken in the event of planned or unplanned inoperable fire protection systems or components." The procedure also limits impairment duration.	S-SAD-FPP-0105, "Compensatory Measures for Inoperable Fire Protection Systems and Components," Rev.01800, Sec. 1.0, 3.0

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.2.3 Procedures (3)* Reviews of fire protection program – related performance and trends	Complies with item for implementation	Reviews of fire protection program-related performance and trends are performed in accordance with NIP-FPP-01, NEP-FPP-02, and CNG-CA-2.01-1000. Implementation Item - The monitoring program required by NFPA 805 will include a process that monitors and trends the fire protection program based on specific goals established to measure effectiveness. See Implementation Item in Attachment S.	NIP-FPP-01, Rev. 01700, Sec. 2.3.3, 2.6 NEP-FPP-02, "Fire Protection Self-Assessments," Rev. 04, Sec. 3.0 CNG-CA-2.01-1000, "Self-Assessment and Benchmarking Process," Rev. 00400, All
3.2.3 Procedures (4) Reviews of physical plant modifications and procedure changes for impact on the fire protection program	Complies	Reviews of plant modifications and procedure changes for FPP impact are governed by NIP-FPP-01, CNG-CM-1.01-1003, CNG-FES-007, CNG-FES-015, and NEP-FPP-01.	NIP-FPP-01, Rev. 01700, Sec. 2.12 CNG-CM-1.01-1003, "Design and Configuration Control Process," Rev. 00401 CNG-FES-007, "Preparation of Design Inputs and Change Impact Screen, Rev. 00010, Sec. 5.4.16 CNG-FES-015, "Design Engineering and Configuration Management Forms," Rev. 00004, Forms 16, 17 NEP-FPP-01, "Fire Protection Engineering," Rev. 01500, Sec. 5.5.B
3.2.3 Procedures (5) Long-term maintenance and configuration of the fire protection program	Complies	Procedural requirements have been established for long-term maintenance and configuration of the FPP per NIP-FPP-01 and CNG-CM-1.01-1003.	NIP-FPP-01, Rev. 01700, Sec. 2.7 CNG-CM-1.01-1003, "Design Engineering and Configuration Control," Rev. 00401, Attach. 12
3.2.3 Procedures (6) Emergency response procedures for the plant industrial fire brigade	Complies	Emergency response procedures for the plant have been established per EPIP-EPP-28 and N1-PFP-0101.	EPIP-EPP-28, "Fire Fighting," Rev. 01500, Sec. 4.0 N1-PFP-0101, "Unit 1 Pre-Fire Plans," Rev. 00000, All

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	A fire prevention program with the goal of preventing a fire from starting has been established, documented, and implemented as part of the fire protection program. As described in sub-sections 1 and 2 below, controls on plant operational activities and design controls on restricting use of combustible materials are provided.	NDD-FPP, Rev. 01400, Sec. 3.2.1
3.3 Prevention (1) Prevention of fires and fire spread by controls on operational activities	Complies	Controls on operational activities to prevent fires and fire spread are imposed by GAP-HSC-01 and GAP-FPP-02.	GAP-HSC-01, "Housekeeping, Tours, and Inspections," Rev. 15, Sec. 3.0 GAP-FPP-02, "Control of Hot Work," Rev. 01300, Sec. 4.0
3.3 Prevention (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	Design controls that restrict use of combustible materials are imposed by NEP-FPP-01, CNG-CM-1.01-1003, CNG-FES-007, and CNG-FES-015.	NEP-FPP-01, "Fire Protection Engineering," Rev. 01500, Sec. 5.5.B, Attach. 2, 3 CNG-CM-1.01-1003, "Design Engineering and Configuration Control," Rev. 00401, Attach. 12 CNG-FES-007, "Preparation of Design Inputs and Change Impact Screen," Rev. 00010, Sec. 5.4.16 CNG-FES-015, "Design Engineering and Configuration Management Forms," Rev. 00004, Forms 16, 17

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	GAP-FPP-02, GAP-INV-02, and GAP-HSC-01 adequately control ignition sources and the use of transient combustible materials during all aspects of plant operation. S-SAD-FPP-0101, EPIP-EPP-28 and NIP-FPP-01 ensure that fires that may occur are kept as small as possible.	GAP-FPP-02, "Control of Hot Work," Rev. 01300, Sec. 4.0 GAP-INV-02, "Control of Material Storage Areas," Rev. 02400, Sec. 4.0, Attach. 4, 5 GAP-HSC-01, "Housekeeping, Tours, and Inspections," Rev. 15, Sec. 3.0, S-SAD-FPP-0101, "Fire Watch/Patrol/Inspection," Rev. 00400, All EPIP-EPP-28, "Fire Fighting," Rev. 01500, Sec. 4.0 NIP-FPP-01, Rev. 01700, Sec. 3.0
3.3.1.1 General Fire Prevention Activities. The fire prevention activities shall include but not be limited to the following program elements:	Complies	The fire prevention program is established and implemented as detailed in NDD-FPP. See following elements for additional specific compliance statements and references. Fire prevention activities include, but are not limited to these following elements.	NDD-FPP, Rev. 01400, All

Chapter 3 Reference and Requirements/Guidance	*Compliance Statement*	*Compliance Bases*	*Reference Documents*
3.3.1.1 General Fire Prevention Activities. (1) Training on fire safety information for all employees and contractors including, as a minimum, familiarization with fire prevention procedures, fire reporting, and plant emergency alarms	Complies	Initial General Employee Training (GET) includes the following minimum fire protection program elements as discussed in FAQ 06-0028: ▪ Location and use of fire prevention procedures for Hot Work and transient combustibles ▪ Individual responsibilities regarding fire barriers such as fire dampers, doors, and seals ▪ Actions an individual is required to take upon discovery of a fire ▪ Individual responsibilities regarding control of transient combustibles and disposal of flammable and combustible materials ▪ Examples of types of Hot Work activities requiring a permit ▪ Recognition of and response to a station fire alarm ▪ *Other plant specific fire prevention activities*	NMP General Employee Training (GET) Program NDD-FPP, Rev. 01400, Sec. 4.5 GAP-HSC-01, Rev. 15, Sec. 3.0
3.3.1.1 General Fire Prevention Activities. (2)* Documented plant inspections including provisions for corrective actions for conditions where unanalyzed fire hazards are identified	Complies	Documented plant inspections are required by NIP-FPP-01 and GAP-HSC-01, including provisions for implementation of corrective actions.	NIP-FPP-01, Rev. 01700, Sec. 3.5.5 GAP-HSC-01, Rev. 15, Sec. 3.4-3.5
3.3.1.1 General Fire Prevention Activities. (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	Administrative controls address review of plant modifications and maintenance to ensure that fire hazards and their impact on fire protection systems and features are minimized.	NIP-FPP-01, Rev. 01700, Sec. 2.12 NIP-CON-01, Rev. 02301, All CNG-CM-1.01-1003, "Design Engineering and Configuration Control," Rev. 00401, Attach. 12 CNG-FES-015, Rev. 00004, Forms 16, 17 NEP-FPP-01, Rev. 01500, Sec. 5.5.B

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	Procedures for control of general housekeeping practices and transient combustibles have been established, documented, and implemented as part of the fire protection program. These controls include, but are not limited to the elements described in sub-sections 1 through 6 below. Based on review of the fire prevention programmatic controls described in sub-sections 1 through 6 below, the NFPA 805, Section 3.3.1.2 requirements are satisfied and no other additional elements were evaluated.	NDD-FPP, Rev. 01400, Sec. 3.2.1 GAP-INV-02, Rev. 02400, Sec. 5.2.k, Attach. 4
3.3.1.2 Control of Combustible Materials. (1)* Wood within the power block shall be listed pressure-impregnated or coated with a listed fire-retardant application. *Exception: Cribbing timbers 6 in. x 6 in. or larger shall not be required to be fire-retardant treated.*	Complies	Per GAP-INV-02, unless specifically authorized by the Fire Protection Supervisor, only fire retardant/treated wood products are used within the power block.	GAP-INV-02, Rev. 02400, Sec. 5.2.k, Attach. 4
3.3.1.2 Control of Combustible Materials. (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 with item for implementation	Implementation item: Implement administrative controls to ensure plastic sheeting materials used in the power block are qualified in accordance with NFPA 701 or equivalent. A specific edition / year of NFPA 701 will be cited. See Implementation Item in Attachment S.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.3.1.2 Control of Combustible Materials. (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	Procedural requirements for removal of debris, combustible wastes, etc. at the end of each shift, or more often if conditions warrant are in place per GAP-HSC-01.	GAP-HSC-01, Rev. 15, Sec. 3.1.2
3.3.1.2 Control of Combustible Materials. (4)* Combustible storage or staging areas shall be designated, and limits shall be established on the types and quantities of stored materials.	Complies	Designated combustible storage / staging areas are identified and documented via GAP-INV-02, with appropriate limits on types and quantities of stored materials.	GAP-INV-02, Rev. 02400, Sec. 4.0, Attach. 4, 5
3.3.1.2 Control of Combustible Materials. (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	Flammable liquids are stored in accordance with NFPA 30. Fire suppression and/or detection systems are provided for identified storage areas. Administrative procedures control the use and storage of flammable and combustible liquids outside the bulk storage areas. Procedural controls on flammable and combustible liquids are provided via GAP-INV-02. No other NFPA standards were determined to be applicable based on guidance in NEI 04-02.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.2.4, pg. 10A-24 GAP-INV-02, Rev. 02400, Sec. 4.0, Attach. 4, 5 NEI 04-02, Rev. 2, App. K, Sec. K.1 (FAQ 06-0020, Rev. 1)

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.3.1.2 Control of Combustible Materials. (6)* Controls on use and storage of flammable gases *shall be in accordance with* applicable NFPA standards.	Complies with item for implementation	Bulk gas storage is not permitted within structures housing safety-related equipment. Procedural controls on flammable gases are provided by the CENG Industrial Safety Manual. No other NFPA standards were determined to be applicable based on guidance in NEI 04-02. Implementation Item: See VFDR-05-026 and VFDR-05-027. Modification will relocate Hydrogen Stand-By Supply Bottles. *See modifications in* Attachment S.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.2.2, pg. 10A-23 Fleet Industrial Safety Manual, Chapter 11, "Compressed Gases," Rev. 0, Sec. 6.4.16
3.3.1.3 Control of Ignition Sources	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	
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 Clarification	The requirements invoked by NFPA 51B are satisfied by site procedures governing plant modifications and GAP-FPP-02. **Clarification:** Compliance with NFPA 241 is addressed by compliance with NFPA 51B. Specifically, Section 5.1.1 of NFPA 241, 2000 edition (as referenced by NFPA 805, 2001 edition) states with respect to hot work, "Responsibility for hot work operations and fire prevention precautions, including permits and fire watches, shall be in accordance with NFPA 51B, Standard for Fire Prevention During Welding, Cutting, and Other Hot Work."	GAP-FPP-02, "Control of Hot Work," Rev. 01300, Sec. 4.0 S-SAD-FPP-0101, "Fire Watch/Patrol/Inspection," Rev. 00400, Sec. 4.0

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 with item for implementation	Per station policy, smoking is only allowed in designated areas. Station policies are not controlled procedures. Procedural control of other types of ignition sources is provided via GAP-FPP-02. Implementation Item: *Incorporate station smoking policy into a formal,* controlled process by either revising an existing procedure or creating a new procedure. See Implementation Item in Attachment S.	GAP-FPP-02, Rev. 01300, Sec. 4.0
3.3.1.3.3 Open flames or combustion-generated smoke shall not be permitted for leak or air flow testing.	Complies	Open flame or combustion-generated smoke is not used for leak testing or similar procedures such as air flow determination for leak testing. GAP-FPP-02 implements the above requirement.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.2.3.2, pg. 10A-17 GAP-FPP-02, Rev. 01300, Sec. 4.0
3.3.1.3.4 Plant administrative procedures 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 with item for implementation	This is a new requirement under NFPA 805 for procedural controls on the use of portable electric heaters in the plant and the need to prohibit the use of portable fuel-fired heaters in plant areas containing equipment important to nuclear safety or where there is a potential for radiological release resulting from a fire. Implementation Item: Implement administrative controls for use of portable electric heaters in the plant and to prohibit use of portable fuel-fired heaters in power block structures. See Implementation Item in Attachment S.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 with item for implementation	Existing power block buildings are constructed of noncombustible materials. Additionally, NDD-FPP invokes 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 3, which requires use of noncombustible and fire resistant materials throughout the facility wherever necessary to preclude fire risk. For the post-transition FPP, administrative controls will be implemented to utilize, to the extent practicable, noncombustible construction as defined by NFPA 220-1999 (or a more recent edition of the code) for walls, floors, and components of new power block buildings and changes to existing power block buildings that are required to maintain structural integrity. Implementation item: Implement administrative controls to utilize, to the extent practicable, noncombustible construction as defined by NFPA 220 for walls, floors, and components of new power block buildings and changes to existing power block buildings that are required to maintain structural integrity. A specific edition / year of NFPA 220 will be cited. See Implementation Item in Attachment S.	NDD-FPP, "Fire Protection Program," Rev. 01400, Sec. 3.3.3

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 with item for implementation	Interior wall and structural components, thermal insulation materials, radiation shielding materials, and soundproofing are noncombustible or have been reviewed for overall impacts to the fire protection program. The use of combustible materials is minimized to the greatest extent possible. Interior finish materials have flame spread, smoke and fuel contribution ratings of 25 or less. Any exceptions to these ratings are reviewed for overall impacts to the fire protection program. Additionally, NDD-FPP invokes 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 3, which requires use of noncombustible and fire resistant materials throughout the facility wherever necessary to preclude fire risk. Administrative controls will be implemented to utilize, to the extent practicable, Class A materials for new interior wall or ceiling finish, and Class I materials for new interior floor finish as defined by NFPA 101-2000 (or a more recent edition of the code). Implementation item: Implement administrative controls to utilize, to the extent practicable, Class A materials for new interior wall or ceiling finish, and Class I materials for new interior floor finish as defined by NFPA 101. See Implementation Item in Attachment S.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.1.4, pg. 10A-20 NDD-FPP, "Fire Protection Program," Rev. 01400, Sec. 3.3.3

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.3.4 Insulation Materials. Thermal insulation materials, radiation shielding materials, ventilation duct materials, and soundproofing materials shall be noncombustible or limited combustible.	Complies	Interior wall and structural components, thermal insulation materials, radiation shielding materials, and soundproofing are noncombustible or have been reviewed for overall impacts to the fire protection program. The use of combustible materials is minimized to the greatest extent possible. Interior finish materials have flame spread, smoke and fuel contribution ratings of 25 or less. Any exceptions to these ratings are reviewed for overall impacts to the fire protection program. Additionally, NDD-FPP invokes 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 3, which requires use of noncombustible and fire resistant materials throughout the facility wherever necessary to preclude fire risk.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.1.4, pg. 10A-20 NDD-FPP, "Fire Protection Program," Rev. 01400, Sec. 3.3.3
3.3.5 Electrical.	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 Submit for NRC Approval	NMP1 guidance for design changes in CNG-FES-007 states, "Minimize wiring above suspended ceilings. 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." This guidance was used to perform inspections of areas with suspended ceilings. The suspended ceilings and their supports are of noncombustible construction and storage of materials above the suspended ceilings is prohibited. Wiring above the following suspended ceilings meet the criteria: Control Room (Control Complex El 277') I&C Maintenance Shop (Turbine Building El 277') Low Level Laboratory Area (Turbine Building 261') Wiring above the following suspended ceiling does not meet the criteria: Rad Waste Control Room (Waste Disposal Building 261') There are three (3) cable trays located above the suspended ceiling in the Rad Waste Control room that are not equipped with solid metal top and bottom covers. NRC approval is being requested in Attachment L of the Transition Report, Approval Request 1.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.1.6, pg. 10A-20 CNG-FES-007, Rev. 00010, Attach. 3

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	Noncombustible materials are used in the construction of cable trays. CNG-FES-007 states, "Use metal tray and metal conduits 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." Additionally, as described in NEI 04-02, where used, cable air drops of limited length (~3 feet) are acceptable.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.3.1, pgs. 10A-25, Sec. IX.B. 3.5.1, pg. IX-17 CNG-FES-007, Rev. 00010, Attach. 3 NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10CFR50.48(c)," Rev. 2, App. K, Sec. K.4 (FAQ 06-0021, Rev. 2)

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.3.5.3 Electric cable construction shall comply with a flame propagation test as acceptable to the AHJ. Note 1: Pursuant to 10CFR50.48(c)(2)(v) 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. Note 2: The Exception to this section is not endorsed by 10CFR50.48(c)(2)(v) and has been removed.	Complies by previous NRC approval	The NRC SER dated 7/26/79 states, "Originally installed cable construction does not comply with the requirements of IEEE 383 flame test. Therefore, detection will be provided for all safety-related cable trays and sprinklers will be provided for heavy concentrations of cable trays...We find that, subject to the implementation of the above discussed modifications, the electrical cables satisfy the objectives identified in Section 2.2 of this report and are, therefore, acceptable." Plant modifications for installation of fire detection and suppression capability, commensurate with those described by the NRC SER, were completed for protection from fires involving electrical cables. Plant modifications or other changes have not invalidated the bases for previous NRC approval. The NRC has provided a listing of electric cable flame propagation tests that have been deemed acceptable via FAQ 06-0022. NMP issued FPEE-0-03-001 to identify contemporary cable flame propagation tests for qualification of new electric cables. These flame propagation test standards are consistent with those deemed acceptable via FAQ 06-0022. Finally, CNG-FES-007 states, "For new cable installations, electrical cable construction shall, as a minimum, pass the IEEE 383 flame test acceptance criteria. Exception, Existing cable in place prior to adoption of NFPA 805 shall be permitted to remain as-is."	UFSAR, Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.3.6 pg. 10A-25a NRC Safety Evaluation Report (SER), 7/26/79, Sec. 4.8, pg. 23 FAQ 06-0022, Rev. 3 FPEE-0-03-001, "Cable Insulation Flame Spread Ratings – Acceptability of Newer Test Standards in Lieu of IEEE 383," Rev. 0 CNG-FES-007, Rev. 00010, Attach. 3

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 with Clarification Complies with item for implementation	**Clarification:** NFPA 256 was not an original design requirement for the plant, was not referenced in BTP 9.5-1 and was not a condition in previous NRC Safety Evaluation Reports. Therefore, NMP1 was not evaluated to NFPA 256, but met the equivalent requirement in BTP 9.5-1, which is that metal deck *roof construction should be non-combustible* (see the building materials directory of the Underwriters' Laboratory, Inc.) or listed as Class I by FM System Approval Guide (FM Class I is equivalent to NFPA 256 Class A). The UFSAR states, "Metal deck roof construction is listed as Class I by the FM System Approval Guide." This is consistent with the NRC guidance in position D.1.e of Appendix A to BTP APCSB 9.5-1 and meets the intent of NFPA 805, Section 3.3.6. Implementation item: Incorporate requirement for new metal roof deck construction to be designed and installed so the roofing system will not sustain a self-propagating fire on the underside when exposed to a fire inside the building. Additionally, roof coverings shall be Class A as determined by tests described in NFPA 256. See Implementation Item in Attachment S.	UFSAR, Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.1.5, pg. 10A-20

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 with item for implementation	Administrative procedures have been implemented to prohibit bulk storage of combustible materials inside or adjacent to safety-related buildings or systems during operation or maintenance. Bulk gas storage is not permitted within structures housing safety-related equipment. Contrary to the above statements, bulk compressed hydrogen gas, the Hydrogen Stand-By Supply, is installed on the 261'-0" elevation of the Turbine Building. No prior NRC approval of the Hydrogen Stand-By Supply location was identified. Therefore, a modification to relocate the Hydrogen Stand-By Supply will be done. Implementation Item: See VFDR-05-026 and VFDR-05-027. Modification will relocate Hydrogen Stand-By Supply Bottles. See modifications in Attachment S.	UFSAR, Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.2.2, 2.4.2.2, pgs. 10A-16, 10A-24
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 Complies with item for implementation	Bulk gas storage is not permitted within structures housing safety-related equipment. Bulk Hydrogen and nitrogen storage tanks are located outside. The hydrogen storage configurations that expose power block structures are the hydrogen tube rack located north of the Turbine Building and west of the Reactor Building, and the Hydrogen Stand-By Supply located on the 261'-0" elevation of the Turbine Building. These hydrogen storage configurations meet the applicable requirements of NFPA 50A. Implementation Item: See VFDR-05-026 and VFDR-05-027. Modification will relocate Hydrogen Stand-By Supply Bottles. See modification in Attachment S.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.2.2, pg. 10A-24 NFPA 50A-1999, "Standard for Gaseous Hydrogen Storage at Consumer Sites" FPEE 1-90-019, "Hydrogen Storage Tank Explosion Potential," Rev. 0, All

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 use of EEEE	Bulk hydrogen and nitrogen storage tanks are located outside with their long axes parallel to the Turbine Building. However, the hydrogen and nitrogen storage tanks are perpendicular to the west wall of the Reactor Building. An analysis of the path of travel of these vessels following a rupture has indicated that the wall will withstand the impact without failure. The hydrogen storage tank is also protected by an automatically-initiated water spray system. The acceptability of the hydrogen storage tank orientation has been demonstrated via FPEE 1-90-019 and calculation M31.1-RX261-CW01. The NRC SER dated 7/26/79 states, "Automatic water spray systems are provided to protect the transformers and the hydrogen rack. The installed automatic suppression systems should be adequate to control or extinguish a fire in the protected areas. Protection for the hydrogen rack is designed to prevent an exposure fire at the rack from causing over pressurization of the hydrogen storage cylinders." There have been no plant modifications or other changes to the hydrogen gas storage configuration.	UFSAR Sec. X.N, , Appendix 10A, Rev. 21, Sec. 2.4.2.2, 3.11, pgs. 10A-24, 10A-83 – 10A-84 FPEE 1-90-019, "Hydrogen Storage Tank Explosion Potential," Rev. 0, All NMPC Calculation M31.1-RX261-CW01, "Fire Protection; H2 & N2 Tanks," Rev. 0, All NRC Safety Evaluation Report (SER), 7/26/79, Sec. 5.9, pg. 67
3.3.7.3 Flammable gas storage cylinders not required for normal operation shall be isolated from the system.	Complies	The Fleet Industrial Safety Manual states, "Regulators shall be removed and valve protection caps installed when leaving 'in use' cylinders unattended at the end of a shift or for an extended length of time. Note: This requirement does not apply to cylinders connected to permanent plant systems or when in use, or connected for use, on bottles bracket-mounted to the wall."	Fleet Industrial Safety Manual, Chapter 11, "Compressed Gases," Rev. 0, Sec. 6.4.16

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 with use of EEEE	Bulk storage of flammable and combustible liquids conforms to the applicable requirements of NFPA 30 – 2000 and meets the intent of the NFPA 805, Section 3.3.8 requirement as documented in EIR 51-9156616.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.2.2, 2.4.2.4, pgs. 10A-16, 10A-24 EIR 51-9156616-001, "Nine Mile Point Unit 1 Code Compliance Evaluation for NFPA 30 Flammable and Combustible Liquids Code, 2000 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 with Clarification	**Clarification:** Periodic inspection of the site oil spill retention basin in accordance with S-ENVSP-10 meets the intent of the NFPA 805, Section 3.3.9 requirement.	S-ENVSP-10, "Oil Spill Prevention Control and Countermeasure Plan," Rev. 00900, Sec. 3.7
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 with item for implementation	Implementation Item: Incorporate need for plant tours and normal housekeeping activities to inspect for lubricating oil coming in contact with hot pipes and surfaces, including insulated pipes and surfaces into appropriate plant procedures. See Implementation Item in Attachment S.	
3.3.11 Electrical Equipment. Adequate clearance, free of combustible material, shall be maintained around energized electrical equipment.	Complies	The provisions in GAP-INV-02 and GAP-HSC-01 control material near electrical equipment.	GAP-INV-02, Rev. 02400, Sec. 5.2 GAP-HSC-01, Rev. 15, Attach. 3

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	Reactor Coolant Pump Lube Oil Collection Systems are not required at NMP1 since primary containment is inerted during at power operations.	
3.3.12 Reactor Coolant Pumps. (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	Reactor Coolant Pump Lube Oil Collection Systems are not required at NMP1 since primary containment is inerted during at power operations.	
3.3.12 Reactor Coolant Pumps. (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	Reactor Coolant Pump Lube Oil Collection Systems are not required at NMP1 since primary containment is inerted during at power operations.	
3.3.12 Reactor Coolant Pumps. (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	Reactor Coolant Pump Lube Oil Collection Systems are not required at NMP1 since primary containment is inerted during at power operations.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.3.12 Reactor Coolant Pumps. (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	Reactor Coolant Pump Lube Oil Collection Systems are not required at NMP1 since primary containment is inerted during at power operations.	
3.3.12 Reactor Coolant Pumps. (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	Reactor Coolant Pump Lube Oil Collection Systems are not required at NMP1 since primary containment is inerted during at power operations.	
3.4 Industrial Fire Brigade	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	
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 sub-sections for specific compliance statements and references.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.4.1 On-Site Fire-Fighting Capability. (a) 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: (1) NFPA 600, Standard on Industrial Fire Brigades (interior structural fire-fighting) (2) NFPA 1500, Standard on Fire Department Occupational Safety and Health Program (3) NFPA 1582, Standard on Medical Requirements for Fire Fighters and Information for Fire Department Physicians	Complies with use of EEEE	At all times, a Fire Brigade of five members shall be maintained on the Nine Mile Point site. The Fire Brigade is organized, trained, and equipped to address fire emergencies at the plant. The onsite Fire Brigade is appropriately staffed, trained, and equipped. Compliance of Fire Brigade operations with applicable NFPA 600 requirements has been assessed and documented by EIR 51-9155774. NFPA 1500 and 1582 are not applicable to NMP1 per Section K.6 of NEI 04-02 which states, "The NFPA standards divide fire brigades into two types, based on organization and duties: "Industrial fire Brigades" and "Industrial Fire Departments." Practically this means that a fire fighting organization at a nuclear power plant must comply with either NFPA 600 (for an Industrial Fire Brigade) or both NFPA 1500 and NFPA 1582 (for an Industrial Fire Department)."	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.1.1.2.1, 2.2.5 pgs. 10A-13, 10A-17 EIR 51-9155774-001, "Nine Mile Point Unit 1 Code Compliance Evaluation for NFPA 600, Standard on Industrial Fire Brigades, 2000 Edition," Attach. B, Sec. 2-1.2.1
3.4.1 On-Site Fire-Fighting Capability. (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	NIP-FPP-01 states, "The Fire Brigade shall not include the Station Supervisor, other members of the minimum shift crew necessary for safe shutdown of the Unit, or any other personnel required for other essential functions during a fire emergency."	NIP-FPP-01, Rev. 01700, Sec. 3.2

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.4.1 On-Site Fire-Fighting Capability. (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 to (c): Sufficient training and knowledge shall be permitted to be provided by an operations advisor dedicated to industrial fire brigade support.	Complies	Fire Brigade members are trained in accordance with approved training procedures to familiarize individuals with fire protection systems & equipment, plant fire hazards and emergency response. This training program also ensures that the Brigade leader and at least two members have sufficient training and knowledge of plant safety-related systems to understand the effects of fire and fire suppression on safe shutdown capability.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.2.6, pgs. 10A-18
3.4.1 On-Site Fire-Fighting Capability. (d)* The industrial fire brigade shall be notified immediately upon verification of a fire.	Complies	The Fire Brigade is dispatched by the Control Room upon confirmation via the first responder or multiple indications are received.	EPIP-EPP-28, Rev. 01500, Sec. 5.0
3.4.1 On-Site Fire-Fighting Capability. (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 fire-fighting operations. The physical examination shall determine the ability of each member to use respiratory protection equipment.	Complies	Per CNG-MD-1.01-3000, all Fire Brigade personnel receive an annual physical examination to determine their ability to perform strenuous activities. In addition, this physical examination also determines the ability of each member to use respiratory equipment.	CNG-MD-1.01-3000, "Scheduling, Processing, Performing and Transmitting Results of Physical Examinations for Employees and Contractors," Rev. 00100, All

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.4.2 Pre-Fire Plans. Current and detailed pre-fire plans shall be available to the fire brigade for all areas in which a fire could jeopardize the ability to meet the performance criteria described in Section 1.5.	Complies	Copies of the pre-fire plans are maintained in the Unit 1 Control Room, Site Fire Protection Office, Unit 1 Simulator, 261' Turbine Building Fire Cabinets, Fire Protection Supervisor's Office, and the Technical Support Center (TSC).	S-SAD-FPP-0106, "Preparation and Control of Pre-Fire Plans," Rev. 05, Sec. 3.3 N1-PFP-0101, "U1 Pre-fire Plans"
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	The pre-fire plans contain the following information for each Fire Zone: Sketches depicting the specific area covered and relative location of the specific area covered, General Area Description, Primary and Secondary Access, Specials Hazards, Major Equipment, Automatic Suppression, Fire Detection, Ventilation, and Special Notes.	S-SAD-FPP-0106, Rev. 05, Sec. 3.2 N1-PFP-0101, "U1 Pre-fire Plans"
3.4.2.2 Pre-fire plans shall be reviewed and updated as necessary.	Complies	S-SAD-FPP-0106 specifies circumstances requiring update of the pre-fire plans and instructions. Additionally, the SEP requires annual review and recertification of the Emergency Plan and Emergency Plan Implementing Procedures.	S-SAD-FPP-0106, Rev. 05, Sec. 3.3 SEP, "Site Emergency Plan," Rev. 57, Sec. 8.2.1 N1-PFP-0101, "U1 Pre-fire Plans"
3.4.2.3 Pre-fire plans shall be available in the control room and made available to the plant industrial fire brigade.	Complies	Copies of the pre-fire plans are maintained in the Unit 1 Control Room, Site Fire Protection Office, Unit 1 Simulator, 261' Turbine Building Fire Cabinets, Fire Protection Supervisor's Office, and the Technical Support Center (TSC).	S-SAD-FPP-0106, Rev. 05, Sec. 3.3 N1-PFP-0101, "U1 Pre-fire Plans"
3.4.2.4 Pre-fire plans shall address coordination with other plant groups during fire emergencies.	Complies	EPIP-EPP-28 provides instructions for Fire Brigade coordination with: a) Control Room b) Security c) Radiation Protection d) Offsite Assistance	EPIP-EPP-28, Rev.01500, Sec. 5.0, Attachments 1-4 N1-PFP-0101, "U1 Pre-fire Plans"

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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.	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	
3.4.3 Training and Drills. (a) Plant Industrial Fire Brigade Training. All of the following requirements shall apply. (1) 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	The plant fire brigade members receive training consistent with applicable NFPA 600 requirements.	EIR 51-9155774-001, All
3.4.3 Training and Drills. (a)(2) Industrial fire brigade members shall be given quarterly training and practice in firefighting, 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 with item for implementation	NMP-TR-1.01-1.07 requires quarterly drills for each fire brigade shift such that each fire brigade member participates in at least two drills per year. Additionally, all fire brigade members attend quarterly fire brigade meetings to review FPP changes (e.g., equipment status, new equipment, personnel changes, recent industry operating experience, etc.). However, radioactivity and health physics considerations are not explicitly identified as standing topics for quarterly meetings. Implementation Item: Revise fire brigade training program to include radioactivity and health physics considerations in quarterly fire brigade meetings. See Implementation Item in Attachment S.	NMP-TR-1.01-107, "Nuclear Fire Brigade Training Program," Rev. 00900, Sec. 5.0

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.4.3 Training and Drills. (a)(3) A written program shall detail the industrial fire brigade training program.	Complies	NMP-TR-1.01-107 documents the fire brigade training program.	NMP-TR-1.01-107, Rev. 00900, All.
3.4.3 Training and Drills. (a)(4) Written records that include but are not limited to initial 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	NMP-TR-1.01-107 specifies the requirements for fire brigade training record maintenance by Records Management for the Permanent Plant File.	NMP-TR-1.01-107, Rev. 00900, Sec. 7.0
3.4.3 Training and Drills. (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	EPMP-EPP-11 and NTP-TQS-202 provide training requirements for non-industrial fire brigade personnel who respond with the brigade, including RP, security, and offsite personnel. The SEP also specifies training and drill requirements for maintaining organizational preparedness.	EPMP-EPP-11, "Emergency Preparedness Training Program," Rev. 00600, Sec. 5.0 SEP, "Site Emergency Plan," Rev. 57, Sec. 8.1
3.4.3 Training and Drills. (c) Drills. All of the following requirements shall apply. (1) Drills shall be conducted quarterly for each shift to test the response capability of the industrial fire brigade.	Complies	NMP-TR-1.01-1.07 requires quarterly drills for each fire brigade shift.	NMP-TR-1.01-107, Rev. 00900, Sec. 5.0

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.4.3 Training and Drills. (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	The SEP requires conduct of drills to test the state of emergency preparedness of all participating personnel, organizations, and agencies. Each exercise is conducted to: ensure that the participants are familiar with their respective duties and responsibilities, verify the adequacy of the NMPNS Emergency Plan, Corporate Emergency Response/Recovery Plan, and the methods used in the appropriate Implementing Procedures, test communication networks and systems, check the availability of emergency supplies and equipment, verify the operability of emergency equipment, and/or verify the adequacy of interrelationships with off-site agency plans. NMP-TR-1.01-107 requires conduct of drills to evaluate the fire brigade's ability to react, respond, and demonstrate proper fire-fighting techniques to control and extinguish the fire and smoke conditions being simulated.	NMP-TR-1.01-107, Rev. 00900, Attach. 1 SEP, "Site Emergency Plan," Rev. 57, Sec. 8.1.2
3.4.3 Training and Drills. (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	NMP-TR-1.01-107 requires that drill locations are not repeated for a period of one year and has provisions for conduct of drills in non-safety related plant areas, safety-related plant areas, and inside the RCA.	NMP-TR-1.01-107, Rev. 00900, Sec. 5.2, Attach. 1
3.4.3 Training and Drills. (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	Drill records required by NMP-TR-1.01-107 detail the drill scenario, fire brigade member response, and ability of the brigade to perform as a team.	NMP-TR-1.01-107, Rev. 00900, Sec. 7.0, Attach. 1

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.4.3 Training and Drills. (c)(5) A critique shall be held and documented after each drill.	Complies	NMP-TR-1.01-107 and the SEP require conduct and documentation of a critique following each drill.	NMP-TR-1.01-107, Rev. 00900, Sec. 5.2, Attach. 1 SEP, "Site Emergency Plan," Rev. 57, Sec. 8.1.2
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 with Clarification	The Fire Brigade is equipped to address fire emergencies at the plant. A variety of protective clothing and equipment, breathing equipment, salvage covers, and/or forcible entry and rescue tools are provided in various locations on site in order to effectively respond to expected emergencies. Items specified as requiring a listing by an industry organization such as Underwriters' Laboratories (UL) or Factory Mutual (FM), will be procured with the required listing mark. **Clarification:** Conformance with applicable NFPA standards is limited to the specification and procurement of fire-fighting equipment, and those standards in effect at time of purchase of that equipment. Care and maintenance is determined based on equipment condition and performance.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.2.5, 2.3.2, pgs. 10A-17, 10A-18b
3.4.5 Off-Site Fire Department Interface.	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	
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	The SEP states, "NMPNS provides for training to local offsite support organizations as specified in respective letters of agreement and as required to ensure a high state of emergency preparedness and response capability of these organizations. The local organizations that may provide onsite emergency assistance are encouraged to become familiar with the NMPNS (including the physical plant layout and key station personnel), and are invited to attend emergency preparedness training conducted by NMPNS. Such training is provided annually to the appropriate organizations and individuals."	SEP, "Site Emergency Plan," Rev. 57, Sec. 8.1.1, Appendix A

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	The SEP states, "NMPNS provides for training to local offsite support organizations as specified in respective letters of agreement and as required to ensure a high state of emergency preparedness and response capability of these organizations. The local organizations that may provide onsite emergency assistance are encouraged to become familiar with the NMPNS (including the physical plant layout and key station personnel), and are invited to attend emergency preparedness training conducted by NMPNS. Such training is provided annually to the appropriate organizations and individuals."	SEP, "Site Emergency Plan," Rev. 57, Sec. 8.1.1, Appendix A
3.4.5.3 Security and Radiation Protection. Plant security and radiation protection plans shall address off-site fire authority response.	Complies	The Site Emergency Plan provides for interface between NMP Security and RP for training and coordination of offsite fire response.	SEP, "Site Emergency Plan," Rev. 57, Sec. 8.1.1
3.4.6 Communications. An effective emergency communications capability shall be provided for the industrial fire brigade.	Complies	Effective emergency communications capabilities are provided.	SEP, "Site Emergency Plan," Rev. 57, Sec. 7.2
3.5 Water Supply.	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.5.1 A fire protection water supply of adequate reliability, quantity, and duration shall be provided by one of the two following methods. (a) Provide a fire protection water supply of not less than two separate 300,000-gal supplies. (b) Calculate the fire flow rate for 2 hours. This fire flow rate shall be based on 500 gpm 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 by previous NRC approval	NMP1 complies with method (a) via previous approval as described below. The UFSAR states, "The fire pumps have separate water intakes, but are located in the same supply water sump which also feeds the service water system. This sump is part of the Ultimate Heat Sink (UHS) intake from Lake Ontario. Sufficient water is available for both systems, and a failure of the fire protection system does not affect the service water system." The cited SER states, "The fire protection water supply for the plant is provided by Lake Ontario, which also serves as the ultimate heat sink. We find that the water supply satisfies the objectives identified in Section 2.2 of this report and is, therefore, acceptable." The technical basis for approval still applies, remains valid, and is consistent with the plant configuration.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.2.6, pg. 10A-42 NRC Safety Evaluation Report (SER), 7/26/79, Sec. 4.3.1.1, pg. 12

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	NMP1 complies with Exception No. 1. The fire pumps have separate water intakes, but are located in the same supply water sump which also feeds the service water system. This sump is part of the UHS intake from Lake Ontario. Sufficient water is available for both systems, and a failure of the fire protection system does not affect the service water system.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.2.6, pg. 10A-42

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	The design and installation of the fire pumps is in accordance with the requirements of NFPA 20, as documented in the reference documents.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Table 1.2.2, Sec. 2.5.2.3, pgs. 10A-86 through 87a, 10A-39 EIR 51-9077284-000, "NMP-1 Code Compliance Reviews," Rev. 0, Sec. 4.0, Appendix H FPEE 0-06-002, "Evaluation of Diesel Fire Pump Testing Frequencies," Rev. 0 Calculation S13.1-100F002, Rev. 2, "Fire Protection Water Supply," Attach. H
3.5.4 At least one diesel engine-driven fire pump or two more *seismic Category I Class IE* electric motor-driven fire pumps connected to redundant Class IE emergency power buses capable of providing 100 percent of the required flow rate and pressure shall be provided.	Complies	Unit 1 has two vertical shaft, turbine-type, centrifugal fire pumps, one electric motor-driven and one diesel engine-driven. Each pump is independently capable of meeting the maximum fire demand flow. The required flow rate and pressure is established by Calculation S13.1-100F002. The capability for supplying additional concurrent non-fire protection demands for emergency back-up applications exists via cross-connection to the separate and independent Unit 2 fire protection water supply and distribution system.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.2.3, pg. 10A-39 Calculation S13.1-100F002, Rev. 2, Attach. H
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	The diesel driven fire pump (including the pump controller, air tanks, and fuel day tank) is separated from the electric fire pump by walls with 3-hr rated barriers, ceilings with 2-hr rated barrier, and suppression.	N1-SD-018, "Fire Suppression Systems – System Description," Rev. 05, Sec. 2.1 Fire Rated Walls and Slabs Drawing Series: B-40143-C, SH 1, Rev. 10 B-40144-C, SH 1, Rev. 9

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.5.6 Fire pumps shall be provided with automatic start and manual stop only.	Complies	Per N1-SD-018, the diesel fire pump will start automatically on: 1) low fire main pressure, or 2) low starting air pressure. The diesel fire pump can also be remotely started from the Control Room and locally in the diesel fire pump room. The diesel fire pump can only be stopped locally by placing the manual switch in the STOP position. The electric pump will start automatically on low fire main pressure. The electric pump can also be remotely started from the Control Room and locally at the pump location. The electric fire pump can only be stopped locally by placing the manual switch in the STOP position.	N1-SD-018, Rev. 05, Sec. 2.1
3.5.7 Individual fire pump connections to the yard fire main loop shall be provided and separated with sectionalizing valves between connections.	Complies	Per N1-SD-018, both fire pumps are connected to a single fire header. The header is divided in the Screenhouse Building into two separate lines to form a closed loop extending throughout the unit supplying all fire protection systems. The header is connected to the 12-in. yard loop which in turn is connected to the NMP2 yard loop through two separate connections, one with a motor-operated valve and one with a post indicator valve. Exterior post indicator valves and interior OS&Y valves subdivide the loop so that a single section can be isolated without impairing the entire system.	N1-SD-018, Rev. 05, Sec. 2.1
3.5.8 A method of automatic pressure maintenance of the fire protection water system shall be provided independent of the fire pumps.	Complies	Per N1-SD-018, the fire main pressure is automatically maintained at approximately 124 psig by two automatic fire header jockey pumps.	N1-SD-018, Rev. 05, Sec. 2.1

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	N1-SD-018 describes the alarms that are transmitted to the Control Room for status of the fire pumps. Pump running, fire pump trouble, and low starting air - fuel oil alarm indication is provided for the diesel fire pump. Pump running, fire pump trip overload, and fire header low pressure alarm indication is provided for the electric fire pump. The fire detection system transmits alarm and supervisory signals concerning operation and impairment of both pumps to the Control Room.	N1-SD-018, Rev. 05, Sec. 2.1
3.5.10 An underground yard fire main loop, designed and installed in accordance with NFPA 24, Standard for the Installation of Private Fire Mains and Their Appurtenances, shall be installed to furnish anticipated water requirements.	Complies with use of EEEE	Compliance of the underground yard fire main loop with the applicable NFPA 24-1973 requirements has been assessed and documented in EIR 51-9077284. No code differences of significance were identified.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.2.1 – 2.5.2.2, pg. 10A-39 EIR 51-9077284-000, "NMP-1 Code Reviews," Sec. 4.0, Appendix I
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	Sectional control valves are provided in the piping system to facilitate any maintenance or repair work without isolating the entire underground piping network. Each automatic sprinkler and manual hose station standpipe have independent connections to the water supply system for safety-related structures. Sprinkler systems and manual hose station standpipes are connected to the building/underground supply main and arranged so that a single failure will not impair both the automatic fire protection system and a manual means to provide backup protection. Building supply mains make multiple connections to the underground water supply system to minimize service interruptions during a single-failure event.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.2.1 & 2.5.3.1, pgs. 10A-39, 10A-44

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	Couplings and equipment are compatible with local fire department thread design.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.2.7, pg. 10A-43
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	Standpipe risers and sprinkler system supply headers are equipped with manually-operable supply isolation valves (e.g., OS&Y). NMP1 does not have seismic standpipes; therefore, ANSI B31.1, Code for Power Piping, criteria does not apply.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.2.2 and 2.5.3.1, pgs. 10A-39, 10A-44

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	Water supply system valves up to sprinkler system control valves and hydrant isolation valves are supervised in the correct position through the use of one or more of the following methods: electric supervision, periodic valve position verification, chained and locked, tamper seals. Additionally, N1-FST-FPE-A001 implements an annual fire system (water) valve check program.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.3.2, pg. 10A-44 N1-FST-FPE-A001, "Fire Protection Equipment Annual Inspection," Rev. 00300, Sec. 7.2

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.5.15 Hydrants shall be installed approximately every 250 ft 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 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 Complies with use of EEEE	Yard fire hydrants are installed approximately every 250 ft on the yard main system. Each hydrant is provided with a curb box-operated hydrant isolation valve. Sufficient equipment is provided to establish an effective hose stream. The fire hydrant hose houses have been retired from service. Fire hose, combination nozzles, and other auxiliary equipment specified in NFPA 24 are stored in two central fixed locations inside the plant, one on the west side and one the east side. FPEE-0-03-005 assessed and documented the acceptability of this change.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.2.7, pg. 10A-43 FPEE-0-03-005, "Removal of Fire Hydrant Hose Houses and Consolidation of Hose House Equipment," Rev. 0, All

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 Clarification based on Exception No. 1	In addition to fire protection use, N1-SD-018 and the UFSAR describe the following non-fire protection, nuclear safety / emergency uses of the fire protection water supply system: a. Provide a source of make-up water to the emergency condenser make-up tanks. b. Provide an emergency source of water for containment and reactor vessel flooding. c. Provide an emergency source of make-up water to the spent fuel pool. d. Provide a back-up water source for the emergency service water system. e. Provide a back-up water source for the diesel generator cooling water system. The following additional considerations apply to non-fire protection uses described above: a. Use of the electric motor-driven or diesel engine-driven fire pump as a source of emergency make-up to the emergency condenser make-up tanks would not be required for a minimum of 48-hours after depletion of the emergency condenser and CST inventories. Therefore, concurrent fire protection use is unlikely. b. Use of the electric motor-driven or diesel engine-driven fire pump for emergency containment and reactor vessel flooding would only be required in the unlikely event that all other means of core injection are lost. On this basis, concurrent fire protection use is unlikely.	N1-SD-018, Rev. 05, Sec. 1.0 UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5, pg. 10A-29 UFSAR Sec. X.N, Appendix 10B, Rev. 21, Appendix D, Sec. 1.c, pg. 10B-220 Calculation S13.1-100F005, "Diesel Fire Pump / Reactor Vessel Flooding," Rev. 0, All Calculation S13.1-100F006," Pressure Drop Calculation, NMP2 Main Fire Pumps Supply to NMP1 Fire Water Distribution System," Rev. 0, All Calculation 13.1-100F007, "Hydraulic Analysis of Diesel Fire Pump Supply to ESW #11 and Emergency Diesel Cooling Water Systems," Rev. 0, All

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.5.16 (cont.)		**(cont.)** c. Use of the electric motor-driven or diesel engine-driven fire pump as a source of emergency spent fuel pool inventory would only be required upon loss of the Fuel Pool Make-up System, condensate transfer via hoses, and demineralized water with hoses from refueling service connections. On this basis, concurrent fire protection use is unlikely. d. and e. Use of the diesel engine-driven fire pump as a source of ESW and EDG cooling water would only be required upon occurrence of a fire in the Screenhouse that could disable all other Screenhouse pumps. **Clarification:** The required flow rate and pressure demand for each of the non-fire protection, nuclear safety / emergency uses of the fire protection water supply system described above can be supplied by use of one (1) of the redundant fire pumps. Therefore, in the unlikely event that concurrent fire protection and non-fire protection use is required, either the remaining Unit 1 fire pump, or the Unit 2 fire protection water supply system will be available. To ensure adequate water supply for fire suppression activities concurrent with other uses, a modification for a cross-connection will be installed. See modifications in Attachment S. Additionally, since the source of the Unit 1 and Unit 2 fire protection water supply is Lake Ontario, the water available to supply fire protection and/or non-fire protection demands is not a concern.	
3.6 Standpipe and Hose Stations.	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	Standpipe and hose systems are provided for all power block structures. Per N1-SD-018, standpipe risers are located at various points throughout Nine Mile Point Unit 1 to serve hose stations. The standpipes/hose stations are so spaced as to permit hose stream coverage of all points in the buildings including primary containment. Hose stations are equipped with 100 feet of 1-1/2 inch hose with adjustable spray nozzles. Hose stream coverage is in accordance with NFPA 14 Class III systems.	N1-SD-018, Rev. 05, Sec. 2.2.5 UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.3.4, Table 1.2.2, pgs. 10A-46, 10A-47, 10A-86 FPEE 0-98-003, "Acceptable Use of Aluminum Fire Hose Couplings," Rev. 0 EIR 51-9077284-000, "NMP-1 Code Reviews," Sec. 4.0, Appendix F
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	Based on the fire protection water supply design information contained in S13.1-100F002, the NMP1 fire protection water supply system is capable of providing adequate flow and pressure for all hose stations and exceeds NFPA 14-1963 design requirements. Pressure reduction devices are not installed for hose stations. This has been deemed acceptable because fire hoses connected to the standpipe system are intended for use exclusively by trained fire brigade personnel.	S13.1-100F002, "Fire Protection Water Supply," Rev. 02 UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.3.4, Table 1.2.2, pgs. 10A-46, 10A-47, 10A-86

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	Hose nozzles are suitable for the type of hazards listed in the fire hazards analysis for each area. Nozzles are equipped with shutoff handles and adjustable fog nozzles which can be varied down to a 10-degree minimum spray pattern to render them safe for use on energized electrical equipment.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.3.5, pg. 10A-50

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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). Note: The Exception to this section is not endorsed by 10CFR50.48(c)(2)(vi) and has been removed. Licensees who wish to use the exception to Section 3.6.4 must submit a request for a license amendment in accordance with 10CFR50.48(c)(2)(vii).	Complies by previous NRC approval	The UFSAR states, "The Unit 1 standpipe design does not contemplate simultaneous earthquake and fire conditions; therefore, the requirement for seismically-qualified standpipes is not incorporated into the design. Further justification is that Unit 1 is not in an area of high seismic activity." NMP1 was an operating plant when Appendix A to BTP APCSB 9.5-1 was issued in 1976. Regulatory Position E.3.d relative to the capability to supply water to at least standpipes following a SSE was not applicable to plants under construction or in operation as of July 1, 1976. The standpipes and hose stations were found acceptable by the NRC as documented in the Safety Evaluation Report (SER) dated July 26, 1979. Section 4.3.1.3 of the SER states: "All yard fire hydrants, automatic and manual water suppression systems, and interior hose stations are supplied by the loop main. Exterior post indicator valves and interior OSY valves subdivide the loop into a number of sections so that a single section could be isolated without impairing the entire system. However, there are locations where the isolation of a single section could impair the availability of both automatic sprinklers and the backup hose stations in areas containing or exposing safety-related equipment. The licensee has proposed to modify a portion of the system to preclude the loss of both automatic suppression systems and interior hose stations in areas so protected. We will require the license to submit the details of such modifications prior to implementation."	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.3.4, pg. 10A-47 NRC Safety Evaluation Report (SER), 7/26/79, Sec. 4.3.1.3 and 4.3.1.4, pgs. 15-16

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.6.4 (cont.)		(cont.) "We find that, subject to the implementation of the above described modifications, the fire water piping system satisfies the objectives identified in Section 2.2 of this report and is, therefore, acceptable." *In addition, Section 4.3.1.4 of the SER states:* "Interior hose stations equipped with 100 feet of 1-1/2 inch diameter woven jacket-lined hose are provided in all areas of the plant except primary containment. The licensee will provide 1 inch diameter hose stations inside primary containment. "Electrically safe" fire hose nozzles have been provided in areas containing electrical equipment. The licensee will perform a hose stretch test to assure that all points in safety-related areas, and in other plant areas which contain major fire hazards, can be reached effectively by at least one hose stream with a maximum hose length of 100 feet. Seven additional hose stations will be provided. The licensee has verified that 100 gpm at a minimum residual pressure of 65 psig is available at every hose stations outlet in the plant. We find that, subject to implementation of the above described modifications, the interior fire hose stations satisfy the objectives identified in Section 2.2 of this report and are, therefore, acceptable." [Note: Section 2.2 of the SER references Appendix A to BTP APCSB 9.5-1 as the applicable guidance used by the NRC in their review.] The technical basis for approval still applies, remains valid, and is consistent with the plant configuration.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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.	N/A	NMP1 has no seismic required hose stations. See Compliance Bases for 3.6.4 above.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.3.4, pg. 10A-47
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 use of EEEE	Portable fire extinguishers are in accordance with the requirements of NFPA 10-1975, as documented in cited reference documents.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Table 1.2.2, pg. 10A-85 EIR 51-9077284-000, "NMP-1 Code Reviews," Sec. 4.0, Appendix A
3.8 Fire Alarm and Detection Systems.	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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: (1) Actuation of any fire detection device (2) Actuation of any fixed fire suppression system (3) Actuation of any manual fire alarm station (4) Starting of any fire pump (5) Actuation of any fire protection supervisory device (6) Indication of alarm system trouble condition	Parent Element: Complies with use of EEEE Sub-Elements (1) & (2): Complies Sub-Element (3): N/A Sub-Element (4): Complies Sub-Element (5): Complies Sub-Element (6): Complies	Evaluation of the NMP1 fire detection and alarm system with the applicable NFPA 72A-1964, Local Protective Signaling Systems requirements was assessed and documented in EIR 51-9077284. There were no code differences of significance in addition to these identified by the UFSAR. 1) & 2) Fire Detection/Suppression alarms are provided to the fire control panels in the Control Room. 3) Manual pull fire alarm boxes are not used in the station. 4) Fire Pump Supervision is provided. 5) Fire protection supervisory devices are provided as required. 6) Silencing switches provided and lamps are also provided such that the trouble signal can be silenced and the lamp will remain illuminated indicating the trouble condition until it clears	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Table 1.2.2, pg. 10A-88 N1-SD-017, "Fire Detection System – System Description," Rev. 3, Sec. 2.0 EIR 51-9077284-000, "NMP-1 Code Reviews," Sec. 4.0, Appendix K
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	The plant emergency communications consist of a page party/public address system as the primary means to initiate emergency evacuation signals and other emergency alarms throughout the plant. Other fixed and portable communication systems are also available to station personnel to communicate to control room personnel.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.5.3, pgs. 10A-28 – 10A-29

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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: (1) General site population in all occupied areas (2) Members of the industrial fire brigade and other groups supporting fire emergency response (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.	Parent Element: Complies Sub-Element (1): Complies Sub-Element (2): Complies Sub-Element (3): Complies	Per the SEP, the Nine Mile Point communication capabilities include multiple systems and redundancies which ensure the performance of vital functions in transmitting and receiving information throughout the course of an emergency. Multiple modes and paths are available for necessary emergency communications. Systems available at the various emergency facility locations or available for use by response organizations are: Telephone Systems, NRC Emergency Notification System, Radiological Emergency Communications System, Other Dedicated Telephone Line Systems, Public Address and Page System, Radio Systems, and Emergency Response Data System. Sub-Element (1): The Public Address and Paging System will be used at a minimum to notify the General site population. Sub-Element (2): At a minimum, the Telephone Systems, Public Address and Page System, and Radio Systems are available to notify members of the industrial fire brigade and other groups supporting fire emergency response. Sub-Element (3): At a minimum, the Telephone System, Radiological Emergency Communications System, and radio systems are available for communication in the event of a fire requiring an off-site fire emergency response agency.	SEP, "Site Emergency Plan," Rev. 57, Sec. 7.2

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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	The fire detection devices are installed in accordance with the requirements of NFPA 72E-1974 and as described in the UFSAR and referenced EEEEs.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.1, Table 1.2.2, pgs. 10A-30 –10A-31, 10A-87a – 10A-88 EIR 51-9077284-000, "NMP-1 Code Reviews", Sec. 4.0, Appendix K FPEE 1-97-002, "Performance-Based Fire Detector Testing, " Rev. 0
3.9 Automatic and Manual Water-Based Fire Suppression Systems.	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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: (1) NFPA 13, Standard for the Installation of Sprinkler Systems (2) NFPA 15, Standard for Water Spray Fixed Systems for Fire Protection (3) NFPA 750, Standard on Water Mist Fire Protection Systems (4) NFPA 16, Standard for the Installation of Foam-Water Sprinkler and Foam-Water Spray Systems	Sub-Elements (1)&(2): Complies with use of EEEE Sub-Elements (3)&(4): N/A	Sprinkler and deluge systems were reviewed against the requirements of NFPA 13 and NFPA 15 with justifications for minor deviations provided in the referenced engineering evaluation. See Table 4-3 for required suppression systems. There are no Water Mist (NFPA 750) or Foam-Water (NFPA 16) systems at NMP1.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Table 1.2.2, pgs. 10A-85 – 10A-86 EIR 51-9077284-000, "NMP-1 Code Reviews," Sec. 4.0, Appendices D, E, and G
3.9.2 Each system shall be equipped with a water flow alarm.	Complies	Automatic sprinkler systems are provided with switches (e.g., pressure, flow) which indicate waterflow through these systems in the form of an alarm at their respective Local Fire Control Panel and the Main Fire Control Panel.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.3.1, pg. 10A-44

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.9.3 All alarms from fire suppression systems shall annunciate in the control room or other suitable constantly attended location.	Complies	Per N1-SD-017, the plant has a protective signaling system which transmits various fire and supervisory signals to the control room and to local fire panels. In addition to signals from smoke, heat, and other detectors located in selected areas of the plant, the system also transmits alarm and supervisory signals concerning operation or impairment of the fire pumps, carbon dioxide system, Halon 1301 systems, deluge and automatic sprinkler systems and closing of some valves in the fire protection water system.	N1-SD-017, "Fire Detection System – System Description," Rev. 3, Sec. 2.0
3.9.4 Diesel-driven fire pumps shall be protected by automatic sprinklers.	Complies	The diesel engine-driven fire pump, including fuel storage tank, controller, and pneumatic starting system are located in a room protected by a sprinkler system of wet-pipe type design. The sprinkler system provided for the diesel engine-driven fire pump room is of wet-pipe type design.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.2.3, pg. 10A-40
3.9.5 Each system shall be equipped with an OS&Y gate valve or other approved shutoff valve.	Complies with use of EEEE	Each water-based fire suppression system is connected to the supply header with an approved control valve.	EIR 51-9077284-000, "NMP-1 Code Reviews," Sec. 4.0, Appendices D, E, and G
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	Water supply system valves up to sprinkler system control valves and hydrant isolation valves are supervised in the correct position through the use of one or more of the following methods: electric supervision, periodic valve position verification, chained and locked, tamper seals.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.3.2, pg. 10A-44
3.10 Gaseous Fire Suppression Systems	N/A	N/A - Section title, no technical requirements. See sub-sections for specific compliance statements and references.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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: (1) NFPA 12, Standard on Carbon Dioxide Extinguishing Systems (2) NFPA 12A, Standard on Halon 1301 Fire Extinguishing Systems (3) NFPA 2001, Standard on Clean Agent Fire Extinguishing Systems	Sub-Element (1): Complies with use of EEEE Sub-Element (2): Complies with use of EEEE Sub-Element (3): N/A	Sub-Element (1): The NMP1 automatic CO_2 systems have been placed in the alarm-only mode due to life safety concerns. All systems are manually operated at the present time, which is an acceptable method of actuation as documented in FPEE 0-03-002. Sub-Element (2): Compliance of the Halon 1301 systems with the applicable requirements of NFPA 12A (1980) was assessed and documented in EIR 51-9077284. No code differences of significance were identified. Sub-Element (3): Clean agent fire extinguishing systems are not used at NMP1.	EIR 51-9077284-000, "NMP-1 Code Reviews," Sec. 4.0, Appendices B, C UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.5.5, pg. 10A-52 FPEE-0-02-001, "CO_2 Fire Suppression Systems at NMP1 and NMP2 for Compliance with Environmental Protection Agency (EPA) Recommendations," Rev. 0 FPEE-0-03-002,"Evaluation of Interim Action taken to Prevent Personnel Injury From CO_2," Rev. 0
3.10.2 Operation of gaseous fire suppression systems shall annunciate and alarm in the control room or other constantly attended location identified.	Complies	The plant has a protective signaling system which transmits various fire and supervisory signals to the control room and to local fire panels. In addition to signals from smoke, heat, and other detectors located in selected areas of the plant, the system also transmits alarm and supervisory signals concerning operation or impairment of the fire pumps, carbon dioxide system, Halon 1301 systems, deluge and automatic sprinkler systems and closing of some valves in the fire protection water system.	N1-SD-017, "Fire Detection System – System Description," Rev. 3, Sec. 2.0

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 with use of EEEE	As described in EIR 51-9077284-000, agent leakage at doors, dampers, etc. typically provides sufficient relief for normal total flooding applications, without need for additional venting. Adequate sealing of spaces protected by Halon 1301 systems is addressed by provision of penetration seals, and HVAC damper closure upon fire detection system activation. This has been demonstrated by system discharge testing that proved gaseous fire suppression systems at NMP1 held agent concentration without adverse effects of over-pressurization. Confinement of potential radioactive contaminants is ensured by response to fires in radiologically controlled plant areas by trained radiation protection (RP) personnel.	EIR 51-9077284-000, "NMP-1 Code Reviews," Sec. 4.0, Appendices B, C N1-SD-020, "Halon 1301 System Description," Rev. 2, Sec. 2.0
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.	Complies	There are no areas where the primary and backup gaseous systems utilize a common supply of gaseous agent or common system piping network. Therefore, a crack in any pipe will not impair both the primary and backup fire suppression capability.	PID Drawings: F-45094-C, "Halon," Rev. 6 C-18039C-003, "Cardox Fire Extinguishing System," Series
3.10.5 Provisions for locally disarming automatic gaseous suppression systems shall be secured and under strict administrative control.	Complies	S-SAD-FPP-0105 contains provisions for locally disarming automatic gaseous suppression systems and requires administrative controls in terms of compensatory measure when such systems are disarmed.	S-SAD-FPP-0105, "Compensatory Measures for Inoperable Fire Protection Systems and Component," Rev. 01800, Sec. 3.3, 5.7-5.8
3.10.6 Total flooding carbon dioxide systems shall not be used in normally occupied areas.	Complies with use of EEEE	The NMP1 automatic CO_2 systems have been placed in the alarm-only mode due to life safety concerns. All systems are manually operated at the present time, which is an acceptable method of actuation as documented in FPEE 0-03-002.	FPEE 0-03-002 Rev. 1

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 with use of EEEE	The NMP1 automatic CO_2 systems have been placed in the alarm-only mode due to life safety concerns. All systems are manually operated at the present time, which is an acceptable method of actuation as documented in FPEE 0-03-002. The carbon dioxide systems are provided with alarms and an odorizer.	FPEE 0-03-002 Rev. 1
3.10.8 Positive mechanical means shall be provided to lock out total flooding carbon dioxide systems during work in the protected space	Complies	Carbon dioxide systems can be locked out through mechanical means.	CNG-MN-4.01-1000, "Integrated Work Planning"
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 with use of EEEE	Thermal shock was considered at the time of design for CO_2. CO_2 discharge testing has been performed. There are no effects caused by thermal shock documented for this testing. NMP1 also performed an analysis that took into account damage to safety related electrical equipment due to thermal shock caused by gaseous suppression system actuations. CO_2 systems have been manually locked out and placed in "alarm only" mode; thereby, eliminating concerns for spurious actuation. Halon does not present a secondary thermal shock risk.	N1-SD-004, "Carbon Dioxide System Description," Rev. 2, Sec. 2.0 Correspondence from J.R. Corcoran to P.A Burt, Acceptance Test of Cardox System, Nine Mile Point," dated 08/22/69 EIR 51-9175332-00A, "NMP-1 NFPA 805 Fire Suppression Effects Analysis" N1-SD-020, "Halon 1301 System Description," Rev. 2, Sec. 2.0 SFPE Handbook of Fire Protection Engineering, Third Edition, Section 4, Chapter 6

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.10.10 Particular attention shall be given to corrosive characteristics of agent decomposition products on safety systems.	Complies	NMP1 has Halon systems. Halon is non-corrosive as documented in cited reference documents and does not produce any corrosive decomposition products.	N1-SD-020, "Halon 1301 System Description," Rev. 2, Sec. 2.0 SFPE Handbook of Fire Protection Engineering, Third Edition, Section 4, Chapter 6
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.	

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 or space that meets the requirements of NFPA 80A, Recommended Practices for Protection of Buildings from Exterior Exposure Fires. 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	NMP1 did not use a performance-based analysis to determine the adequacy of building separation. Separation of major buildings within the power block is accomplished by a combination of barriers having a fire resistance rating of 3 hours, open space of at least 50 ft., or meeting applicable NFPA 80A protection criteria. Where major plant buildings abut one another, the exterior walls are designated and maintained as 3 hour rated barriers. Where major buildings do not abut, exterior walls are not designated as being fire rated even though exterior walls are of substantial, noncombustible construction. NFPA 80A provides guidelines concerning protection of buildings from exterior exposure fires. The code describes methods of determining recommended separation between buildings to mitigate radiant energy impingement during an exposure fire and identifies means of protecting buildings that do not possess the desired spatial separation. The primary means of protecting exposed exterior walls described in NFPA 80A is provision of equivalent 3 hour rated walls and closure of openings with materials having a fire rating equivalent to the walls. The substantial concrete and steel construction of the exterior walls of major plant buildings and the limited door and window openings are consistent with NFPA 80A protection criteria.	Fire Rated Walls and Slabs Drawing Series: B-40141-C, SH 1, Rev. 5 B-40142-C, SH 1, Rev. 9 B-40143-C, SH 1, Rev. 10 B-40144-C, SH 1, Rev. 9 B-40145-C, SH 1, Rev. 6 B-40146-C, SH 1, Rev. 8 B-40147-C, SH 1, Rev. 5 B-40148-C, SH 1, Rev. 6 FPEE 1-85-007, "Waste Building Truck Bay West Wall," Rev. 0 FPEE 1-01-003, "Turbine Building – Diesel Generator Building Separation Above 300 ft.," Rev. 1 FPEE 0-98-002, "Guidelines for Placement of Temporary Structures," Rev. 0 FPEE 1-12-001, "Evaluation of NMP1 Appendix R Exemptions for Transition to NFPA 805," Rev. 0

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 with use of EEEEs	Fire barriers are depicted on station drawings and have been assessed by fire protection engineering evaluations. NMP1 utilizes primarily 2-hr and 3-hr rated fire barriers to separate fire areas or protect safety-related equipment from exposure fire hazards. The rating of these barriers is determined based on the hazard present and the evaluation of significance of equipment in the area. Thermal shield walls are also utilized in specific applications where a hazard cannot be sufficiently bounded by rated construction to be termed as a distinct fire area due to configuration or original plant design. Certain portions of these walls are sealed with configurations which would meet a rated fire barrier to protect important equipment outside the area from a fire inside the area. These walls have been evaluated with respect to the protection required.	UFSAR Sec. X.N, Appendix 10A, Rev. 21, Sec. 2.4.1.10, pgs. 10A-21 Fire Rated Walls and Slabs Drawing Series: B-40141-C, SH 1, Rev. 5 B-40142-C, SH 1, Rev. 9 B-40143-C, SH 1, Rev. 10 B-40144-C, SH 1, Rev. 9 B-40145-C, SH 1, Rev. 6 B-40146-C, SH 1, Rev. 8 B-40147-C, SH 1, Rev. 5 B-40148-C, SH 1, Rev. 6 FPEE 1-89-004, "Thermal Shield Walls," Rev. 0 FPEE 1-98-001, "Assessment of Damaged Pyrocrete on Unit 1 Structural Steel," Rev. 0 FPEE 1-98-002, "Structural Steel Fireproofing," Rev. 0 FPEE 1-12-001, "Evaluation of NMP1 Appendix R Exemptions for Transition to NFPA 805," Rev. 0

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 3.11.4 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: (1) NFPA 80, Standard for Fire Doors and Fire Windows	Sub-Element (1): Complies with use of EEEEs.	The operative requirements of NFPA 805 for closure of door openings in fire barriers is use of doors (including frames and hardware) which have been tested in accordance with NFPA 252, Standard Methods of Fire Tests of Door Assemblies, and have been listed / labeled accordingly for the required fire resistance rating. Fire doors at NMP1 meet the requirement for use of suitably tested and listed / labeled fire door assemblies. The fire resistance ratings of fire doors are consistent with that of the barrier, or have been evaluated and determined to be adequate for the hazards.	B-46000-C Drawing Series, "Door Summary / Modification Sheets," All FPEE 1-91-003, "Excessive Door Gap at Floor for Fire Doors D-84, D-108, D-111, D-245, and D-246," Rev. 1 FPEE1-12-001, "Evaluation of NMP1 Appendix R Exemptions for Transition to NFPA 805," Rev. 0

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.11.3 (cont.) (2) NFPA 90A, Standard for the Installation of Air Conditioning and Ventilating Systems	(cont.) Sub-Element (2): Complies with use of EEEEs	(cont.) The operative requirement of NFPA 805 for closure of ventilation openings in fire barriers is use of dampers (including sleeves) which have been tested in accordance with ANSI / UL 555, Standard for Safety Fire Dampers, and have been listed / labeled accordingly for the required fire resistance rating. Fire dampers at NMP1 meet the requirement for use of suitably tested and listed / labeled fire damper assemblies. The fire resistance ratings of fire dampers are consistent with that of the barrier, or have been evaluated and determined to be adequate for the hazards. The C-42355-C Drawing Series provides the required fire resistance rating of each damper. Therefore, NMP1 meets the intent of NFPA 805, Section 3.11.3 for conformance to applicable NFPA 90A requirements for fire dampers.	(cont.) Drawing C-42355-C, "Fire Damper Schedule," SH 1, Rev. 2 Drawing C-42355-C, "Fire Damper Schedule," SH 2, Rev. 3 Drawing C-42355-C, "Fire Damper Schedule," SH 3, Rev. 0 FPEE 1-89-003, "Fire Protection Engineering Evaluation" [Existing 12" x 16" Duct Passing Through Reactor Building Airlock], Rev. 0 FPEE-0-92-002, "Inaccessible Fire Damper Operability," Rev. 0 FPEE-1-95-001, "Fire Damper 211-54 Installation Deviation," Rev. 0 FPEE-1-02-001, "Performance-Based Review of Fire Damper Test Procedures," Rev. 0 UFSAR Sec. X.N, Appendix 10A, Rev. 21, Table 1.2.2, pg. 10A-89a

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.11.3 (cont.) (3) NFPA 101, Life Safety Code 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 barriers 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.	(cont.) Sub-Element (3): Complies with clarification	(cont.) **Clarification:** The requirements of NFPA 101 applicable to fire doors and fire dampers are bounded 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.	(cont.) Drawing B-42346-C, "Fire Barrier Penetration Seal Details – Fire Damper Seals, Seal Detail FS-11," SH 12, Rev. 1 NFPA 90A-1999, "Standard for the Installation of Air-Conditioning and Ventilating Systems"

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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. (a) 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.	Sub-Element (a): Complies with use of EEEEs	The primary fire test qualification bases for NMP1 penetration seal designs is the test program conducted by Underwriters Laboratories in accordance with IEEE 634-1978 documented via UL Project 79NK8678. Additionally, in 1988 the NRC issued IN 88-04 to alert licensees that some fire barrier penetration seal designs might not be adequately qualified for the design rating of the penetrated fire barrier. On this basis, in the 1988-1989 timeframe, NMPC implemented a 100-percent penetration seal validation effort. A detailed review of the UL fire test was performed and documented via FPEE-1-88-001. Some penetration seal discrepancies were addressed on a generic basis. The balance of penetration seal issues identified were then addressed on a case-by-case basis via FPEEs. Therefore, the referenced documents serve to document compliance of the NMP1 penetration seal program with the commitments described in the NRC SER dated July 26, 1979.	UL Fire Test Report, File NC601-1, -2, -3, -4, Project 79NK8678, "Report on Floor and Wall Penetrations Fire Stops," Nov. 17, 1980 CTL Fire Test Report, "Fire and Hose Stream Tests for Penetration Seal Systems (NMP2-PSS5)," April 1986 B-42346-C Series Drawings, "Fire Barrier Penetration Seal Details," SH-01 through SH-31 FPEE-1-88-001, "Penetration Seal Evaluation," Rev. 0 FPEE 1-89-005, Thermal and Structural Evaluation of Penetrations P-11 and P-22 FPEE 1-90-002, "NMP-1 Spacer Detail for Fire-Rated Penetration Seals," Rev. 0 FPEE 1-90-003, "NMP-1 Fire-Rated Penetration Seal Detail FS-1," Rev. 1A FPEE 1-90-004, "NMP-1 Fire Rated Penetration Seal Detail FS-2, Rev. 1A

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.11.4 (cont.)			**(cont.)** FPEE 1-90-012, "Penetrations for Support Bolts for Station Battery Racks," Rev. 0 FPEE 1-90-020, "NMP-1 Fire-Rated Penetration Seal Detail FS-8," Rev. 1 FPEE-1-91-005, "Penetration Seal Detail FS-2 for Penetration 3RF-80," Rev. 0 FPEE 1-92-001, "Penetration Seal Enlarged Boot Detail," Rev. 0 FPEE 1-92-003, "Existing Penetration Seals Use of Asbestos," Rev. 0 FPEE 1-03-002, "Sealing Requirement for Gaseous Suppression System Boundaries," Rev. 0 FPEE 1-01-004, "Applicability of Seal Detail FS-25 for Use in Metallic Enclosures," Rev. 0 NRC Safety Evaluation Report (SER), 7/26/79

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.11.4 (cont.) (b) 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* conduits 4 in. 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 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. shall constitute an acceptable smoke and hot gas seal in this application.	Sub-Element (b): *Complies with use of* EEEE	To address installation of internal conduit seals on a *generic basis, a plant-wide walkdown was conducted* in 1991 to inspect all conduits greater than 2-in. diameter that penetrate fire-rated barriers. Conduit smaller than 2-in diameter do not require internal seals and are considered satisfactory. Internal conduit seals were required to be installed or repaired for a small number of conduits. FPEE 1-91-006 was issued to document the walkdown methodology and results. Provision of fire-rated internal conduit seals or non-fire rated smoke and hot gas seals are based on conduit size and distance from the barrier criteria prescribed by seal detail drawing series B42346C. Conduit seal maintenance activities are controlled by procedure N1-EPM-GEN-001 'Penetration Maintenance." This procedure is used to ensure conduit seals meet the required acceptance criteria and are determined to be acceptable. Additionally, specification E127 "Standard Specification for Electrical Installation Activities at Nine Mile Point Nuclear Station #1" provides additional guidance on new and existing conduit seals.	FPEE-1-91-006, "Interior Conduit Sealing Walkdown," Rev. 0 Drawing Series B42346C, "Fire Barrier Penetration Seal Details" N1-EPM-GEN-001, "Penetration Maintenance" Revision 00201 E127 "Standard Specification for Electrical Installation Activities at Nine Mile Point Nuclear Station #1" Revision 20

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
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 GL 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.	Complies with use of EEEE	ERFBS are credited for FA 18 only. The Eternit, Inc. Promat-H ERFBS installed in FA18 and credited by the NSCA for protection of Conduit 171-66 was not tested in accordance with GL 86-10, Supplement 1. However, FPEE-1-95-001 has deemed the qualification testing that was performed to be adequate, and the Promat-H ERFBS to be capable of resisting the hazards in the area for a 3-hour fire resistance rating. The credited bases for acceptance remain valid and the EEEE meets applicable quality requirements. ERFBS are not credited in other fire areas to meet the NFPA 805, Chapter 4 requirements for any of the fire areas described in the Fire Area Analysis (FAA) and NEI 04-02 B-3 Table.	UFSAR Sec. X.N, Appendix 10B, Rev. 21, Sec. 6.4, pg. 10B-156 FPEE-1-95-002, "Deviations to Appendix R 3-Hour Fire Barriers," Rev. 1

Chapter 3 Reference and Requirements/Guidance	Compliance Statement	Compliance Bases	Reference Documents
3.11.5 (cont.) 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 Test Acceptance Criteria for Fire Barrier Systems Used to Separate Safe Shutdown Trains Within the Same Fire Area, Supplement 1, functionality of the cable at these elevated temperatures shall be demonstrated. Qualification demonstration of these cables shall be performed in accordance with the electrical testing requirements of Generic Letter 86-10, Supplement 1, Attachment 1, Acceptable 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

101 Pages Attached

2.4.2.1 Nuclear Safety Capability Systems 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 mal-operation of those components needed to meet the nuclear safety criteria shall be included. Availability and reliability of equipment selected shall be evaluated.

NEI 00-01 Ref	NEI 00-01 Guidance
3 Deterministic Methodology	This section discusses a generic deterministic methodology and criteria that licensees can use to perform a post-fire safe shutdown analysis to address regulatory requirements. 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.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

This document addresses the comparison of the deterministic methodology used for the existing the Nine Mile Point 1 (NMP1) Safe Shutdown Analysis and the requirements of 10CFR50 Appendix R, Sections III.G.1, III.G.2 and III.G.3 against the compliance requirements and criteria specified in NFPA 805. The subsequent sections determine the extent the analysis meets the requirements as described in NEI 04-02. This documents a line by line review and comparison against the methodology and criteria provided in Chapter 3 of NEI 00-01, Revision 2. The deterministic methodology described in Section 3 and Figure 3-1 of NEI 00-01 was utilized as documented in the following Table B-2 sections. The NMP1 safe shutdown methodology utilizes a computer oriented database to model data relationships for systems, components, and cables used to comply with the requirements for post fire safe shutdown. This review of modifications, procedural controls, repair procedures and previously approved configurations and boundaries demonstrates that the safe shutdown analysis generally meets the Nuclear Safety Performance Criteria including the information provided in Appendix B of NFPA 805 related to circuit criteria and Multiple Spurious Operations (MSOs).

Reference Document

EIR 51-9133191, NSCA, Section 9.0

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 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 effects 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 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

To achieve post-fire safe shutdown, the Safe Shutdown Systems, their functions, and components required to support the safe shutdown functions were identified. P&IDs and Electrical drawings were marked up and annotated to select equipment and specific flow paths for each system required to support safe shutdown. This information was populated into a computer database to provide a database oriented approach to model Appendix R data relationships.

Safe shutdown systems, components, and cables were selected using the criteria and assumptions of NEI 00-01, Rev. 2, Sections 3.1.1 and 3.1.2 to satisfy each safe shutdown function performance goals, including process monitoring and support systems for each path. NEI 00-01 was used as the guidance in developing the Safe Shutdown Equipment List and the Safe Shutdown success paths. Safe Shutdown paths were designed based on the combination of systems in the respective fire area.

The ability to achieve post-fire Safe Shutdown (SSD) is assured by having at least one safe shutdown path of the required systems, structures, and components to remain free of fire damage. This assurance that the safe shutdown equipment is available supports the required performance goals identified in the guidance, maintains the integrity of the fuel, reactor pressure vessel and primary containment. The SSD path used to achieve post-fire safe shutdown is comprised of SSD systems and components that remain free of fire damage.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0, 5.0, 5.1, 5.2, 8.6.2, and 8.6.3
HNP RAI 3-6, RAI 3-9, RAI 3-10, and RAI 3-11, NRC Requests for Additional Information dated August 6, 2009 (ML092170715)
Closure of National Fire Protection Association 805 Transition Program Frequently Asked Question Number 06-0006 (ML070030117)
ONS RAI 3-35, NRC Request for Additional Information dated November 18, 2009 (ML092920347)

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	**NEI 00-01 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 paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.1 Safe Shutdown Paths For BWRs	[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 primary means of achieving and maintaining hot shutdown following a fire coincident with a Loss of Offsite Power (LOOP) is via the Emergency Condenser (EC) system. The EC system consists of two redundant emergency cooling loops; each loop is capable of independently accomplishing hot shutdown. Therefore, this option provides two redundant paths *for obtaining hot shutdown.*

The EC system operates by natural circulation. Steam flows from the vessel through the EC tubes. Condensate returns to the vessel through a reactor recirculation loop. Boiling of water in the secondary side of the ECs, which is vented to the atmosphere, provides the necessary cooling.

In the event the preferred shutdown method is not available, the plant can be shut down by opening three Electromatic Relief Valves (ERVs) and discharging steam to the Torus to reduce pressure. When reactor pressure reaches approximately 365psig, Core Spray (CS) may be initiated. CS is a two loop system. Operation of one LOOP is adequate to achieve shutdown. Eventually, the reactor vessel floods to the point where the ERVs are passing fluid to the Torus rather than steam, in essence placing the Reactor Coolant System in recirculation through the Torus. During this process, decay heat is removed by operation of the Containment Spray System in conjunction with the Containment Spray Raw Water system. This shutdown method will bring the plant directly to CSD. Fully flooding the Reactor Pressure Vessel negates the need for another system to provide inventory makeup. AC power is required to initiate this shutdown method.

Reference Document

EIR 51-9133191, NSCA, Sections 5.1 and 5.2

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	**NEI 00-01 Guidance**
3.1.1.2 SRVs and LPCI/CS	[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

For NMP1, using the Electromatic Relief Valves (ERVs) to remove decay heat to the torus and low pressure Core Spray system is an acceptable method for achieving safe shutdown. *The Emergency Condensers provide one shutdown flow path while the ERVs provide an Alternate Shutdown method.*

Reference Document

EIR 51-9133191, NSCA, Section 4.0
NMP1 Safety Evaluation # 84-18, ADS Logic Modification

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	Comments
Not Applicable	NMP1 is a BWR plant.

Alignment Statement

Not Required

Alignment Basis

NMP1 is a BWR plant.

Reference Document

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.4 Alternative Shutdown Classification	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

The criteria of 10CFR50 Appendix R, Section III.G.3, requires alternate or dedicated shutdown capability for all plant areas where the protection of systems for hot shutdown does not satisfy the requirements of Sections III.G.1 and III.G.2 of Appendix R. The hot shutdown system to be used for a control room evacuation event is the EC system.

NMP1 has two Remote Shutdown Panels installed for the purpose of monitoring the Plant Shutdown in the event of a Control Room evacuation. Modifications were performed which hardened the EC system from spurious isolations due to the effects of a control complex fire. Upon receiving either a high reactor pressure signal or a low-low reactor water level signal, the ECs will automatically initiate, independent of the control complex, due to the shutdown supervisory control system redundant initiation logic located in the reactor building, although Operator action will initiate the safe shutdown systems prior to its automatic initiation to conserve reactor vessel inventory.

Reference Document

EIR 51-9133191, NSCA, Section 2.1.1
NMP1 Safety Evaluation #83-29, Emergency Condensers

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.5 Operable and Available	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

As stated in the Fire Area Assessments at the onset of the fire, all systems not affected by the fire are considered to be available and capable of functioning as designed. All safe shutdown systems (including redundant trains) are assumed to be operational and available for post-fire safe shutdown. Systems are assumed to be operational with no repairs, maintenance, testing, Limiting Conditions for Operation, etc. The unit is assumed to be operating at full power under normal conditions and normal lineups.

Reference Document

EIR 51-9133191, NSCA, Section 4.0

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

No accidents or other design basis events (e.g. loss of coolant accident, earthquake), single failures or non-fire induced transients are considered in conjunction with the fire.

The fire does not occur simultaneously or coincident with any other transient or abnormal condition, except for loss of offsite power (LOOP) and those conditions resulting directly from the effects of a fire. No credit is taken for offsite power availability; however, offsite power is considered to be available if the fire effects produce more conservative results.

Reference Document

EIR 51-9133191, NSCA, Section 9.0

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.7 Offsite Power Availability	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	ONS RAI 3-40 is addressed; No credit is taken for offsite power.

Alignment Statement

Aligns

Alignment Basis

Offsite power is 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 is 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 is demonstrated both where offsite power is available and where offsite power is not available for 72 hours. Offsite power has not been specifically analyzed. There are no fire areas where offsite power is credited.

Reference Document

EIR 51-9133191, NSCA, Section 9.0
ONS RAI 3-40, NRC Request for Additional Information dated July 30, 2010 (ML102110394)

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Credited safe shutdown systems and components are not always safety related. Most components are safety related due to their emergency functions, but they are not required to be safety-related.

Reference Document

EIR 51-9133191, NSCA, Section 4.0

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.9 72-hour Coping Period	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

Alignment Basis

Appendix R requires cold shutdown of the reactor within 72 hours for fire events, with or without offsite power available. The Safe Shutdown Analysis identifies the safe shutdown systems and components which are powered by on-site sources, where at least one train can be repaired or made operable within the 72 hours. This is accomplished for each fire area with one of four designated cold shutdown trains. Offsite power is not credited with providing any power or beneficial shutdown methods during the 72 hours, thereby meeting this requirement.

Cold Shutdown Options:

A. *Train 11 - Shutdown cooling*, RBCLC, ESW, *CRD system (makeup)*.

B. Train 12 - Shutdown cooling, RBCLC, ESW, CRD system (makeup).

C. Train 11 - ERVs, core spray, containment spray, containment spray raw water.

D. Train 12 - ERVs, core spray, containment spray, containment spray raw water.

Note: As part of the NMP1 defense-in-depth approach to fire protection, provisions have been made for permanent installation of a feedwater/fire protection water spool piece, which will provide an emergency makeup source from the diesel fire pump for cold shutdown inventory control.

Shutdown Cooling with LOOP (Options A and B)

The primary means for achieving and maintaining cold shutdown following a fire coincident with a LOOP is via the shutdown cooling system. The shutdown cooling system, supported by the RBCLC and the ESW systems, removes decay heat from the reactor vessel to the UHS. Emergency AC power (onsite DGs) is required for this mode of cold shutdown.

Reference Document

EIR 51-9133191, NSCA, Section 4.0 and 5.0
HNP RAI 3-7 and RAI 3.8, NRC Request for Additional Information dated August 6, 2009 (ML092170715)

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.10 Manual Initiation of Systems	Manual initiation from the main control room or emergency control stations of systems required to achieve and maintain safe shutdown is acceptable where permitted by current regulations or approved by NRC (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

Manual initiation of safe shutdown related equipment from either the control room, emergency control stations or approved locations other than the primary control stations is an acceptable means for compliance based on the current regulations. The safe shutdown performance and design requirements for the reactivity control function can be met without automatic scram capability. The post-fire safe shutdown analysis need only provide the capability to manually scram the reactor. Automatic functions of components have been included for selected systems. Impacts due to spurious actuation of automatic systems are included in the evaluation of the analysis.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0, 5.1, 5.2, and 5.4

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.1.11 Multi-unit Plant	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

As stated in the UFSAR, simultaneous fires affecting NMP1 and NMP2 are not anticipated, due to the spatial separation between the two units and the designed separation (fire barrier) at facility interfaces. NMP1 and NMP2 do not share common facilities for the support of reactor operation or generation of electricity. However, there is the capability to cross-tie the firewater system between NMP1 and NMP2 via manual cross-tie valves.

Reference Document

UFSAR, Appendix 10A, Section 2.1.9

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.1 Reactivity Control	[BWR] Control Rod Drive System The safe shutdown performance and design requirements for the reactivity control function can be met without automatic scram/trip capability. Manual scram/reactor trip is credited. The post-fire safe shutdown analysis must only provide the capability to manually scram/trip the reactor. 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. [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	Comments
Applicable	None

Alignment Statement

Aligns With Intent

Alignment Basis

As documented in the current NMP1 Safe Shutdown Analysis, the reactivity control function is capable of achieving and maintaining cold shutdown reactivity conditions. The safe shutdown performance and design requirements for the reactivity control function can be met without automatic scram capability. The post-fire safe shutdown analysis provides the capability to manually scram the reactor. Manual reactor scram is accomplished via the scram valves. Plant operators can manually scram the plant from the control room or the remote shutdown panel. The capability to manually scram/trip the reactor is provided in NMP1 Special Operating Procedures N1-SOP-21.1, Fire in Plant, and N1-SOP-21.2, Control Room Evacuation, with reference to the Emergency Operating Procedures for operators to manually vent the instrument air header or to remove RPS power. This action is not considered a "time critical" action because the Mode switch is placed in shutdown and all control rods inserted prior to evacuating the control room. Also, NMP1 complies with the position in BWROG document BWROG-TP-11-011 entitled "BWROG Assessment of Generic Multiple Spurious Operations (MSOs) in Post-Fire Safe Shutdown Circuit Analysis for the Operation of BWR Plants" for manual scram; thereby, supporting that this is not a time critical action.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0 and 5.3
NMP1 Emergency Operating Procedure N1-EOP-2, RPV Control
NMP1 Emergency Operating Procedure 3, Failure to Scram
NMP1 Emergency Operating Procedure 3.1, Alternate Control Rod Insertion
NMP1 Special Operating Procedure, N1-SOP-21.2, Control Room Evacuation, pg. 3
NMP1 Special Operating Procedure, N1-SOP-21.1, Fire in Plant, pg. 2

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The pressure control function is capable of safely reducing reactor vessel pressure.

For the preferred shutdown method, pressure control is achieved through control of the cooldown rate of the emergency condensers. The Electromatic Relief Valves (ERVs) are maintained closed.

In the event the preferred shutdown method is not available, the plant can be shut down by opening three ERVs and discharging steam to the Torus to reduce pressure. For this secondary method, the ERVs are opened manually to depressurize the vessel to allow injection using low pressure systems. Automatic initiation of the ADS is not a required function.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0, 5.1, 5.2, and 5.3

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.3 Inventory Control	[BWR] Systems selected for the inventory control function should be capable of supplying sufficient reactor coolant to achieve and maintain hot shutdown. Manual initiation of these systems is acceptable. Automatic initiation functions are not required. Spurious actuation of automatic systems, however, should be evaluated (High Pressure Coolant Injection, High Pressure Core Spray, Reactor Core Isolation Cooling, etc.). [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	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

For the preferred hot shutdown method, reactor vessel make-up is required 8 hours after operation of the Emergency Condensers (EC) ensues. A Control Rod Drive pump is used to provide vessel make-up. The inventory makeup provided by the CRD pump is part of cold shutdown to raise the Reactor water level and make the shutdown cooling system more effective. Restoration of the CRD pump is via repair actions to secure power sources, etc.

NMP1 has a defense-in-depth inventory control approach for fire protection by using the firewater to feedwater connection in accordance with NMP1 operating procedures, which provides an emergency makeup source from the diesel fire pump for cold shutdown inventory control.

As a secondary method, the vessel can be flooded by the Core Spray system after depressurization. This fulfills the vessel inventory make-up by default.

Spurious Actuations are addressed specifically in the Fire Area Assessments. They are considered to exist from the onset of the fire and for the duration of the shutdown process.

Reference Document

EIR 51-9133191, NSCA, Sections 5.3 and 8.6

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

***NEI 00-01* Ref**	***NEI 00-01* Guidance**
3.1.2.4 Decay Heat Removal	[BWR] Systems selected for the decay heat removal function(s) should be capable of:

- Removing sufficient decay heat from primary containment, to prevent containment over-pressurization and failure.
- Satisfying the net positive suction head requirements of any safe shutdown systems taking suction from the containment (suppression pool).
- Removing sufficient decay heat from the reactor to achieve cold shutdown. (This is not a hot shutdown requirement).

[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	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The decay heat removal function is capable of achieving and maintaining hot and cold shutdown. The EC system operates by natural circulation where steam flows upward to the condenser(s) and returns as condensate to the Reactor Pressure Vessel. Decay heat is removed through the transfer of heat from the reactor coolant to the shell side water of the EC which vents the developed steam to atmosphere. Operation of either EC loop can sustain Hot Shutdown conditions for 8 hours without the need for makeup from the Condensate Storage Tank. The decay heat removal process also reduces reactor pressure. When reactor pressure is reduced to 120 psig and reactor temperature is reduced to 350 degrees F, the plant can be transitioned to Cold Shutdown by initiating shutdown cooling.

In the event the preferred shutdown method is not available, the plant can be shut down by opening three ERVs and discharging steam to the Torus to reduce pressure. When reactor pressure reaches approximately 365 psig, Core Spray (CS) may be initiated. CS is a two loop system. Operation of one loop is adequate to achieve shutdown. Eventually, the reactor vessel floods to the point where the ERVs are passing fluid to the Torus rather than steam, in essence placing the Reactor Coolant System in recirculation through the Torus. During this process, decay heat is removed by operation of the Containment Spray System in conjunction with the Containment Spray Raw Water system. Containment spray and containment spray raw water systems are utilized to remove heat from the torus water and maintain it within the core spray and containment spray pumps' net positive suction head (NPSH) requirements. This shutdown method will bring the plant directly to cold shutdown. Fully flooding the Reactor Pressure Vessel negates the need for another system to provide inventory makeup. AC power is required to initiate this shutdown method.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0 and 5.1

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 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* 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
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The process monitoring function is to be provided for all safe shutdown paths. NEI 00-01 refers to NRC IN 84-09, Attachment 1, Section IX, as providing guidance on the instrumentation acceptable to and preferred by the NRC for meeting the process monitoring function. This instrumentation is that which monitors the process variables necessary to perform and control the functions specified in Appendix R Section III.L.1. Such instrumentation must be demonstrated to remain unaffected by the fire. The IN 84-09 list of process monitoring is applied to alternative shutdown (III.G.3). In general, process monitoring instruments similar to those listed below are needed to successfully use existing operating procedures related to post-fire shutdown (including Abnormal Operating Procedures).

-Reactor coolant level and pressure
-Suppression pool level and temperature
-Level indication for tanks needed for safe shutdown
-Diagnostic instrumentation for safe shutdown systems

The Reactor Protection System is utilized to satisfy the process monitoring objectives throughout hot and cold 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.

The following process monitoring functions are provided to support post-fire shutdown:

Primary and Secondary Methods

-Reactor coolant level
-Reactor coolant pressure
-Reactor coolant temperature
-Emergency Diesel Generator parameters

Primary Method

-*Emergency Condenser level*

Secondary Method

-Torus level
-Torus temperature
-Drywell pressure
-Drywell temperature
-Containment Spray water temperature
-Containment Spray pump discharge pressure

Reference Document

EIR 51-9133191, NSCA, Sections 4.0 and 5.3

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	***NEI 00-01 Guidance***
3.1.2.6 Support Systems	Blank Heading - No Specific Guidance

Applicability	**Comments**
Applicable	None

Alignment Statement

Not Required

Alignment Basis

Generic Heading: Alignment discussed in subsequent sub-sections.

Reference Document

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.2.6.1 Electrical Systems	AC Distribution System Power for the Appendix R safe shutdown equipment is typically provided by a medium voltage system such as 4.16 KV Class 1E busses either directly from the busses or through step down transformers/load centers/distribution panels for 600, 480 or 120 VAC loads. For redundant safe shutdown performed in accordance with the requirements of Appendix R Section III.G.1 and 2, power may be supplied from either offsite power sources or the emergency diesel generator depending on which has been demonstrated to be free of fire damage. No credit should be taken for 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

No credit has been taken for the loss of offsite power, however in the event that offsite power is lost, NMP1 has two Emergency Diesel Generators (EDGs). The EDGs will supply the required medium and low voltage safe shutdown loads with AC power. The EDGs are designed to start automatically on loss of offsite power to re-energize emergency busses 102 and 103.
The 125V DC distribution panels supply power to the 120V AC distribution panels via static inverters. These distribution panels typically supply power for instrumentation necessary to complete process monitoring functions. The 125V DC distribution system supplies control power to various 125V DC control panels including switchgear breaker controls. Vital AC power can be provided via RPS uninterruptible power supplies (UPS) 162A, 162B, 172A and 172B.

This 125V DC distribution system is credited to support post fire safe shutdown. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the EDGs to become operational. Once the EDGs are operational, the 125V DC distribution system can be powered from the diesels through the battery chargers.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0 and 5.3

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Various cooling water systems are required to support safe shutdown system operation:

-The Reactor Building Closed Loop Cooling and Emergency Service Water Systems provide cooling to the control room ventilation system and Shutdown Cooling Heat Exchangers
-The EDGs are cooled by the EDG Raw Water pumps
-The Containment Spray and Containment Spray Raw Water Systems cool the torus for the secondary cooldown method
-The Chilled Water System is provided for control room ventilation

Reference Document

EIR 51-9133191, NSCA, Section 5.3

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 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

HVAC Systems required to support post-fire shutdown are:

-Main control room ventilation (except for control room evacuation)
-EDG rooms (fans and roll-up doors)

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. Paths for the ventilation system are as follows:

-Path A – Recirculation flow path
-Path B – Smoke purge flow path

-Path C – Train 11 positive pressure flow path.
-Path D – Train 12 positive pressure flow path

Given the failure modes of dampers and EDG power requirement, Paths C & D are generally credited for post-fire shutdown. Any ventilation path can be supported by Train 11 or Train 12 Chilled Water.

Reference Document

EIR 51-9133191, NSCA, Section 5.3

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.3 Methodology For Shutdown System Collection	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	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Safe shutdown systems and functions required to satisfy the safe shutdown performance goals were developed and identified from available plant documentation. This documentation includes but is not limited to electrical one line diagrams, schematics, piping and instrumentation diagrams (P&IDs), operating procedures, UFSAR, Fire Hazards Analysis, and the systems descriptions.

Reference Document

EIR 51-9133191, NSCA, Section 5.4

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Alignment Statement

Aligns

Alignment Basis

Safe shutdown systems, components, and cables were selected using the criteria and assumptions of NEI 00-01, Sections 3.1.1 and 3.1.2 to satisfy each safe shutdown function performance goal, including process monitoring and support systems. NEI 00-01 was used as the guidance in developing the Safe Shutdown Equipment List and the Safe Shutdown success paths.

The following combination of systems are capable of achieving safe shutdown functions.

1) Reactivity Control – Control Rod Drive System

The safe shutdown performance and design requirements for the reactivity control function can be met without automatic scram capability. The post-fire safe shutdown analysis need only provide the capability to manually scram the reactor.

2) Pressure Control Systems - ERVs

The ERVs are opened to maintain hot shutdown conditions or to depressurize the vessel to allow injection using low pressure systems. These are operated manually. Automatic initiation of the ADS is not a required function.

3) Inventory Control

Systems selected for the inventory control function are 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.

4) Decay Heat Removal

Systems selected for the decay heat removal function(s) are 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 torus.
- Removing sufficient decay heat from the reactor to achieve cold shutdown.

5) Process Monitoring

The process monitoring function is to be provided for all safe shutdown paths. NEI 00-01 refers to NRC IN 84-09, Attachment 1, Section IX, as providing 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 functions is applied to alternative shutdown (III.G.3). In general, process monitoring instruments similar to those listed below are needed to successfully use existing operating procedures related to post-fire shutdown (including Abnormal Operating Procedures).

- Reactor coolant level and pressure
- Torus level and temperature
- Level indication for tanks needed for safe shutdown
- 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.

Note – For NMP1, Emergency Condenser level is a required process monitoring function.

6) Support Systems

A. Electrical Systems

AC Distribution System

Power for safe shutdown equipment is supplied from either offsite power sources or the emergency diesel generator. No credit is taken for a fire causing a loss of offsite power.

DC Distribution System

The 125V DC distribution system supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels also supply power to the 120V AC distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to complete 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 125V DC distribution systems can be powered from the diesels through the battery chargers.

B . Cooling Systems

Various cooling water systems are required to support safe shutdown system operation, based on plant-specific considerations. Cooling system uses include:

- SDC Heat Exchanger cooling water
- Safe shutdown pump cooling (seal coolers, oil coolers)
- Diesel generator cooling
- HVAC system cooling water

C. HVAC Systems

HVAC Systems are 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 uses include:

- Main control room, cable spreading room, relay room
- ECCS pump compartments
- Diesel generator rooms
- Switchgear Rooms

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. Paths for the ventilation system are as follows:

-Path A – Recirculation flow path
-Path B – Smoke purge flow path
-Path C – Train 11 positive pressure flow path.
-Path D – Train 12 positive pressure flow path

Given the failure modes of dampers and EDG power requirement, Paths C & D are generally credited for post-fire shutdown. Any ventilation path can be supported by Train 11 or Train 12 Chilled Water.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0, 5.0, and 5.4

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 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

Safe shutdown systems, components, and cables were selected using the criteria and assumptions of NEI 00-01, Sections 3.1.1 and 3.1.2 to satisfy each safe shutdown function performance goal, including process monitoring and support systems for each path. NEI 00-01 was used as the guidance in developing the Safe Shutdown Equipment List and the Safe Shutdown success paths. The shutdown paths and equipment selection for the shutdown performance goals are identified in the NSCA.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0 and 5.0

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.1.3.4 Assign Shutdown Paths to Each Combination of Systems	Assign a path designation to each combination of systems. The path will serve to document the combination of systems relied upon for safe shutdown in each fire area. Refer to Attachment 1 to this document 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

Alignment Basis

Safe shutdown systems, components, and cables were selected using the criteria and assumptions of NEI 00-01, Sections 3.1.1 and 3.1.2 to satisfy each safe shutdown function performance goal, including process monitoring and support systems for each path. NEI 00-01 was used as the guidance in developing the Safe Shutdown Equipment List and the Safe Shutdown success paths. Safe Shutdown Paths were designed based on the combination of systems in the respective fire area.

The major systems for safe shutdown success paths are as follows:

- SUCCESS PATH "A"

Hot shutdown is achieved via the use of ECs 111 & 112 for Decay Heat Removal. Cold shutdown is accomplished via the use of Train 11 SDC System components supported by the Train 11 RBCLC and ESW Systems. The Train 11 CRD pump or the DFP is used for inventory control/RPV makeup. Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system while other components may require local operator action due to a potential loss of instrument air.

- SUCCESS PATH "B"

Hot shutdown is achieved via the use of ECs 121 & 122 for Decay Heat Removal. Cold shutdown is accomplished via the use of Train 12 SDC System components supported by the Train 12 RBCLC and ESW Systems. The Train 12 CRD pump or the DFP is used for inventory control/RPV makeup. Path B is generally supported by the Train 12 AC and DC power systems. Some components in Path B systems are powered by the Train 11 AC and/or DC power system while other components may require local operator action due to a potential loss of instrument air.

- SUCCESS PATH "C"

Both hot shutdown and cold shutdown are achieved via use of the Train 11 ERVs (PSV-01-102A, PSV-01-102B, PSV-01-102E) to depressurize the reactor. When pressure drops to the appropriate level, the CS System is used to flood the vessel and place the RCS in a recirculation mode to the Torus. Torus cooling is provided by the CTS system supported by the CTSRW System. These systems are maintained in service to achieve cold shutdown. Path C is generally supported by the Train 11 AC and DC power systems. Some components in Path C systems are powered by the Train 12 AC and/or DC power system.

- SUCCESS PATH "D"

Both hot shutdown and cold shutdown are achieved via use of the Train 12 ERVs (PSV-01-102C, PSV-01-102D, PSV-01-102F) to depressurize the reactor. When pressure drops to the appropriate level, the CS System is used to flood the vessel and place the RCS in a recirculation mode to the Torus. Torus cooling is provided by the CTS System supported by the CTSRW System. These systems are maintained in service to achieve cold shutdown. Path D is generally supported by the Train 12 AC and DC power systems. Some components in Path D systems are powered by the Train 11 AC and/or DC power system.

Path A and Path B are the preferred shutdown paths for both hot shutdown and cold shutdown as these paths present the least thermal hydraulic impact to the plant.

It is possible that for some fire areas, one Path may be employed for hot shutdown and another for cold shutdown. This depends upon the impacts to power supplies and other components in any particular fire area, the controls design of credited components and the use of damage repair procedures. Consequently, it is possible that Path B may be credited for hot shutdown and Path D for cold shutdown, or any other combination. However, if Path C or D is credited for hot shutdown, that path would also be credited for cold shutdown due to the dynamics of the shutdown process.

Reference Document

EIR 51-9133191, NSCA, Section 5.3

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2 Safe Shutdown Equipment Selection	The previous section described the methodology for selecting the systems and paths necessary to achieve and maintain safe shutdown for an exposure fire event (see Section 5.0 DEFINITIONS for "Exposure Fire"). This section describes the criteria/assumptions and selection methodology for identifying the specific safe shutdown equipment necessary for the systems to perform their Appendix R 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 shutdown 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	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	*Comments*
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	*NEI 00-01 Guidance*
3.2.1.1 Safe Shutdown Equipment Categories	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

Alignment Basis

The Safe Shutdown Equipment List (SSEL) is a list of analyzed components that are utilized in the post-fire safe shutdown analysis to ensure that one success path (structures, systems, and components) necessary to achieve safe shutdown is free of fire damage without crediting plant or system repair.

The current NMP1 SSEL and Safe Shutdown Analysis was reviewed against the criteria outlined in NEI 00-01 (Sections 3.1 and 3.2) for safe shutdown systems and equipment selection. This review addressed potential fire induced circuit failure issues, either within or beyond the plant's existing licensing basis. Additional equipment has been included to address any multiple spurious operation concerns.

The SSEL is divided into primary and secondary components. Primary and secondary components were defined as being consistent with NEI 00-01 guidance. Equipment identified as primary components is included in the SSEL. Equipment identified as secondary components is included in the SSEL Database with the primary component(s) that would be affected by fire damage to the secondary component. By doing this, the SSEL is kept to a manageable size and the equipment included in the SSEL can be readily related to required post-fire safe shutdown systems and functions.

Secondary components were generally combined with primary components except where groups of secondary components were defined as "pseudo-components."

A "pseudo-component" is an artificial association of equipment and cables that perform a common function as a single entity for analysis purposes. The "pseudo-component" is assigned for analysis purposes only and is not an actual plant hardware designation. The concept of a "pseudo-component" was developed to account for those cables which constitute a circuit common to several components. The use of "pseudo-components" precludes the need to repeat cable selection and circuit analysis of these cables for each primary component. This generic name is interlocked with the affected primary components for analysis purposes and it inherits the attributes (path, system, train, etc.) of the components that it may affect. The nomenclature of the "pseudo-component" is similar to other equipment as defined in the plant equipment database.

Reference Document

EIR #51-9133191, NSCA, Section 5.4
EIR 51-9177678-000, Definitions Section

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1.2 Manual Valves and Piping	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	Instrument tubing failure damage due to a fire is addressed in NSCA (EIR #51-9133191, Section 8.4).

Alignment Statement

Aligns With Intent

Alignment Basis

The NMP1 fire area assessments assume that an exposure fire does not adversely affect the ability of manual valves and piping to perform their pressure boundary or safe shutdown function. Fire damage to valves has been 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. Post-fire operation of manual valves within the affected fire area has been evaluated in the Fire Area Analysis.

Instrument sensing lines were reviewed for susceptibility to physical fire damage that may cause a loss of inventory. Sensing lines for SSEL components are constructed of either stainless steel or carbon steel. Consequently, they are not susceptible to physical damage as the result of a postulated fire.

Reference Document

EIR #51-9133191, NSCA, Sections 5.4 and 8.4
HNP RAI 3-15, NRC Request for Additional Information dated August 6, 2009 (ML092170715)

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1.3 Valves in Normal Position	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

Manual valves are assumed to be in their normal operating position as shown on P&IDs or as identified in the plant operating procedures.

Comments

None

Reference Document

EIR 51-9133191, NSCA, Sections 5.4, 8.1, and 9.0

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Check valves are assumed to close in the direction of potential flow diversion and seat properly with sufficient leak tightness to prevent flow diversion or inventory loss. Check valves do not adversely affect flow rate capability of the safe shutdown systems.

Comments

None

Reference Document

EIR 51-9133191, NSCA, Section 5.4

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.1.5 Instrument Failure	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

Instruments are assumed to fail in the most undesirable worst state, whether upscale, downscale, or mid-scale. An instrument providing a control function is assumed to provide an undesired signal to the control circuit.

Comments

None

Reference Document

EIR 51-9133191, NSCA, Sections 5.4 and 8.1

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

The Safe Shutdown Equipment List (SSEL) is a list of analyzed components that are utilized in the post-fire safe shutdown analysis to ensure that one success path (structures, systems, and components) necessary to achieve safe shutdown is free of fire damage without crediting plant or system repair. The SSEL is divided into primary and secondary components. Primary and secondary components were defined as being consistent with NEI 00-01 section 3.2.1.1 guidance. Secondary components were generally combined with primary components.

Combinations of components and systems with the capability to satisfy the required safe shutdown functions were designated as safe shutdown flow paths. P&IDs and Electrical drawings were marked up and annotated to highlight the selected primary safe shutdown equipment and flow paths for each system in support of each shutdown path. The specific group of equipment supporting each system was populated into the safe shutdown database.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0 and 6.0

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.2.2 Identify the Equipment in Each Safe Shutdown System Flow Path	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

The safe shutdown flow paths identify the primary components that are required to meet the safe shutdown performance goals. The safe shutdown components were compiled based on each system's performance and safe shutdown function. These components establish the primary safe shutdown flowpath for system operation. Also included in the safe shutdown flow paths are those components whose spurious operation could impact safe shutdown system operability. Systems identified as necessary for the operation of the safe shutdown system under review are included in the safe shutdown equipment list and designated with the same shutdown path as the primary safe shutdown system. These components may involve branch flow paths that must be isolated and remain isolated to assure that flow will not be diverted from the primary flow path. The list of primary components may also include selected mechanical components required to support safe shutdown.

The criteria used in evaluating spurious actuation of components are those identified in NEI 00-01, Section 4, Identification and Treatment of Multiple Spurious Operations (MSO). The Nuclear Safety Capability Fire Area Assessments includes the potential impact of multiple spurious component actuations per the guidance of NEI 00-01. MSO component combinations, as documented in the Technical Report on Identification & Classification of the NMP1 MSO Scenarios Using an Expert Panel - Review of New Generic Scenarios, were addressed in EIR #51-9133191 and included in the fire area assessments.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0, 5.0, 6.0, 8.6.1, and 8.6.3
Technical Report on Identification & Classification of the NMP-1 MSO Scenarios Using an Expert Panel - Review of New Generic Scenarios

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

NEI 00-01 Ref	NEI 00-01 Guidance
3.2.2.3 Assign Safe Shutdown Flow Paths	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

Alignment Basis

The Safe Shutdown Equipment List (SSEL) includes equipment for each system supporting the flow paths needed to achieve safe shutdown. This equipment, identified from the highlighted P&IDs, includes both the normal and diversion flow paths required to meet the system performance goals and safe shutdown functions. These components also include valves or equipment that could impact safe shutdown by spuriously operating or whose failure would threaten the capability to achieve safe shutdown. The components were populated into the SSEL database and assigned to safe shutdown system success paths.

During the cable selection process additional support components such as electrical distribution equipment were added to the SSD paths and populated into the database. The database reports produce tables listing equipment and related information which is very similar to the table provided in Attachment 3, of NEI 00-01. This group of components and the various equipment related attributes makes up the SSEL.

Instrument sensing lines for level, pressure, flow, etc. that are exposed to a fire are considered to have the potential of causing erratic or unreliable indication. The instrument tubing lines were traced and their routes correlated to fire areas. Cable identifications were given to the sensing lines and were subjected to the same compliance issues and analytical techniques as safe shutdown cables and similarly analyzed for separation in the fire area assessments.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0, 5.3, 6.0, and 8.4

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

The systems and components of the Safe Shutdown Equipment List (SSEL) were selected to meet the performance goals to achieve post-fire shutdown as specified in 10CFR50, Appendix R. Additional secondary components which are modeled as a result of the primary component selections were also populated into the database.

This additional equipment related information necessary to perform the Fire Area Assessments was collected and included in the SSEL. The SSEL contains the following information; system, train, component, component description, path, Hi/Lo pressure interface determination, normal position, hot shutdown position, cold shutdown position, failed electrical position, failed air position, Fire Area, and Fire Zone.

The SSEL database contains equipment and related information similar to the information identified in Attachment 3 of NEI 00-01. The SSEL contains the primary components which are required for hot shutdown. The secondary components are typically items found within the circuitry for a primary component and provide a supporting role. Components that are important to safe shutdown are all components not classified as required for hot shutdown.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0 and 6.0

2.4.2.1 Nuclear Safety Capability Systems and Equipment Selection

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

As part of the process of preparing the Safe Shutdown Equipment List (SSEL) and defining equipment and cables for safe shutdown, additional equipment and cables that support the SSEL components were identified (such as interlocked components, normal and alternate electrical power supplies, cascading power supplies). The process included development of a cascading interlock analysis.

This information was populated into a relational type database necessary to analyze for post-fire safe shutdown.

Reference Document

EIR 51-9133191, NSCA, Sections 4.0, 5.0, and 6.0

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 mal-operation of the equipment identified in 2.4.2.1. This evaluation shall consider fire-induced failure modes such as hot shorts (external and internal), open circuits, and shorts to ground, to identify circuits that are required to support the proper operation of components required to achieve the nuclear safety performance criteria, including spurious operation and signals. This will ensure that a comprehensive population of circuitry is evaluated.

2.4.2.2.2 Other Required Circuits. Other circuits that share common power supply and/or common enclosure with circuits required to achieve nuclear safety performance criteria shall be evaluated for their impact on the ability to achieve nuclear safety performance criteria.

(a) Common Power Supply Circuits. Those circuits whose fire induced failure could cause the loss of a power supply required to achieve the nuclear safety performance criteria shall be identified. This situation could occur if the upstream protection device (i.e., breaker or fuse) is not properly coordinated with the downstream protection device.

(b) Common Enclosure Circuits. Those circuits that share enclosures with circuits required to achieve the nuclear safety performance criteria and whose fire-induced failure could cause the loss of the required components shall be identified. The concern is that the effects of a fire can extend outside of the immediate fire area due to fire-induced electrical faults on inadequately protected cables or via inadequately sealed fire area boundaries.

NEI 00-01 Ref	**NEI 00-01 Guidance**
3.3 Safe Shutdown Cable Selection And Location	This section provides industry guidance on 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 paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.2 Nuclear Safety Capability Circuit Analysis

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

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.1.1 Cable Failures	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

The cables necessary to operate and/or maintain the status of each Safe Shutdown component were evaluated by a detailed review of the drawings. Cables that could impact safe shutdown equipment were identified using the component's control schematic, instrument loop, wiring diagram, if available, or the component's electrical elementary wiring diagrams, one-line drawings, or other available wiring diagrams. These drawings were used as a guide to perform a point to point review of the associated connection diagrams. Cables associated with power, control, instrumentation, indication, interlock and any other cable that could impact the component were considered.

Fault analysis during cable identification led to the cable fault codes P, L, O, C, and I as defined in EIR #51-9133191. This made the final compliance analysis bounding. Further analysis determined the effects of a fire induced hot short, open circuit and short to ground during the fire area compliance assessment task. Additional schematic diagrams were reviewed for secondary or interlocked circuits, as necessary, which could impact the operation of components required for safe shutdown.

Reference Document

EIR 51-9133191, NSCA, Sections 2.6, 7.0, and 8.0
CNG-FES-017, NFPA 805 Safe Shutdown Equipment Cable Selection

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	***NEI 00-01 Guidance***
3.3.1.1.2 Cable Failures Affecting Multiple Safe Shutdown Equipment	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	**Comments**
Applicable	None

Alignment Statement

Aligns With Intent

Alignment Basis

For cases where single cables have the potential to impact multiple components, the cable would be listed against each component. For control logic circuits, where multiple components receive signals from common control logic, the control logic was analyzed as a primary component and given a pseudo-component identification.

A pseudo-component is an artificial association of equipment and cables that performs a common function that is combined into a single entity for analysis purposes only and is not an actual plant hardware designation. The pseudo-component was interlocked to the other associated primary components so that the effect of the control logic could be evaluated on an individual component level.

This methodology was used for similar circuit scenarios such as common power supplies. Whereas this approach does not assign the cable to each individual component, the effect on each component due to fire damage is analyzed. This method serves to reduce the duplication of cable data when the same cables are assigned to multiple components.

Pseudo-components and other primary components, whose associated cabling can affect another primary component based on interposing contacts, were identified on the Cable Selection Worksheet of the affected component as an interlocked primary component.

Reference Document

EIR 51-9133191, NSCA, Section 6.0
EIR 51-9177678-000, Definitions Section

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.1.2.1 Electrical Devices	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

Cables were identified and selected using the component's control schematic, electrical elementary diagrams, one-lines, and/or instrument loop diagrams. These drawings were used as a guide while performing a point to point review of the associated connection diagrams.

Electrical isolation devices prevent malfunctions in one section of a circuit from causing unacceptable effects in other portions of the circuit or other circuits (e.g., open contacts, fuses, switches, instrument isolation modules). Devices credited as providing electrical isolation were identified in the circuit analysis for the affected component.

Fault analysis during cable identification led to the P, L, O, C, and I fault codes. All circuits/cables that are electrically connected to the circuit under the analysis are identified up to a credited isolation device including the instrument loops.

Reference Document

EIR 51-9133191, NSCA, Sections 2.6 and 8.3

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.1.3 Screening Out Cables With No Impact	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

Cables necessary to operate and/or maintain the status of each safe shutdown component were identified and evaluated by a detailed review of the component's control schematic, elementary diagrams, one-lines, instrument loop diagrams or other available wiring diagrams. These drawings were used as a guide while performing a point to point review of the associated connection diagrams. Cables associated with outputs from auxiliary contacts to computer points, annunciators or motor heaters were excluded from the cable selection when it was concluded that the cable failure would not impact the primary component or performance of the circuit.

Panel wires that are completely contained within a panel were not explicitly listed as SSD cables. These wires are inherently included in the analysis in the same manner as secondary components.

Reference Document

EIR 51-9133191, NSCA, Sections 7.0 and 8.3

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.1.4 Power Supply to Safe Shutdown Equipment	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

The power cables were selected using the component's control schematic or electrical elementary diagrams, one-lines, or instrument loop diagrams or other available wiring diagrams. During the cable selection process, the supporting power sources and interlocks for each primary component were identified. The cascading power supplies (pseudo-components created for power supply interlocks) and the cascading interlocks all serve to identify required power supplies to ensure safe shutdown components are supplied with electrical power. The relationship between the power source and their load components was documented and their dependency was considered during the Fire Area Assessment phase by reviewing the power source load list report from the database.

Breaker coordination calculation #32-9151404-000, Nine Mile Point Unit 1 - NFPA 805 Coordination Study, reviewed the existing and any new electrical circuits that could impact safe shutdown. This calculation identified fire protection program breaker coordination issues concerning proper coordination between the supply breaker/fuse and the load breaker/fuses for power sources required for hot shutdown. This document meets the nuclear safety capability requirements of NFPA-805 and guidance of NEI 00-01.

Reference Document

EIR 51-9133191, NSCA, Section 8.0
EIR 51-9177678-000, Definitions Section
EIR 32-9151404-000, Nine Mile Point Unit 1 - NFPA 805 Coordination Study

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.1.4.1 Automatic Initiation Logics	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

The Safe Shutdown Analysis takes credit for automatic transfer to an alternate power source if the transfer circuit and power source is not affected by the fire. As an example, the EC system can be initiated either manually or automatically. The RPS instruments and logic that automatically initiate the EC system on high reactor pressure or low-low reactor level have been included in the analysis. Manual initiation of the EC system can be accomplished from either the Control Room or RSP depending on the fire location. AC power is not required to manually initiate DHR via the ECs.

Adverse effects due to fire have been considered for the automatic initiation logic circuits. Fire area compliance assessments demonstrate that safe shutdown capability is not adversely affected by a fire in any plant area that disables automatic functions (including initiation logic).

Reference Document

EIR 51-9133191, NSCA, Section 4.0

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.1.5 Associated Circuits	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

The concern for cabling of the electrical distribution system for primary safe shutdown components involves those breakers that feed associated circuits and may not be fully coordinated with upstream breakers. This involves circuits that are the direct power feed to primary safe shutdown components and/or the power feed to motor control centers that support components or other motor control centers. The concern for these circuits is not the load itself but the upstream power source.

For the NMP1 electrical distribution system, it was assumed that the safe shutdown components and their associated power load were coordinated with their upstream power supplies when identifying cables for all existing and any new safe shutdown components.

For safe shutdown circuits, the cables included the direct power feed from the load center to the component and any cables associated with that component. In addition, coordination was also assumed for any branch circuits related to the safe shutdown component's power source.

Associated circuits are those circuits which are not completely independent of the safe shutdown systems or components. Failure or spurious operation of these circuits could potentially defeat the safe shutdown capability of a safety system. A fire in a given fire area could potentially affect systems and components thought to be independent of that particular fire area.

For the purpose of this analysis, an associated circuit must be associated with both a fire area and a safe shutdown system or component. The associated circuits include circuits which share a common power supply with safe shutdown component and circuits whose spurious operation would adversely affect the safe shutdown capability.

Breaker coordination calculation #32-9151404-000, Nine Mile Point Unit 1 - NFPA 805 Coordination Study, demonstrates the existing coordination status (including any new electrical circuits) for required common power supplies that could impact safe shutdown. This calculation identifies if there are any fire protection program breaker coordination issues concerning proper coordination between the supply breaker/fuse and the load breaker/fuses for power sources required for hot shutdown. This document meets the nuclear safety capability requirements of NEI 00-01 and NFPA-805.

Reference Document

EIR 51-9133191, NSCA, Section 8.5
EIR 32-9151404-000, Nine Mile Point Unit 1 - NFPA 805 Coordination Study

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.1.1.6 Exclusion Analysis	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
Not Applicable	None

Alignment Statement

Not Required

Alignment Basis

Exclusion analysis was not used to demonstrate a lack of potential for any impacts to post-fire safe-shutdown.

Reference Document

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref

3.3.2 Associated Circuit Of Concern Cables

NEI 00-01 Guidance

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*
- Common power source
- Common enclosure.

Each of these cables is classified as an associated circuit of concern cable.

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.

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.

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

Applicable

Comments

None

Alignment Statement

Not Required

Alignment Basis

This section is a generic discussion concerning the separation of Associated Circuit Cables and non-safety circuits of components required for safe shutdown. This information is discussed in more detail in subsequent sub-sections 3.3.3, 3.5.2.4 and 3.5.2.5.

Reference Document

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.3 Methodology for Cable Selection and Location	Refer to Figure 3-4 for a flowchart illustrating the various steps involved in selecting the cables necessary for performing a post-fire safe shutdown analysis. Use the following methodology to define the cables required for safe shutdown including cables that may be circuits of concerns for a post-fire safe shutdown analysis. Criteria for making the determination as to which circuits 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

Not Required

Alignment Basis

This paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 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 diagram ▪ 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

Cable selection was performed to identify all conductors/wires that may be required for a component to perform its safe shutdown function, or whose failure could be adverse to the component's safe shutdown function. The cables were selected by a point-to-point review of the applicable connection diagrams, single line electrical diagrams, elementary wiring diagrams or instrument loop wiring diagrams.

During the initial review, any cable that had a potential to impact the safe shutdown component was identified and associated to that component. Cables identified for each Safe Shutdown component, including any additional reference drawings, were populated in the database during the cable selection process. Cables that are computer inputs or that have adequate isolation are excluded.

Figures 3-2 and 3-3 of NEI 00-01 were used to develop the Safe Shutdown Systems, the systems Paths and Safe Shutdown Equipment List. These lists included electrical distribution equipment identified for circuits whose failure may cause a coordination concern. The power related electrical equipment included upstream power sources up to either offsite power or the emergency power source.

Coordination of power supplies was assumed when assigning cables to the safe shutdown components; however, this may not encompass any new components and circuits being added to the program. Breaker coordination calculation #32-9151404-000, Nine Mile Point Unit 1 - NFPA 805 Coordination Study, demonstrates the existing coordination status for the required common power sources. This calculation identifies if there are any fire protection program breaker coordination issues concerning proper coordination between the supply breaker/fuse and the load breaker/fuses for power sources required for hot shutdown. This document meets the nuclear safety capability requirements of NEI 00-01, Section 3.5.2.4 and Figure 3.5.2-6, and NFPA 805, Section 2.4.2.2.2.

Reference Document

EIR 51-9133191, NSCA, Sections 7.0 and 8.0
EIR 32-9151404-000, Nine Mile Point Unit 1 - NFPA 805 Coordination Study

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.3.2 Identify Interlocked Circuits and Cables that Could Affect Safe 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 With Intent

Alignment Basis

For control logic circuits where multiple components receive signals from a common logic, the control logic was analyzed as a primary component and a pseudo-component identification was created for the control logic. A pseudo-component is an artificial association of equipment and cables that performs a common function into a single entity for analysis purposes. The pseudo-component is not an actual plant hardware designation.

The pseudo-component was interlocked to the other associated primary components so that the effect of the control logic could be evaluated on an individual component level. This methodology was used for similar circuit scenarios such as common power supplies. Whereas this approach does not assign the cable to each individual component, the effect on each component due to fire damage is analyzed. This method serves to reduce the duplication of cable data when the same cables are assigned to multiple components.

Pseudo-components and other primary components whose associated cabling can affect another primary component based on interposing contacts were identified on the Cable Selection Worksheet of the affected component as an interlocked primary component. This meets the intent of the guidance.

Reference Document

EIR 51-9133191, NSCA, Sections 5.4 and 6.0
EIR 51-9177678-000, Definitions Section

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.3.3.3 Assign Cables to 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

Cable selection for Safe Shutdown components was performed by identifying all cables required for a component to perform its safe shutdown function. These cables were selected by a point-to-point review using the component's elementary diagram. The cables were selected in accordance with NEI 00-01, Section 3.3.1 and then entered into the database. For cases where cables affected multiple components, either the cable was assigned to each component or a pseudo-component was used with the cables assigned to the pseudo-component instead of the primary component.

In addition to the cables, any component interlocks were identified to investigate their impact on the operation of the safe shutdown component. The relationship between these interlocks and the primary component were documented and their dependency was considered during the Fire Area Compliance Assessment.

Coordination of power supplies is addressed in Section 3.5.2.4 of this document.

Reference Document

EIR 51-9133191, NSCA, Section 8.0
EIR 51-9177678-000, Definitions Section

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5 Circuit Analysis and Evaluation	This section on circuit analysis provides information on the potential impact of fire on circuits used to monitor, control and power 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

Not Required

Alignment Basis

This paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.1 Criteria/Assumptions	Apply the following criteria/assumptions when performing fire-induced circuit failure evaluations. 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 paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 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.

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.

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.)

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

During the cable selection process, a circuit fault analysis for each component cable was initially performed to determine the effects of a fire-induced hot short, open circuit and short to *ground, as applicable. Per the NEI 00-01 guidance, all combinations of circuit failures (hot shorts, open circuit, and short-to ground) on each conductor for each unprotected safe* shutdown cable were considered. Further analysis was performed, as required, for secondary or interlocked circuits.

The circuit failures were evaluated to determine the potential impact of a fire on the safe shutdown equipment (including the path) that is associated with that cable/conductor. In some cases, the cables and components had adequate separation from their redundant circuits and components as required by the regulations and were not required to be analyzed.

It was assumed that fire damage results in an unusable cable that cannot be considered functional with regard to ensuring proper circuit operation. The insulation and external jacket material of electrical cables are susceptible to fire damage. Damage may assume several forms including deformation, loss of structure, cracking, and ignition. The relationship between exposure of electrical cable insulation to fire conditions, the failure mode, and time to failure may vary with the configuration and cable type. To accommodate these uncertainties in a consistent and conservative manner, the circuit analysis assumes that the functional integrity of electrical cables is lost when cables are exposed to a fire, except where protected by a fire rated barrier.

The types of circuit failures considered for this analysis are those identified in NEI 00-01, Appendix B, Table B.1.0, "Types of Fire-Induced Circuit Failures Required to Be Considered." Consistent with NEI 00-01, hot shorts were considered to be either internal cable wire-to-wire shorts or external cable-to-cable shorts. No credit was taken for physical cable attributes (armored, thermo-set, etc.) preventing cable-to-cable hot shorts.

Reference Document

EIR 51-9133191, NSCA, Section 8.2
HNP RAI 3-16, NRC Request for Addition Information (ML092170715)

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 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. o 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

An MSO Expert panel reviewed the generic list of scenarios listed in NEI-00-01 Appendix G, screening out any scenario that was a combination of five or more spurious operations. There were no screening criteria based on number, timing, or type of circuit failures. The result of the review was a list of MSO scenarios, both generic and site specific, that were included in the Fire Area Analysis for further evaluation.

Reference Document

EIR 51-9133191, NSCA, Section 8.1
ONS RAI 3-38, NRC Request for Additional Information dated July 30, 2010 (ML102110394)
Technical Report on Identification & Classification of the NMP-1 MSO Scenarios Using an Expert Panel - Review of New Scenarios, Rev.1

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.1.3 Circuit Contact Position	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

The analysis assumes that the circuit contacts are positioned (i.e., open or closed) consistent in the normal mode/position of the safe shutdown equipment as shown on the schematic drawings or defined by procedure. The fire damage impact on the position of the safe shutdown equipment was considered for each shutdown scenario.

Reference Document

EIR 51-9133191, NSCA, Section 8.6.1

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2 Types Of Circuit Failures	Appendix R requires that nuclear power plants must be designed to prevent exposure fires from defeating the ability to achieve and maintain post-fire safe shutdown. Fire damage to circuits that provide control and power to equipment 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 paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.1 Circuit Failures Due to an Open Circuit	This section provides guidance for addressing the effects of an open circuit for 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.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Open circuits are analyzed as referenced in the NEI 00-01 guidance, Section 3.5.2.1 and Figure 3.5.2-1. An open circuit is a condition experienced when an individual conductor within a cable loses electrical continuity due to a fire induced break. This could cause the loss of power from de-energizing the circuit or the ability to control affected components, or on energized equipment could cause a change of position of the component. In addition, as stated in the guidance, an open circuit on a high voltage ammeter CT circuit may result in secondary damage to that circuit.

The Nuclear Safety Capability Assessment (NSCA) assumed that fire damage results in an unusable cable that cannot be considered functional with regard to ensuring proper circuit operation. The insulation and external jacket material of electrical cables are susceptible to fire damage. Damage may assume several forms including deformation, loss of structure, cracking, and ignition. The relationship between exposure of electrical cable insulation to fire conditions, the failure mode, and time to failure may vary with the configuration and cable type. To accommodate these uncertainties in a consistent and conservative manner, the circuit analysis assumes that the functional integrity of electrical cables was lost when cables are exposed to a fire, except where protected by a fire rated barrier.

Other associated circuit concerns are related to open secondary circuits and 4 KV bus CT that may result in high currents producing an additional fire at the transformer location. This associated circuit concern is related to CT where an open secondary circuit may develop high voltages within a transformer potentially resulting in a secondary fire at the transformer location. This issue is evaluated in Fire Protection Engineering Evaluation FPEE-1-04-002, Rev. 0, Fire Effects on CTs and Instrument Sensing Lines and The Plant Safe Shutdown Capability. The evaluation concludes that, for all CTs in use at NMP1, an open transformer secondary will not develop voltages that are high enough to threaten the Safe Shutdown capability.

Reference Document

EIR 51-9133191, NSCA, Sections 8.2 and 8.5
Fire Protection Engineering Evaluation FPEE-1-04-002, Rev. 0, Fire Effects on CTs
ONS RAI 3-48, NRC Request for Additional Information dated July 30, 2010 (ML102110394)
HNP RAI 3-17, NRC Request for Additional Information dated August 6, 2009 (ML092170715)

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.2 Circuit Failures Due to a Short-to-Ground	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.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The methodology assumes multiple fire induced failures including short-to-ground. The short-to-ground issue incorporates circuit failures for both ungrounded and grounded circuits. Postulated cable and component failures were identified utilizing the techniques referenced in NEI 00-01 Figure 3.5.2-2 and Figure 3.5.2-3. The safe shutdown analysis may exclude certain cables if their postulated fire induced faults have no adverse effect on the component.

A short-to-ground fault for grounded circuits could cause the tripping of a circuit thereby causing a loss of power to the control circuit. For certain cases of energized components, a loss of control power may result in a loss of power to relays and other devices interlocked with the device.

Unless otherwise justified by circuit analysis, short- to-ground for ungrounded circuits are treated the same as short- to-ground for grounded circuits, and are postulated to result in a loss of motive power or control power. This is consistent with NFPA 805 Appendix B, Section B.3.2.g which states: "For ease of analysis when analyzing an ungrounded DC circuit for the effects of a short-to-ground, it should be assumed that an existing ground fault from the same power source is present."

Reference Document

EIR 51-9133191, NSCA, Sections 8.1 and 8.2
NFPA 805, Appendix B, Section B.3.2.g

2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.3 Circuit Failures Due to a Hot Short	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.
Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

A hot short is a condition experienced when an energized individual conductor of the same or different cable comes into contact with another conductor of the same or different cable resulting in electrical continuity between the conductors. The potential effect is that the energized conductor becomes an undesired source of power for the circuit being analyzed. Hot shorts were considered to be either internal conductors of the same cable, identified as internal shorts, or shorts between conductors of different cables, identified as external shorts. The potential of circuit failures due to hot shorts can cause components to operate or cause them to fail to operate in an undesired manner.

The Nuclear Safety Capability Assessment (NSCA) assumed that fire damage results in an unusable cable that cannot be considered functional with regard to ensuring proper circuit operation. The insulation and external jacket material of electrical cables are susceptible to fire damage. Damage may assume several forms including deformation, loss of structure, cracking, and ignition. The relationship between exposure of electrical cable insulation to fire conditions, the failure mode, and time to failure may vary with the configuration and cable type. *To accommodate these uncertainties in a consistent and conservative manner, the circuit analysis assumes that the functional integrity of electrical cables is lost when cables* are exposed to a fire, except where protected by a fire rated barrier.

Consistent with NEI 00-01, hot shorts were considered to be either internal cable wire-to-wire shorts or cable-to-cable (external) shorts. No credit was taken for physical cable attributes (armored, thermo-set, etc.) preventing cable-to-cable hot shorts.

For cable failures due to hot shorts on grounded or ungrounded circuits, the methodology initially assumes the hot short would have sufficient potential to cause a spurious operation of the component. Two types of cable hot short conditions are considered to be of sufficiently low likelihood that they are not assumed credible, except for analysis involving high/low pressure interface components. These hot shorts are 3-phase AC power circuit cable-to-cable proper phase sequence faults and 2-wire ungrounded DC circuit cable-to-cable proper polarity faults.

Instrument circuits that operate at low signal levels (4-20 mA, 0-1 V, 1-5 V, etc.) and are enclosed in a grounded metal shield are not considered to be susceptible to hot shorts from other adjacent instrument circuits external to the shield. External circuits are assumed to short to ground via the shield and do not have the potential of creating a signal of proper polarity and amplitude to simulate a valid instrument signal.

Reference Document

EIR 51-9133191, NSCA, Section 8.2
ONS RAI 3-41, NRC Request for Additional Information dated July 30, 2010 (ML102110394)

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 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

This task involved identifying the routing and location for all raceways and endpoints for cables associated with safe shutdown equipment. The cable routes and their endpoint location were populated into the database. The database is a relational database that contains all the required information for safe shutdown cable routing and endpoint information. The original cable routing and cable endpoint data was provided from the NMP1 cable raceway database (TRAK2000).

Comments

None

Reference Document

EIR 51-9133191, NSCA, Sections 2.1 and 8.4
TRAK2000, Revision 6.01

2.4.2.3 Nuclear Safety Equipment and Cable Location

NEI 00-01 Ref	**NEI 00-01 Guidance**
3.3.3.5 Identify 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	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Fire area locations were identified for each cable raceway and cable endpoint by obtaining the location coordinates from applicable cable tray, conduit or equipment layout arrangement drawings or by field walkdown, if necessary. The fire area locations were identified by comparing the cable tray/conduit arrangement drawings, equipment arrangement drawings, or field walkdown data with Fire Area Floor Plans drawings. This correlation between cable raceway locations and Fire Areas and Rooms was populated into the database to produce computer generated reports. The reports contained the cable related raceway information required to prepare the Fire Area Analysis.

Reference Document

EIR 51-9133191, NSCA, Sections 5.4 and 8.4
Fire Area Floor Plans Drawings B40141C through B40148C

2.4.2.3 Nuclear Safety Equipment and Cable Location

NEI 00-01 Ref	NEI 00-01 Guidance

3.5.2.4 Circuit Failures Due to Inadequate Circuit Coordination	The evaluation of 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.
	
	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.
	  ▪ 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

The NMP1 Fire Area Assessments are performed to support transition to a performance based fire protection licensing basis. While performing the primary component circuit analysis for safe shutdown components, it was assumed that electrical coordination exists for all power supplies for each level of electrical power.

Breaker coordination calculation #32-9151404-000, Nine Mile Point Unit 1 - NFPA 805 Coordination Study, demonstrates the existing coordination status for the required common power sources. This calculation identifies any fire protection program breaker coordination issues concerning proper coordination between the supply breaker/fuse and the load breaker/fuses for power sources required for hot shutdown. This document meets the nuclear safety capability requirements of NEI 00-01, Section 3.5.2.4, Figure 3.5.2-6 and NFPA 805, Section 2.4.2.2.2. Any identified issues have been addressed in the NSCA.

Other potential coordination concerns involve associated non-safe shutdown circuits that are not independent of safe shutdown circuits that could also potentially defeat the functions of safe shutdown circuits if not properly protected. These circuits must be associated with both a fire area and a safe shutdown system or component to warrant consideration. These associated circuits are divided into three categories.

- Circuits that share a common power supply with safe shutdown circuits.
- Circuits that share a common enclosure with safe shutdown circuits.
- Circuits for components the spurious operation of which would adversely affect the shutdown process.

The associated circuits are defined in the NMP1 Coordination study which reviews the 4.16 kV, 600 VAC, 480 VAC, 208/120 VAC and 125 VDC power supplies credited for post-fire shutdown.

Proper circuit coordination for power supplies was reviewed, analyzed and addressed in EIR 51-9133191, NSCA.

Reference Document

EIR 51-9133191, NSCA, Section 8.5
HNP RAI 3-18 and RAI 3-19, NRC Request for Additional Information dated August 6, 2009 (ML092170715)
EIR 32-9151404-000, Nine Mile Point Unit 1 - NFPA 805 Coordination Study

2.4.2.3 Nuclear Safety Equipment and Cable Location

NEI 00-01 Ref	NEI 00-01 Guidance
3.5.2.5 Circuit Failures Due to Common Enclosure Concerns	The common enclosure 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

Circuit failures due to common enclosure concerns are addressed by breaker coordination calculation #32-9151404-000, Nine Mile Point Unit 1 - NFPA 805 Coordination Study. This calculation demonstrates the existing coordination status for electrical circuits that could impact safe shutdown concerning proper coordination between the supply breaker/fuse and the load breaker/fuses for power sources required for hot shutdown. This document meets the nuclear safety capability requirements of NEI 00-01, Section 3.5.2.4, Figure 3.5.2-6 and NFPA 805, Section 2.4.2.2.2.

Reference Document

EIR 51-9133191, NSCA, Section 8.5
HNP RAI 3-18 *and RAI 3-19, NRC Request for Additional Information dated August 6, 2009* (ML092170715)
EIR 32-9151404-000, Nine Mile Point Unit 1 - NFPA 805 Coordination Study

2.4.2.4 Fire Area Assessment

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

NEI 00-01 Ref	NEI 00-01 Guidance
3.4 Fire Area Assessment and Compliance 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 effects 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 effects 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 paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.4 Fire Area Assessment

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

Applicability	Comments
Applicable	None

Alignment Statement

Not Required

Alignment Basis

This paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	**NEI 00-01 Guidance**
3.4.1.1 Assume a Single Fire	Assume only one fire in any single fire area at a time.

Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Only one fire is assumed to occur in any single fire area at a time.

Reference Document

EIR 51-9133191, NSCA, Section 9.0

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.2 Fire Affects All Unprotected Cables and Equipment	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

For a conservative approach which bounds the exposure fire required by the regulations, the analysis assumes a fully involved fire and that all equipment and unprotected cabling within a given fire area are damaged by the fire.

Reference Document

EIR 51-9133191, NSCA, Section 9.0

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.3 Address all Cable and Equipment Impacts Affecting the Required Safe Shutdown Path	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

Fire Area Assessments were performed on a Fire Area basis in order to ensure compliance in accordance with the safe shutdown requirements of NFPA 805. The Safe Shutdown System and Component drawings were analyzed for each Fire Area to ensure that a success path is available based upon the postulated equipment and/or cable losses in the area. The potentially affected equipment and cables in each Fire Area were reviewed and impacts on safe shutdown success paths analyzed.

The route and location of all safe shutdown cables were loaded into the safe shutdown database. This data was used to generate Fire Area Component Impact Reports, which identified affected systems and components on a Fire Area basis. The Fire Area Component Impact Reports were used as a means to determine the least impacted safe shutdown path for each fire area.

The Fire Area Analysis methodology assumed multiple fire-induced failures and multiple spurious actuations, based on the cables and components present in the fire area of concern. All postulated cable and component failures were identified and a resolution provided at the component level.

The least impacted safe shutdown success path was analyzed so that mitigating strategies could be developed and documented in the Fire Area Assessment. A success path determination for all safe shutdown functions was performed. Generally the path with the least amount of failures was recovered to demonstrate a success path for safe shutdown. Support systems were reviewed in order to assess the impact on the systems being supported. The credited safe shutdown success path was documented in the fire area compliance assessment.

Reference Document

EIR 51-9133191, NSCA, Section 9.0

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.4 Classify Each Cable/Component	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

Not Required

Alignment Basis

Using the criteria from Appendix H to classify each impacted cable/component as either a required or important to SSD cable/component was not required to support the transition to NFPA 805. *Therefore, cables and components were not classified as "required for safe shutdown" or "important for safe shutdown." The safe shutdown flow paths identify the primary* components that are required to meet the safe shutdown performance goals. The safe shutdown cables/components were compiled based on each system's performance and safe shutdown function. These components establish the primary safe shutdown flowpath for system operation. Also included in the safe shutdown flow paths are those cables/components whose spurious operation could impact safe shutdown system operability. Systems, components, and cables identified as necessary for the operation of the safe shutdown system under review are included in the safe shutdown equipment and cables lists and are designated with the same shutdown path as the primary safe shutdown system. The components may involve branch flow paths that must be isolated and remain isolated to assure that flow will not be diverted from the primary flow path. The list of primary components may also include selected mechanical components required to support safe shutdown.

Reference Document

EIR 51-9133191, NSCA, Section 9.0

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.5 Manual Actions	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

Manual actions performed as prescribed in procedures or otherwise are documented in each Fire Area Assessment (FAA). Included in each FAA was the identification of any required operator actions outside the Main Control Room. The operator actions are those directed by operating procedures, repair procedures, or otherwise identified as necessary during the course of the individual FAA. Actions performed at locations other than primary control stations are identified as recovery actions requiring further review as part of the fire risk evaluations. These actions are identified in the Report of Manual Actions and Report of Procedure Directed Manual Actions included in each FAA.

Reference Document

EIR 51-9133191, NSCA, Section 9.0

2.4.2.4 Fire Area Assessment

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Repairs which are relied upon to achieve and maintain cold shutdown will be performed as required. Repairs are directed by plant procedures.

Reference Document

EIR 51-9133191, NSCA, Sections 2.1, 8.5, and 9.0

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	**NEI 00-01 Guidance**
3.4.1.7 Appendix R Compliance Criteria	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); ▪ 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 With Intent

Alignment Basis

Each NMP1 fire area containing safe shutdown equipment or cables was reviewed in a deterministic fashion for the ability to achieve post-fire safe shutdown. The affected shutdown related cables and components in each area were identified and the resultant information used to determine the preferred shutdown path to achieve safe shutdown.

The credited safe shutdown success paths were analyzed and mitigating strategies (procedural actions, repair actions or modification) were developed and documented in the Safe Shutdown Analysis, fire area compliance assessments. The results of the assessments confirm that in the event of a postulated exposure fire, the safe shutdown capability of NMP1 will be maintained.

The non-safe shutdown circuits which are not completely independent of safe shutdown circuits are associated circuits. These associated circuits are those circuits that could adversely affect the safe shutdown capability or components. These circuits would be associated with a safe shutdown system or component and analyzed the same as other safe shutdown circuits affected within that fire area.

Reference Document

EIR 51-9133191, NSCA, Sections 5.1, 5.2, and 5.3

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.1.8 Alternate/Backup Equipment Selection	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	Consideration of Multiple Spurious Operations is addressed in the NSCA (EIR #51-9133191, Section 8.6).

Alignment Statement

Aligns

Alignment Basis

Component selection was performed for all fire areas in order to populate the database with equipment information required to be analyzed against the requirements of 10CFR50, Appendix R. The components selected are documented in the Safe Shutdown Equipment List (SSEL).

The objective of the SSEL is to provide a list of analyzed components that are utilized in the post-fire safe shutdown analysis to ensure that (1) one success path (structures, systems, and components) necessary to achieve safe shutdown is free of fire damage without crediting plant or system repair capabilities and (2) one success path (structures, systems, and components) necessary to achieve cold shutdown within 72 hours is free of fire damage, or available within 72 hours after crediting plant or system repair capabilities.

The current SSEL was reviewed against the criteria outlined in NEI 00-01 and considered where additional equipment may need to be included to address multiple spurious operation concerns or other separation concerns.

Consideration of component spurious actuation is not limited to the licensing basis criteria documented in UFSAR Appendix 10B, Section 5.9.4. As part of the transition to a NFPA-805 licensing basis, the criteria used in evaluating spurious actuation of components are those identified in NEI 00-01, Section 4, Identification and Treatment of Multiple Spurious Operations, which envelopes the plant licensing basis as discussed in the UFSAR. MSO component combinations were included in the assessments.

The number of potential spuriously operating valves in a line was not limited by number. The NSCA incorporates equipment identified during the review and the cable selection phase by providing an updated SSEL Report of the safe shutdown primary components and the Safe Shutdown Success Paths. In addition, the NSCA supports incorporation of secondary components into the database that were modeled as a result of the primary component selections.

A success path determination for all safe shutdown functions was performed. Generally the path with the least amount of failures was utilized to demonstrate a success path for safe shutdown. Support systems were reviewed in order to assess the impact on the systems being supported.

Primary and secondary shutdown methodologies have been developed. The primary method employs the use of Emergency Condensers. The secondary method employs the use of ERVs and ECCS equipment. This approach provides for versatility by employing diverse equipment resulting in four potential shutdown success paths.

Reference Document

EIR 51-9133191-000, NSCA, Section 8.6
HNP RAI 3-14, NRC Request for Additional Information dated August 6, 2009 (ML092170715)
Technical Report on Identification & Classification of the NMP-1 MSO Scenarios Using an Expert Panel - Review of New Scenarios, Rev.1
UFSAR Appendix 10B, Section 5.9.4

2.4.2.4 Fire Area Assessment

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

Applicability	Comments
Applicable	Instrument tubing failure due to a fire is addressed in the NSCA (EIR #51-9133191, Section 1.14).

Alignment Statement

Aligns

Alignment Basis

Instrument sensing lines for level, pressure, flow, etc. that are exposed to a fire are considered to have the potential of causing erratic or unreliable signals or indication, unless a fire hazards analysis demonstrates that this failure is not credible. Fire damage to instrument sensing lines can be as detrimental to the instruments as fire damage is to safe shutdown cables and components. Even though the integrity of the tubing is expected to withstand the fire, the accuracy of the instrument may not be reflected correctly due to the heating of the fluid.

The instrument sensing lines route locations are developed and inputted as design input to the analysis. The input consisted of a list of instrument sensing lines (located outside Containment) including the fire areas and associated routing locations through the plant. This information was entered into the database as cables using fictitious cable numbers, including the route and endpoint identifications.

The analysis treated the tubing like cables and associated it with the instrument. The sensing lines are subject to the same compliance issues and similar analytical techniques as safe shutdown cables. Sensing lines of instruments required for safe shutdown are included within the scope of a fire area assessment. In this manner, the sensing lines are included for consideration along with cables when performing the fire area assessments. If instruments were impacted by the fire, then alternate instruments, not impacted by the fire, would be relied upon for safe shutdown.

Instrument sensing lines were reviewed for susceptibility to physical fire damage that may cause a loss of inventory. Sensing lines for SSEL components are constructed of either stainless steel or carbon steel. Consequently, they are not susceptible to physical damage as the result of a postulated fire.

Reference Document

EIR 51-9133191, NSCA, Sections 5.4 and 8.4
HNP RAI 3-15, NRC Request for Additional Information dated August 6, 2009 (ML092170715)

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.2 Methodology for Fire Area Assessment	Refer to NEI 00-01 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 paragraph provides introductory information and contains no specific requirements. Discussion is provided in subsequent sub-sections.

Reference Document

2.4.2.4 Fire Area Assessment

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The information needed to support the Post Fire Safe Shutdown Analysis is maintained in a safe shutdown database which contains the safe shutdown systems, components, cables, and their associated fire area location. This information is available in report format and can be sorted by fire area, system, train, component, cable, safe shutdown path, or various combinations of each. These reports are used to assess potential damage due to fire in each area of the plant. The database reports provide the same information identified on Attachment 5 of NEI 00-01.

Reference Document

EIR 51-9133191, NSCA, Sections 5.4 and 9.0

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	***NEI 00-01 Guidance***
3.4.2.2 Determine the Least Impacted Shutdown Path	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

The Fire Area Compliance Assessments demonstrate the ability to achieve safe shutdown by ensuring at least one safe shutdown success path is available to accomplish the performance goals identified in NEI 00-01. NEI 00-01, Section 3 was used as guidance in performing the assessments. The elements of the assessments performed for NMP1 reflect the NEI 00-01 guidance as discussed in the following.

Fire Area Assessments were performed on a Fire Area basis in order to ensure compliance in accordance with the requirements of 10 CFR 50, Appendix R. The safe shutdown database reports provide the potentially affected equipment and cables in each Fire Area, which were analyzed for impacts on safe shutdown success paths.

The Safe Shutdown Equipment List (SSEL) contains equipment data such as the equipment type, description, safe shutdown path, drawing reference, fire area, fire zone, and room location. Other equipment information would include normal position, hot shutdown position, cold shutdown position, failed air position, failed electrical position, high/low pressure interface concern, and spurious operation concern.

The route and location of all Appendix R cables (located by Fire Area) was used to generate the Fire Area Component Impact Reports, which identified affected systems and components on a Fire Area basis. The Fire Area Component Impact Reports were used as a means to determine the least impacted safe shutdown path for each fire area.

The cable selection task involved identifying all cables associated with the control and operation of a safe shutdown component. These cables were analyzed to determine the impact of fire induced cable failure on the selected equipment. A circuit analysis was performed as part of the scope for selected cables/components, as required, in order to demonstrate that the cable is or is not required so that the analyzed component can be credited to perform its required function for the safe shutdown path.

The Fire Area Analysis methodology identified fire-induced component and cable failures and spurious actuations, based on the cables and components present in the fire area of concern. All postulated cable and component failures were assessed and a resolution provided at the component level.

Once the above was complete, the least impacted safe shutdown success path was identified so that mitigating strategies could be developed and documented in the Fire Area Assessment.

Reference Document

EIR 51-9133191, NSCA, Sections 5.1, 5.2, 5.3, and 5.4

2.4.2.4 Fire Area Assessment

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The Safe Shutdown Equipment List was developed based on system requirements and other plant impacts. During the identification of the Safe Shutdown Equipment List for component cables, a circuit fault analysis for each component's cables was performed to determine the effects of a fire-induced hot short, open circuit and short-to-ground. The circuits associated with the components operation and whose failure could affect the components operation was considered as required. The fire area analysis assumed multiple fire-induced failures and multiple spurious actuations (MSOs), based on the cables and components present in the fire area of concern. The cable and component failures were evaluated and a resolution and disposition was provided for component and cable impacted in that fire area.

In addition, spurious operating equipment concerns are addressed in the "MSO Expert Panel Report," which consists of the multiple spurious operation review. The purpose of this review is to document the potential Multiple Spurious Operation combinations.

The results of this activity identifies equipment, whose fire-induced spurious operation could result in consequences that may be adverse to both the Fire PRA risk models and meeting the nuclear safety performance criteria of NFPA 805. The equipment identified in this task that could affect the Fire PRA will be integrated into the Fire PRA Equipment list.

Reference Document

EIR 51-9133191, NSCA, Sections 5.0, 8.0, and 9.0
ONS RAI 3-41 and RAI 3-43, NRC Request for Additional Information dated July 30, 2000 (ML102110394)
Technical Report on Identification & Classification of the NMP-1 MSO Scenarios Using an Expert Panel - Review of New Scenarios, Rev.1

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref

3.4.2.4 Develop a Compliance Strategy or Disposition

NEI 00-01 Guidance

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

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

The safe shutdown analysis provides a compliance strategy and various deterministic methods used for mitigating the effects of circuit failures. Potential impacts to safe shutdown were addressed by using the path least impacted by the fire to assure at least one success path for safe shutdown. This was accomplished by using a combination of the strategies listed in the guidance and taking credit for any existing features whenever possible.

Circuit failures having the potential to adversely impact the shutdown process were identified as Open Items to be transitioned to the Fire Risk Evaluations.

Reference Document

EIR 51-9133191, NSCA, Section 9.0

2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Guidance
3.4.2.5 Document the Compliance Strategy or Disposition	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	

Alignment Statement

Aligns

Alignment Basis

For the Safe Shutdown Analysis, success paths were developed and analyzed for each component impacted by a fire in the subject area using disposition codes that represent consistent standardized compliance statements. The compliance statement reflects the credited fire protection features, analysis of credible cable failures, and serves as the basis for achieving safe shutdown conditions for the analyzed fire areas. The disposition codes (i.e., resolution of component hits) and associated statements were entered into the safe shutdown database. The following is a sample of generic disposition codes used for the Fire Area Compliance Assessment:

- Cable protected by rated fire barrier.
- Failure of cable may result in loss of power/control of component.
- Failure of cable may result in loss of indication or erroneous indication.
- Component fails in desired SSD position/mode.
- Series isolation valve(s) available and can be closed.
- Capability to close valve is available from the MCR.
- Component remains in desired SSD position.
- Valve can spuriously open. Series isolation valve remains closed

Reference Document

EIR 51-9133191, NSCA, Section 9.0

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

298 Pages Attached

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	**REACTOR BUILDING EAST EL 198-0 THRU EL 340-0**

Fire Zone	Fire Zone Description
FBZR237N	REACTOR BUILDING EL 237-0 COL N-Q, ROW 8-9
FBZR261N	REACTOR BUILDING EL 261-0 COL N-Q, ROW 8-9
FBZR281N	REACTOR BUILDING EL 281-0 COL M-Q, ROW 6-7
FBZR281S	REACTOR BUILDING EL 281-0 COL K-L, ROW 7-8
FBZR298N	REACTOR BUILDING EL 298-0 COL N-Q, ROW 7.5-8.5
FBZR298S	REACTOR BUILDING EL 298-0 COL K-L, ROW 7-8
FBZR318N	REACTOR BUILDING EL 318-0 COL M-Q, ROW 6-7
FBZR318S	REACTOR BUILDING EL 318-0 COL K-M, ROW 6-7
FBZR340N	REACTOR BUILDING EL 340-0 COL M-Q, ROW 6-7
FBZR340S	REACTOR BUILDING EL 340-0 COL L-N, ROW 7-8
R1A	CTS PUMP ROOM AND GENERAL FLOOR AREA EAST EL 198-0 & 237-0
R1C	ACCESS STAIRWELL SOUTHEAST EL 237-0 & 261-0
R1D	CS PUMP ROOM AND PROTECTIVE CLOTHING CHANGE AREA EL 198-0 & 237-0
R2A	GENERAL FLOOR AREA EAST EL 261-0
R3A	GENERAL FLOOR AREA EAST EL 281-0
R4A	GENERAL FLOOR AREA EAST EL 298-0
R4C	EMERGENCY CONDENSER ISOLATION VALVE ROOM EL 298-0
R5A	GENERAL FLOOR AREA EAST EL 318-0
R6A	GENERAL FLOOR AREA EAST EL 340-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	Path C is credited to achieve HSD and CSD. Path C is accomplished via the use of Train 11 Systems, ERV's, CS, CTS and CTSRW systems. Path C control Room Ventilation is credited to support shutdown	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control is accomplished by ensuring the Path C ERVs are functional and Path D ERVs maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions.	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	**REACTOR BUILDING EAST EL 198-0 THRU EL 340-0**

(d) Decay Heat Removal Function	Path C is credited to achieve HSD and CSD. Path C is accomplished via the use of Train 11 Systems. The ERV's are opened, CS is used to flood the vessel, CTS and CTSRW are used to remove decay heat from the Torus.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Torus temperature o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

(f) Vital Auxiliaries	1) AC and DC Distribution System The Train 11 emergency diesel generator is credited for supplying power for the shutdown process when AC power is required to operate shutdown equipment. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows: Path C – Train 11 positive pressure flow path	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	**REACTOR BUILDING EAST EL 198-0 THRU EL 340-0**

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-00-004 Rev. 0

Eng Eval Summary: FPEE 0-00-004 Rev. 0 - Engineering Evaluation of Alternate Fire Protection Systems for the Nine Mile Point Halon Protected Areas
Evaluation conducted for the existing Halon protected areas to determine if alternate types of fixed fire suppression systems can be utilized to replace Halon 1301 extinguishing systems. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-04-002 Rev. 0

Eng Eval Summary: FPEE 1-04-002 Rev. 0 - Evaluation of Fire Effects on Current Transformers and Instrument Sensing Lines and The Plant Safe Shutdown Capability
The analysis provided in this FPEE demonstrates that a postulated fire at NMP1 and the resultant fire effects on Current Transformers and instrument tubing lines will not affect the existing assumptions in the NMP1's present Appendix R Safe Shutdown Analysis. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-12-001 Rev. 0

Eng Eval Summary: This Engineering Evaluation provides analysis of the existing NMP1 approved exemptions from 10CFR50, Appendix R and input to NFPA-805 Licensing Amendment Request (LAR) submittal with respect to previously-approved exemptions to Appendix R to 10 CFR Part 50. This document provides the evaluation, analysis and basis to demonstrate that current plant configuration and NMP1 Fire Protection Program complies with the requirements of 10CFR50.48(c), NFPA 805. The evaluation documents the similar requirements of 10CFR50, Appendix R and NFPA 805. This evaluation demonstrates that the current plant configuration and fire protection features comply with the 10CFR50.48(c) licensing basis.

Engineering Eval Id: FPEE 1-85-003 Rev. 0

Eng Eval Summary: This evaluation documents the acceptibility of negligible intervening combustibles between Fire Areas 1 and 2. This evaluation reflects the CENG interpretation of the NRC position on the acceptability of negligible intervening combustibles.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 1

Fire Area Description REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

Engineering Eval Id: FPEE 1-89-003 Rev. 0

Eng Eval Summary: FPEE 1-89-003 Rev. 0 - Existing 12" x 16" Duct Passing Through Reactor Building Airlock
Evaluation justifies lack of fire damper in HVAC duct penetrating 2-hour rated Reactor Building stair tower enclosure. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-92-001 Rev. 0

Eng Eval Summary: FPEE 1-92-001 Rev. 0 - Penetration Seal Enlarged Boot Detail
Evaluation determines the acceptability of boot seal designs for pipe sizes larger than those qualified by test. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
FBZR237N	Detection	DA-4076W	Y	N	N	Y	N	
	Suppression	WP-4076	Y	N	N	Y	N	
FBZR261N	Detection	DA-4116E	Y	N	N	Y	N	
	Detection	DA-4116W	Y	N	N	Y	N	
	Suppression	WP-4116	Y	N	N	Y	N	
	Suppression	WP-4116W	Y	N	N	Y	N	
FBZR281N	Detection	D-4156	Y	N	N	Y	N	
	Detection	DA-4116W	Y	N	N	Y	N	
	Suppression	WP-4116	Y	N	N	Y	N	
FBZR281S	Detection	DA-4116W	Y	N	N	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
	Suppression	WP-4116	Y	N	N	Y	N	
FBZR298N	Detection	DA-4237	Y	N	N	Y	N	
	Suppression	WP-4237	Y	N	N	Y	N	
FBZR298S	Detection	DA-4237	Y	N	N	Y	N	
	Suppression	WP-4237	Y	N	N	Y	N	
FBZR318N	Detection	DA-4237	Y	N	N	Y	N	
	Suppression	WP-4237	Y	N	N	Y	N	
FBZR318S	Detection	DA-4237	Y	N	N	Y	N	
	Suppression	WP-4237	Y	N	N	Y	N	
FBZR340N	Detection	D-4267	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
FBZR340S	Detection	D-4267	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
R1A	Detection	D-4026	N	N	N	Y	N	
	Detection	DA-4076E	N	N	N	Y	N	
	Suppression	WP-4076	N	N	N	N	Y	
R1C	Detection	DA-4076E	N	N	N	N	Y	
	Detection	DA-4116E	N	N	N	N	Y	
	Detection	DA-4116PL	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
R1D	Detection	D-4046	N	N	N	N	Y	
	Detection	DA-4076E	N	N	N	N	Y	
	Suppression	WP-4076	N	N	N	N	Y	
R2A	Detection	DA-4116E	N	N	N	Y	N	
	Suppression	WP-4116	N	N	N	N	Y	
R3A	Detection	D-4156	N	N	N	Y	N	
	Detection	D-4166	N	N	N	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
1 | REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
	Suppression	None	N	N	N	N	N	
R4A	Detection	D-4197	N	N	N	N	Y	
	Detection	D-4207	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
R4C	Detection	DX-4217A/B	N	N	N	N	Y	
	Suppression	H-4217	N	N	N	N	Y	
R5A	Detection	DA-4237	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
R6A	Detection	D-4267	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	

Legend: Table Field: "Required ?"

S - Required for Chapter 4 Separation Criteria
L - Required for NRC Approved Licensing Action
E - Required for Existing Engineering Equivalency Evaluation
R - Required for Risk Significance
D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA1 of a Fire Originating In FA1
The activation of a preaction sprinkler or Halon 1301 suppression system would not adversely impact equipment/components credited by the NSCA in the Reactor Building. Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Suppression effects are not expected to extend beyond the area of fire origin.

Scenario 2: Suppression Affects in FA1 of a Fire Originating Outside of FA1
Based on the limited and localized area affected by fire suppression system activation, operation of a fire suppression system outside of FA1 is not expected to impact fire suppression systems within FA1 that could have an impact on the nuclear safety performance criteria.

Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied. Note that VFDR-01-002 was completely resolved by crediting a modification to the plant.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin.

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks. See additional discussion related to this fire area in N1-FRE-F001.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-In-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02. Negligible intervening combustibles between Fire Areas 1 and 2 per FPEE 1-85-003.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	**REACTOR BUILDING EAST EL 198-0 THRU EL 340-0**

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems for partial coverage and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-001	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between RWCU isolation valves IV-33-02R, IV-33-04, and BV-33-41. One of these valves is required closed to support inventory control. Fire damage to cables 161-204, 161-205, 161-216, 167-6, or 167-7 can fail isolation valve IV-33-02R open. Fire damage to cables 12DV-26, 12DV-44, or 12DV-45 can fail isolation valve IV-33-04 open. Fire damage to cable 11-99, external hot short, or cable 1S-1581, internal hot short, can maintain blocking valve BV-33-41 open.

51-9133191 Appendix A.5, Section A.5.4 Separation Concerns

Disposition: This VFDR has been brought into compliance with NFPA 805 section 4.2.3, deterministic approach, via a plant modification as described in Attachment S.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption assumes a loss of instrument air for a fire in any area of the plant.
A loss of instrument air fails CTS valve BV-80-44 closed, preventing the available CTS pump, PMP-80-04, from discharging to the spray header. The redundant Path C flow paths are not available due to following: Prior to a loss of instrument air an internal wire-to-wire short on cable 1K-10 can close IV-80-16 isolating the normal CTS PMP-80-04 flow path. Prior to a loss of instrument air an internal wire-to-wire short on cable 1K-65 can close IV-80-35 isolating the PMP-80-04 alternate flow path. An internal wire-to-wire short on cable 167-271 can close PMP-80-24 suction valve IV-80-21. Bypass to Torus isolation valve FCV-80-118 may not be available due to a ground or open on cable 167-196 or 167-197 The result is a loss of Torus cooling.

51-9133191 Appendix A.5, Section A.5.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in the area for the ERV's with regards to pressure control. ERV's PSV-01-102C, PSV-01-102D and PSV-01-102F may spuriously open. If this occurs in conjunction with, or any time after, the operator opening the Train 11 ERV's, fuel exposure may exceed desired limits. PSV-01-102C may spuriously open due to a cable-to-cable short on cable 1F-156 or 1F-177 in conjunction with an open in cable 1F-181 or 1F-179. Also individual wire-to-wire shorts on the following cables can cause PSV-01-102C to open: 1F-20, 1F-127, and 1F-174. An external cable-to-cable short on cable 1F-181 or 1F-176 will also open the valve.PSV-01-102D may spuriously open due to a cable-to-cable short on cable 1F-156 or 1F-177 or 1F-181 in conjunction with an open in cable 1F-183. Also individual wire-to-wire shorts on the following cables can cause PSV-01-102D to open: 1F-20, 1F-174, and 1F-128. An external cable-to-cable short on cable 1F-176 will also open the valve.PSV-01-102F may spuriously open due to a cable-to-cable short on cable 1F-156 or 1F-177 in conjunction with an open in cable 1F-179 or 1F-183. Also individual wire-to-wire shorts on the following cables can cause PSV-01-102F to open: 1F-21, 1F-129, and 1F-175. An external cable-to-cable short on cable 1F-183 will also open the valve.

51-9133191 Appendix A.5, Section A.5.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-005	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in the area for the reactor drain valves BV-37-08R and BV-37-09R resulting in an inventory control concern. These valves are required closed for inventory control. An internal wire-to-wire short on cable 155-34 or a cable-to-cable three phase short to cable 155-32 spuriously opens BV-37-08R. A cable-to-cable three phase short to cable 155-36 spuriously opens BV-37-09R. The result is a loss of inventory to a Drywell Equipment Drain Tank.

51-9133191 Appendix A.5, Section A.5.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in the area for the reactor head vent valves BV-37-01 and BV-37-02 resulting in an inventory control concern. These valves are required closed for inventory control. A cable-to-cable three phase short to cable 171-174 spuriously opens BV-37-01. An internal wire-to-wire short on cable 171-179, or a cable-to-cable three phase short to cable 171-177 opens BV-37-02. The result is an inventory loss to a Drywell Equipment Drain Tank.

51-9133191 Appendix A.5, Section A.5.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a plant modification, as described in Attachment S, to reduce risk. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area for process monitoring associated with Torus level. Both trains of Torus level indication are lost due to fire damage. CTS Pump PMP-80-03 discharge pressure indicator PI-80-54A is credited to monitor level. PI-80-54A is powered from UPS-172A or UPS-172B. However, the power supply to both UPS's and Train 12 Battery Chargers BC-B12-1 AND BC-B12-1, Power Board 17, is impacted by a fire in this area resulting in no Torus level indication available after Battery 12 drains. Torus level indication is a required indication to support the credited Path C shutdown method.

Components and related cables or instrument sensing lines affected:
LI-58-06A (Train 11): 1K-106, LI-58-06A-LINEH, LI-58-06A-LINEL
LI-58-05A (Train 12): 1K-107, LI-58-05A-LINEH, LI-58-05A-LINEL

Components lost due to loss of power supply:
PI-80-54A (Train 12): Power source PB-RSP12 affected due to Power Board 17 in the fire area. Loss of Power Board 17 results in loss of UPS-UPS172A and UPS-172B, and battery chargers BC-B12-1 and BC-B12-2.

51-9133191 Appendix A.5, Section A.5.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-009	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for process monitoring associated with Drywell temperature. All trains of Drywell temperature indication are lost due to fire damage. Drywell temperature indication is a required indication to support the credited Path C shutdown method.

Components and related cables affected:
TI-201-27B (Train 11): 1S-1968, 1S-2261
TI-201-50B (Train 11): 1S-1968, RSP-4
TI-201-33B (Train 12): 1S-2260, RSP-62
TI-201-51B (Train 12): 1S-1968, RSP-33, RSP-88

51-9133191 Appendix A.5, Section A.5.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-010	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for process monitoring associated with Reactor temperature. All trains of Reactor temperature indication are lost due to fire damage. Reactor temperature indication is a required indication to support the credited Path C shutdown method.

Components and related cables affected:
TI-32-03B (Train 11): 1S-1968, RSP-2
TI-32-03D Train 12): 171-135, 171-140, 1U-187
TI-32-04B (Train 12): RSP-30, RSP-62

Components lost due to loss of power supply:
TI-32-04D (Train 12): Power source PB-RSP12 affected due to Power Board 17 in the fire area. Loss of Power Board 17 results in loss of UPS-UPS172A and UPS-172B, and battery chargers BC-B12-1 and BC-B12-2.

51-9133191 Appendix A.5, Section A.5.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
1	REACTOR BUILDING EAST EL 198-0 THRU EL 340-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-011	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for process monitoring associated with Reactor fuel zone level indication. Both trains of Reactor fuel zone level indication may be lost. Reactor fuel zone level is a required indication to support the credited Path C shutdown method which credits opening ERV's and steaming to the Torus.

Components and related cables or instrument sensing lines affected:
LI-36-43: 1S-1932, LI-36-43-LINEL
LI-36-44: 1S-1936, 1S-1937, 1S-1941, 1S-1969, LI-36-44-LINEL, RSP-62

51-9133191 Appendix A.5, Section A.5.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-01-012	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	**REACTOR BUILDING WEST EL 198-0 THRU EL 340-0**

Fire Zone	Fire Zone Description
FBZR237N	REACTOR BUILDING EL 237-0 COL N-Q, ROW 8-9
FBZR261N	REACTOR BUILDING EL 261-0 COL N-Q, ROW 8-9
FBZR281N	REACTOR BUILDING EL 281-0 COL M-Q, ROW 6-7
FBZR281S	REACTOR BUILDING EL 281-0 COL K-L, ROW 7-8
FBZR298N	REACTOR BUILDING EL 298-0 COL N-Q, ROW 7.5-8.5
FBZR298S	REACTOR BUILDING EL 298-0 COL K-L, ROW 7-8
FBZR318N	REACTOR BUILDING EL 318-0 COL M-Q, ROW 6-7
FBZR318S	REACTOR BUILDING EL 318-0 COL K-M, ROW 6-7
FBZR340N	REACTOR BUILDING EL 340-0 COL M-Q, ROW 6-7
FBZR340S	REACTOR BUILDING EL 340-0 COL L-N, ROW 7-8
R1B	CTS PUMP ROOM, CS PUMP ROOM, GENERAL FLOOR AREA WEST EL 198-0 & 237-0
R2B	GENERAL FLOOR AREA WEST EL 261-0
R2C	SHUTDOWN COOLING ROOM EL 261-0
R2D	REACTOR BUILDING TRACK BAY EL 261-0
R3B	GENERAL FLOOR AREA WEST EL 281-0
R4B	GENERAL FLOOR AREA WEST EL 298-0
R4C	EMERGENCY CONDENSER ISOLATION VALVE ROOM EL 298-0
R5B	GENERAL FLOOR AREA WEST EL 318-0
R6B	GENERAL FLOOR AREA WEST EL 340-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required. However, the Reactor Mode Switch in the Main Control Room is utilized to actuate the Primary Scram SOV's to meet the Performance Goal in Fire Zone R1B for the Reactivity Control Function.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	REACTOR BUILDING WEST EL 198-0 THRU EL 340-0

(b) Inventory Control Function	Path D is credited to achieve HSD and CSD. Path D is accomplished via the use of Train 12 Systems. The ERV's are opened, CS is used to flood the vessel, CTS and CTSRW are used to remove decay heat from the Torus. Path D Control room ventilation is credited to support shutdown	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control is accomplished by ensuring the Path D ERVs are functional and Path C ERVs maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(d) Decay Heat Removal Function	Path D is credited to achieve HSD and CSD. Path D is accomplished via the use of Train 12 Systems. The ERV's are opened, CS is used to flood the vessel, CTS and CTSRW are used to remove decay heat from the Torus.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Torus temperature o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	REACTOR BUILDING WEST EL 198-0 THRU EL 340-0

(f) Vital Auxiliaries	1) AC and DC Distribution System The Train 12 emergency diesel generator is credited for supplying power for the shutdown process when AC power is required to operate shutdown equipment. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC SystemsHVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	**REACTOR BUILDING WEST EL 198-0 THRU EL 340-0**

Path D – Train 12 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-04-002 Rev. 0

Eng Eval Summary: FPEE 1-04-002 Rev. 0 - Evaluation of Fire Effects on Current Transformers and Instrument Sensing Lines and The Plant Safe Shutdown Capability
The analysis provided in this FPEE demonstrates that a postulated fire at NMP1 and the resultant fire effects on Current Transformers and instrument tubing lines will not affect the existing assumptions in the NMP1's present Appendix R Safe Shutdown Analysis. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-12-001 Rev. 0

Eng Eval Summary: This Engineering Evaluation provides analysis of the existing NMP1 approved exemptions from 10CFR50, Appendix R and input to NFPA-805 Licensing Amendment Request (LAR) submittal with respect to previously-approved exemptions to Appendix R to 10 CFR Part 50. This document provides the evaluation, analysis and basis to demonstrate that current plant configuration and NMP1 Fire Protection Program complies with the requirements of 10CFR50.48(c), NFPA 805. The evaluation documents the similar requirements of 10CFR50, Appendix R and NFPA 805. This evaluation demonstrates that the current plant configuration and fire protection features comply with the 10CFR50.48(c) licensing basis.

Engineering Eval Id: FPEE 1-85-003 Rev. 0

Eng Eval Summary: This evaluation documents the acceptibility of negligible intervening combustibles between Fire Areas 1 and 2. This evaluation reflects the CENG interpretation of the NRC position on the acceptability of negligible intervening combustibles.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	REACTOR BUILDING WEST EL 198-0 THRU EL 340-0

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-005 Rev. 0

Eng Eval Summary: FPEE 1-91-005 Rev. 0 - Penetration Seal Detail FS-2 for Penetration 3RF-80
Evaluation justifies method for sealing penetration 3RF-80 located in the Reactor Building 298 ft. elevation floor slab separating the area around Heat Exchanger No. 13 and the West Instrument Room. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-92-001 Rev. 0

Eng Eval Summary: FPEE 1-92-001 Rev. 0 - Penetration Seal Enlarged Boot Detail
Evaluation determines the acceptability of boot seal designs for pipe sizes larger than those qualified by test. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
FBZR237N	Detection	DA-4076W	Y	N	N	Y	N	
	Suppression	WP-4076	Y	N	N	Y	N	
FBZR261N	Detection	DA-4116E	Y	N	N	Y	N	
	Detection	DA-4116W	Y	N	N	Y	N	
	Suppression	WP-4116	Y	N	N	Y	N	
	Suppression	WP-4116W	Y	N	N	Y	N	
FBZR281N	Detection	D-4156	Y	N	N	Y	N	
	Detection	DA-4116W	Y	N	N	Y	N	
	Suppression	WP-4116	Y	N	N	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
2 | **REACTOR BUILDING WEST EL 198-0 THRU EL 340-0**

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
FBZR281S	Detection	DA-4116W	Y	N	N	Y	N	
	Suppression	WP-4116	Y	N	N	Y	N	
FBZR298N	Detection	DA-4237	Y	N	N	Y	N	
	Suppression	WP-4237	Y	N	N	Y	N	
FBZR298S	Detection	DA-4237	Y	N	N	Y	N	
	Suppression	WP-4237	Y	N	N	Y	N	
FBZR318N	Detection	DA-4237	Y	N	N	Y	N	
	Suppression	WP-4237	Y	N	N	Y	N	
FBZR318S	Detection	DA-4237	Y	N	N	Y	N	
	Suppression	WP-4237	Y	N	N	Y	N	
FBZR340N	Detection	D-4267	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
FBZR340S	Detection	D-4267	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
R1B	Detection	D-4016	N	N	N	Y	N	
	Detection	D-4036	N	N	N	Y	N	
	Detection	DA-4076W	N	N	N	Y	N	
	Suppression	None	N	N	N	N	N	
R2B	Detection	DA-4116W	N	N	N	Y	N	
	Suppression	WP-4116W	N	N	N	N	Y	
R2C	Detection	DA-4116W	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
R2D	Detection	None	N	N	N	N	N	
	Suppression	SP-4126	N	N	N	N	Y	
R3B	Detection	D-4156	N	N	Y	Y	N	
	Suppression	None	N	N	N	N	N	
R4B	Detection	D-4197	N	N	N	N	Y	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
2 | **REACTOR BUILDING WEST EL 198-0 THRU EL 340-0**

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
	Detection	D-4207	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
R4C	Detection	DX-4217A/B	N	N	N	N	Y	
	Suppression	H-4217	N	N	N	N	Y	
R5B	Detection	DA-4237	N	N	N	N	Y	
	Suppression	WP-4237	N	N	N	N	Y	
R6B	Detection	D-4267	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	

Legend:

Table Field: "Required ?"

S - Required for Chapter 4 Separation Criteria
L - Required for NRC Approved Licensing Action
E - Required for Existing Engineering Equivalency Evaluation
R - Required for Risk Significance
D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA2 of a Fire Originating In FA2
The activation of a preaction sprinkler or Halon 1301 suppression system would not adversely impact equipment/components credited by the NSCA in the Reactor Building. Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Suppression effects are not expected to extend beyond the area of fire origin.

Scenario 2: Suppression Affects in FA2 of a Fire Originating Outside of FA2
Based on the limited and localized area affected by fire suppression system activation, operation of a fire suppression system outside of FA2 is not expected to impact fire suppression systems within FA2 that could have an impact on the nuclear safety performance criteria.

Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Reference: Document No.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	**REACTOR BUILDING WEST EL 198-0 THRU EL 340-0**

N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis
B-40146-C, Sheet 3, Rev. 1

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks. See additional discussion related to this fire area in N1-FRE-F001.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense- in Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02. Negligible intervening combustibles between Fire Areas 1 and 2 per FPEE 1-85-003.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	REACTOR BUILDING WEST EL 198-0 THRU EL 340-0

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems for partial coverage and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-001	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area between the credited Containment Spray flow paths. Containment Spray is required to ensure inventory control and decay heat removal. Credited Path D CTS pump PMP-80-03 may not be available due to damage to its power cable (103-17) or spurious closure of suction isolation valve IV-80-02 due to an internal wire-to-wire short on cable 167-273. Path D CTS pump PMP-80-23 is available to support Torus cooling. However, the credited CTS flow path utilizing Pump PMP-80-23 may not be available. An internal wire-to-wire short on cable 1K-65 closes IV-80-35. An internal wire-to-wire short on either cable 1K-131 or 1K-132 closes BV-80-45. This results in isolation of the credited Path D CTS flow path.

51-9133191 Appendix A.6, Section A.6.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	REACTOR BUILDING WEST EL 198-0 THRU EL 340-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in the area for the ERV's with regards to pressure control. ERV's PSV-01-102A, PSV-01-102B and PSV-01-102E may spuriously open. If this occurs in conjunction with, or any time after, the operator opening the Train 12 ERV's, fuel exposure may exceed desired limits. PSV-01-102A may spuriously open due to a cable-to-cable short on cable 1F-151 or 1F-180 in conjunction with an open in cable 1F-178. Also individual wire-to-wire shorts on the following cables can cause PSV-01-102A to open: 1F-124, 1F-170, and 1F-184. An external cable-to-cable short on cable 1F-172 or 1F-180 will also open the valve PSV-01-102A. The ERV PSV-01-102B may spuriously open due to a cable-to-cable short on cable 1F-151 in conjunction with an open in cable 1F-182 or 1F-178. Also individual wire-to-wire shorts on the following cables can cause PSV-01-102B to open: 1F-125, 1F-170, and 1F-184. An external cable-to-cable short on cable 1F-172 or 1F-180 will also open the valve PSV-01-102B. The ERV PSV-01-102E may spuriously open due to a cable-to-cable short on cable 1F-151 in conjunction with an open in cable 1F-178 or 1F-182. Also individual wire-to-wire shorts on the followings cables can cause PSV-01-102E to open: 1F-126, 1F-171, and 1F-185. An external cable-to-cable short on cable 1F-173 or 1F-182 will also open the valve.

51-9133191 Appendix A.6, Section A.6.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in the area that affects the inventory control function. Various combinations of cable-to-cable shorts on cables 1S-1041, 1S-1042, 1S-1891 and 1S-1892 cause IV-44.2-15, IV-44.2-16, IV-44.2-17 and IV-44.2-18 to spuriously open (post scram) resulting in an inventory loss.

51-9133191 Appendix A.6, Section A.6.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-005	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	**REACTOR BUILDING WEST EL 198-0 THRU EL 340-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in the area for the reactor blocking valve BV-37-08R and reactor blowdown valve BV-37-09R resulting in an inventory control concern. These valves are required closed for inventory control. An internal wire-to-wire short on cable 155-34 or a cable-to-cable three phase short to cable 155-32 spuriously opens BV-37-08R. A cable-to-cable three phase short to cable 155-36 spuriously opens BV-37-09R. The result is a loss of inventory to a Drywell Equipment Drain Tank.

51-9133191 Appendix A.6, Section A.6.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for process monitoring associated with Drywell temperature. All trains of Drywell temperature indication in the control room are lost due to fire damage or a loss of power. Drywell temperature indication is a required indication to support the credited Path D shutdown method. Indication is available at RSP12. Dispatching an operator to the RSP to observe the indication is classified as a recovery action per NEI 04-02 and shall be evaluated in the Fire Risk Assessment for this fire area.

Components and related cables affected:
TI-201-27B (Train 11): Loss of Power. Power source to PB-RPS11 affected due to Power Board 16 in the area. Power Board 16 supplies both UPS-UPS162A and UPS-UPS162B, and battery chargers BC-B11-1 and BC-B11-2.
TI-201-33B (Train 12): 1U-284

Component available at RSP12:
TI-201-51B (Train 12)

51-9133191 Appendix A.6.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	**REACTOR BUILDING WEST EL 198-0 THRU EL 340-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for process monitoring associated with Reactor level. All trains of Reactor level indication are lost due to fire damage. Reactor level indication is a required indication to support the credited Path D shutdown method.

Components and related cables or instrument sensing lines affected:
LI-36-09 (Train 11): 1F-60A, 1M-175, 1S-1749, LI-36-09-LINEH, LI-36-09-LINEL
LI-36-19 (Train 11): 1F-151, 1S-1749, 1S-2073, RSP-27
LI-26-28 (Train 11): 1F-151, 1S-1749, 1S-2073, RSP-27, RSP-8
LI-36-10 (Train 12): 1S-1748
LI-36-20 (Train 12): 1F-117, 1M-185, 1S-1748, RSP-55, LI-36-20-LINEH, LI-36-20-LINEL
LI-36-26 (Train 12): 1M-182, 1S-1748, RSP-37, RSP-55
LI-36-43 (Train 11): 1S-1934, 1S-1935, 1S-1940, LI-36-43-LINEH, LI-36-43-LINEL
LI-36-44 (Train 12): 1S-1937, 1S-1941, LI-36-44-LINEH, LI-36-44-LINEL

51-9133191 Appendix A.6, Section A.6.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	**REACTOR BUILDING WEST EL 198-0 THRU EL 340-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-009	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for process monitoring associated with Torus water temperature indication. All trains of Torus water temperature indication may be lost. Torus water temperature indication is a required indication to support the credited Path D shutdown method.

Components and related cables affected:
T1-201.2-519 (Train 11): 1S-1979, 1S-1980, 1S-1982, 1S-1983, 1S-1984, 1S-1985, 1S-1986, 1S-1987, 1S-1988, 1S-1989, 1S-1990, 1S-1991, 1S-1993
T1-201.2-521B (Train 11): 1S-1984, RSP-3A, RSP-3B
TI-201.2-520 (Train 12): 1S-1994, 1S-1995
TI-520.2-522B (Train 12): RSP-32A, RSP-32B

51-9133191 Appendix A.6, Section A.6.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-010	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for Control Room cooling. Control room cooling capability may be lost due to fire damage to cabling for all three RBCLC pumps, PMP-70-01, PMP-70-02, and PMP-70-03.

51-9133191 Appendix A.6, Section A.6.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
2	REACTOR BUILDING WEST EL 198-0 THRU EL 340-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-02-011	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in this area that affects the reactivity control performance goal for reactor scram. Various combinations of cable-to-cable shorts on cables 1S-1041, 1S-1042, 1S-1891 and 1S-1892 impact Channel 11 scram solenoids SOV-113-272, SOV-113-273 and SOV-113-275; and Channel 12 scram solenoids SOV-113-271, SOV-113-274 and SOV-113-276 associated with IV-44.2-15, IV-44.2-16, IV-44.2-17 and IV-44.2-18. The SOV's remaining energized prevents venting the scram air header which, in turn, prevents reactor scram from the control room.

51-9133191 Appendix A.6, Section A.6.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
3	DRYWELL EL 237-0 THRU 318-0

Fire Zone	Fire Zone Description
R1	DRYWELL EL 237 - 318

Regulatory Basis: NFPA 805 Section 4.2.3.1 - Deterministic Approach

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: None

Eng Eval Summary:

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
R1	Detection	DA-4086	N	N	N	N	N	
	Suppression	None	N	N	N	N	N	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria
L - Required for NRC Approved Licensing Action
E - Required for Existing Engineering Equivalency Evaluation
R - Required for Risk Significance
D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA3 of a Fire Originating In FA3
There are no installed fire suppression systems in FA3 that could be actuated as a result of a fire originating within the fire area.

Scenario 2: Suppression Affects in FA3 of a Fire Originating Outside of FA3
There are no installed fire suppression systems in FA3 that could be actuated as a result of a fire originating outside of the fire area.

Since there are no installed fire suppression systems in FA3, there is no flooding potential resulting from the fire suppression systems.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
3	DRYWELL EL 237-0 THRU 318-0

51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

N/A - The Drywell atmosphere is maintained inert during normal power operation; therefore, no fires are postulated.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
4	**FOAM ROOM EL 261-0**

Fire Zone	Fire Zone Description
AB1F	FOAM ROOM EL 261-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	The Train 12 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 support systems. Path A and Path B are credited to achieve CSD. Path A CSD is accomplished via use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems. Path B CSD is accomplished via use of the Train 12 SDC System supported by the Train 12 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
4	FOAM ROOM EL 261-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
4	FOAM ROOM EL 261-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system. Path B is generally supported by the Train 12 AC and DC power systems. Some components in Path B systems are powered by the Train 11 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open)	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
4	FOAM ROOM EL 261-0

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:

Path C – Train 11 positive pressure flow path.
Path D – Train 12 positive pressure flow path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
AB1F	Detection	D-8151	N	N	N	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
4	FOAM ROOM EL 261-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	Required ? L	Required ? E	Required ? R	Required ? D	Notes
	Suppression	None	N	N	N	N	N	

Legend: Table Field: "Required ?"

S - Required for Chapter 4 Separation Criteria
L - Required for NRC Approved Licensing Action
E - Required for Existing Engineering Equivalency Evaluation
R - Required for Risk Significance
D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA4 of a Fire Originating In FA4
There are no installed fire suppression systems in FA4 that could be actuated as a result of a fire originating within the fire area.

Scenario 2: Suppression Affects in FA4 of a Fire Originating Outside of FA4
There are no installed fire suppression systems in FA4 that could be actuated as a result of a fire originating outside of the fire area.

Since there are no installed fire suppression systems in FA4, there is no flooding potential resulting from the fire suppression systems.

Reference: Document No:
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
4	FOAM ROOM EL 261-0

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-04-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18 locally to provide makeup from the CRD system.

51-9133191 Appendix A.7, Section A.7.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
4	**FOAM ROOM EL 261-0**

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-04-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. BV-60-13 may fail closed on loss of instrument air. Valve is required open for hot shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-60-13.

51-9133191 Appendix A.7, Section A.7.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-04-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. LCV-60-17 may fail open on loss of instrument air. Valve is required to close for hot shutdown to support decay heat removal. A Recovery Action may be required to close valve VLV-60-12.

51-9133191 Appendix A.7, Section A.7.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
4	**FOAM ROOM EL 261-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-04-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. LCV-60-18 may fail open on loss of instrument air. Valve is required to throttle for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle valve VLV-60-11.

51-9133191 Appendix A.7, Section A.7.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-04-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

51-9133191 Appendix A.7, Section A.7.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-04-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

51-9133191 Appendix A.7, Section A.7.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
4	FOAM ROOM EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-04-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-10 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-10.

51-9133191 Appendix A.7, Section A.7.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-04-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

51-9133191 Appendix A.7, Section A.7.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	**TURBINE BUILDING EL 240-0 THRU 369-0**

Fire Zone	Fire Zone Description
FBZT261N	TURBINE BUILDING FIRE BREAK ZONE NORTH EL 261-0
FBZT261S	TURBINE BUILDING FIRE BREAK ZONE SOUTH EL 261-0
OG1	GENERAL FLOOR AREA EL 232-0
OG2	GENERAL FLOOR AREA EL 247-0
OG3	GENERAL FLOOR AREA EL 261-0
T1	TURBINE CONDENSER/HEATER BAY AREA EL 250-0
T1A	TURBINE BUILDING EL 240-261 MSIV ROOM & STEAM TUNNEL
T3A	GENERAL FLOOR AREA EAST OF MSIV ROOM AND FIRE ZONE T1 EL 261-318
T3B	GENERAL FLOOR AREA WEST OF MSIV ROOM; ALSO SOUTH AND WEST OF FIRE ZONE 1 EL 237-0 & 261-0
T4A	GENERAL FLOOR AREA EAST OF FIRE ZONE T1 EL 277-0
T4B	GENERAL FLOOR AREA WEST OF FIRE ZONE T1 EL 277-0
T4C	HYDROGEN SEAL OIL UNIT ROOM EL 277-0
T4D	BATTERY ROOM EL 277
T5A	GENERAL FLOOR AREA NORTH EL 291-0
T6A	GENERAL FLOOR AREA NORTH EL 305-6
T6B	TURBINE LAYDOWN AREA EAST EL 300-0
T6C	GENERAL FLOOR AREA SOUTH EL 300-0
T6D	MECHANICAL STORAGE AREA EL 300-0
T7A	GENERAL FLOOR AREA SOUTH EL 320-0
T8A	GENERAL FLOOR AREA NORTH EL 333-0, GENERAL FLOOR AREA NORTH EL 351-0, GENERAL FLOOR AREA EAST EL 369
T8B	GENERAL FLOOR AREA WEST EL 369-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(b) Inventory Control Function	TheTrain 12 CRD pump is credited to provide makeup to the RPV	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 5

Fire Area Description TURBINE BUILDING EL 240-0 THRU 369-0

(c) Pressure Control	Pressure Control is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(d) Decay Heat Removal Function	Path A is credited to achieve HSD via use of EC's 111 and 112 Decay Heat Removal and Train 11 support system. Trains 11, 125V DC system is credited to support shutdown for the duration of Battery 11 Path B is credited to achieve CSD. Path B CSD is accomplished via use of the Train 12 SDC System supported by the Train 12 RBCLC and ESW Systems Path D Control Room Ventilation is credited to support shutdown	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure o Emergency Condenser Level	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

(f) Vital Auxiliaries	1) AC and DC Distribution System The Train 11 emergency diesel generator is credited for supplying power for the shutdown process when AC power is required to operate shutdown equipment. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows: Path C – Train 11 positive pressure flow path.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: EIR 51-9156616 Rev. 1

Eng Eval Summary: EIR 51-9156616 Rev. 1 - Nine Mile Point Unit 1 Code Compliance Evaluation for NFPA 30, Flammable and Combustible Liquids Code, 2000 Edition
Evaluation provides comparison of the plant combustible liquid storage configurations and methods to applicable NFPA 30 requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-03-002 Rev. 0

Eng Eval Summary: FPEE 0-03-002 Rev. 0 - Evaluation of Interim Action To Prevent Personnel Injury From CO2
Evaluation justifies use of temporarily placing automatic CO2 systems in alarm-only mode due to adverse trend of unplanned system actuations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-01-003 Rev. 1

Eng Eval Summary: FPEE 1-01-003 Rev. 1 - Turbine Building Diesel Generator Building Separation Above 300 ft
Evaluation determines that adequate fire protection features exist to mitigate spread of a fire between the east wall of the Turbine Building 300 ft. elevation to the roof of the Diesel Generator Building. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-04-002 Rev. 0

Eng Eval Summary: FPEE 1-04-002 Rev. 0 - Evaluation of Fire Effects on Current Transformers and Instrument Sensing Lines and The Plant Safe Shutdown Capability
The analysis provided in this FPEE demonstrates that a postulated fire at NMP1 and the resultant fire effects on Current Transformers and instrument tubing lines will not affect the existing assumptions in the NMP1's present Appendix R Safe Shutdown Analysis. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

Engineering Eval Id: FPEE 1-12-001 Rev. 0

Eng Eval Summary: This Engineering Evaluation provides analysis of the existing NMP1 approved exemptions from 10CFR50, Appendix R and input to NFPA-805 Licensing Amendment Request (LAR) submittal with respect to previously-approved exemptions to Appendix R to 10 CFR Part 50. This document provides the evaluation, analysis and basis to demonstrate that current plant configuration and NMP1 Fire Protection Program complies with the requirements of 10CFR50.48(c), NFPA 805. The evaluation documents the similar requirements of 10CFR50, Appendix R and NFPA 805. This evaluation demonstrates that the current plant configuration and fire protection features comply with the 10CFR50.48(c) licensing basis.

Engineering Eval Id: FPEE 1-85-007 Rev. 0

Eng Eval Summary: FPEE 1-85-007 Rev. 0 - Waste Building Truck Bay West Wall
Evaluation justifies method to seal conduit penetrations through the west wall of the Waste Building Truck Bay and the north wall of the Turbine Building at the 261-ft. elevation. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-89-004 Rev. 0

Eng Eval Summary: FPEE 1-89-004 Rev. 0 - Thermal Shield Walls
Evaluation justifies use of thermal shield walls in the subject fire areas due to unsealed or non-rated penetrations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-003 Rev. 1

Eng Eval Summary: FPEE 1-91-003 Rev. 1 - Excessive Door Gap at Floor for Fire Doors D-84, D-108, D-111, D-245, and D-246
Evaluation determines adequacy of gaps under fire doors that exceed those allowed by NFPA 80. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 5

Fire Area Description TURBINE BUILDING EL 240-0 THRU 369-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
FBZT261N	Detection	DA-2081S	N	N	Y	Y	N	
	Detection	DA-2083M	N	N	Y	Y	N	
	Detection	DA-2083N	N	N	Y	Y	N	
	Suppression	WP-2083	N	N	Y	Y	N	
FBZT261S	Detection	DA-2161E	N	N	Y	Y	N	
	Detection	DA-2161M	N	N	Y	Y	N	
	Suppression	WP-2161	N	N	Y	Y	N	
OG1	Detection	D-7013	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	
OG2	Detection	D-7013	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	
OG3	Detection	D-7043	N	N	Y	Y	N	
	Suppression	SP-7053	N	N	Y	N	Y	
T1	Detection	D-1011	N	N	Y	Y	N	
	Detection	D-1021	N	N	Y	Y	N	
	Detection	D-1021A/B	N	N	Y	Y	N	
	Detection	D-1031	N	N	Y	Y	N	
	Detection	D-1031A/B	N	N	Y	Y	N	
	Detection	D-1061	N	N	Y	Y	N	
	Detection	D-1072	N	N	Y	Y	N	
	Detection	D-1091	N	N	Y	Y	N	
	Detection	D-1155	N	N	Y	Y	N	
	Detection	D-1165	N	N	Y	Y	N	
	Detection	D-1175	N	N	Y	Y	N	
	Detection	D-1195	N	N	Y	Y	N	
	Detection	D-1195EX	N	N	Y	Y	N	
	Detection	D-2385	N	N	Y	Y	N	
	Detection	D-2405	N	N	Y	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area **Fire Area Description**
5 **TURBINE BUILDING EL 240-0 THRU 369-0**

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES								
Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
	Detection	DX-1011A/B	N	N	Y	Y	N	
	Detection	DX-1091A/B	N	N	Y	Y	N	
	Suppression	C-1155	N	N	N	Y	N	Manual actuation of CO2
	Suppression	C-1185	N	N	N	Y	N	Manual actuation of CO2
	Suppression	WD-101F	N	N	Y	Y	N	
	Suppression	WD-102F	N	N	Y	Y	N	
	Suppression	WD-103F	N	N	Y	Y	N	
	Suppression	WD-109F	N	N	Y	Y	N	
	Suppression	WD-115F	N	N	Y	Y	N	
	Suppression	WD-116F	N	N	Y	Y	N	
	Suppression	WD-117F	N	N	Y	Y	N	
T1A	Detection	None	N	N	N	N	N	
	Suppression	None	N	N	N	N	N	
T3A	Detection	D-1131	N	N	Y	Y	N	
	Detection	D-2083PL	N	N	Y	Y	N	
	Detection	D-2131	N	N	Y	Y	N	
	Detection	D-6113	N	N	Y	Y	N	
	Detection	DA-1131	N	N	Y	Y	N	
	Detection	DA-2081S	N	N	Y	Y	N	
	Detection	DA-2083M	N	N	Y	Y	N	
	Detection	DA-2083N	N	N	Y	Y	N	
	Suppression	C-1131	N	N	N	N	Y	
	Suppression	C-1131TK	N	N	N	N	Y	
	Suppression	SP-2131	N	N	Y	Y	N	
	Suppression	SP-5033	N	N	Y	Y	N	
	Suppression	WD-2263FL	N	N	Y	Y	N	
	Suppression	WP-2083	N	N	Y	Y	N	
T3B	Detection	D-2092PL	N	N	Y	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
5 | TURBINE BUILDING EL 240-0 THRU 369-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES									
Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
	Detection	DA-2081S	N	N	Y	Y	N	
	Detection	DA-2092E	N	N	Y	Y	N	
	Detection	DA-2092MG	N	N	Y	Y	N	
	Detection	DA-2092W	N	N	Y	Y	N	
	Detection	DA-2102	N	N	Y	Y	N	
	Detection	DA-2161E	N	N	Y	Y	N	
	Detection	DA-2161M	N	N	Y	Y	N	
	Detection	DA-2162W	N	N	Y	Y	N	
	Suppression	C-2092MG	N	N	N	N	Y	
	Suppression	WD-2102	N	N	Y	Y	N	
	Suppression	WD-2161	N	N	Y	Y	N	
	Suppression	WD-8072	N	N	Y	Y	N	
	Suppression	WD-8121	N	N	Y	Y	N	
	Suppression	WP-2083	N	N	Y	Y	N	
	Suppression	WP-2092	N	N	Y	Y	N	
	Suppression	WP-2161	N	N	Y	Y	N	
T4A	Detection	D-2194	N	N	Y	Y	N	
	Detection	D-2214	N	N	Y	Y	N	
	Detection	D-2234	N	N	Y	Y	N	
	Detection	D-2263	N	N	Y	Y	N	
	Detection	D-2263FL	N	N	Y	Y	N	
	Detection	D-3054	N	N	Y	Y	N	
	Detection	DA-2083M	N	N	Y	Y	N	
	Detection	DA-2092E	N	N	Y	Y	N	
	Suppression	SP-2214	N	N	Y	Y	N	
	Suppression	WD-2263FL	N	N	Y	Y	N	
	Suppression	WP-2234	N	N	Y	Y	N	
T4B	Detection	D-2204	N	N	Y	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 5

Fire Area Description TURBINE BUILDING EL 240-0 THRU 369-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES								
Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
	Detection	D-2224	N	N	Y	Y	N	
	Detection	D-2224PL	N	N	Y	Y	N	
	Detection	DA-2092W	N	N	Y	Y	N	
	Detection	DA-2234	N	N	Y	Y	N	
	Suppression	SP-2224	N	N	Y	N	Y	
	Suppression	WP-2092	N	N	Y	N	Y	
T4C	Detection	D-1114	N	N	Y	Y	N	
	Detection	DA-1114	N	N	Y	Y	N	
	Suppression	C-1114	N	N	N	N	Y	
	Suppression	WD-115F	N	N	Y	Y	N	
T4D	Detection	D-2194	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	
T5A	Detection	D-2294	N	N	Y	N	Y	
	Detection	D-2304	N	N	Y	N	Y	
	Suppression	SP-2314	N	N	Y	N	Y	
	Suppression	SP-2324	N	N	Y	N	Y	
T6A	Detection	D-2345	N	N	Y	Y	N	
	Detection	D-4237PL	N	N	Y	Y	N	
	Detection	DA-2365	N	N	Y	Y	N	
	Suppression	C-2365	N	N	N	N	Y	
	Suppression	SP-2314	N	N	Y	N	Y	
T6B	Detection	D-2355	N	N	Y	N	Y	
	Detection	D-2405	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	
T6C	Detection	D-2385	N	N	Y	N	Y	
	Detection	D-2395	N	N	Y	N	Y	
	Detection	D-2395FL	N	N	Y	N	Y	
	Detection	D-2395PL	N	N	Y	N	Y	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 5

Fire Area Description TURBINE BUILDING EL 240-0 THRU 369-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
	Suppression	WD-2395FL	N	N	Y	N	Y	
T6D	Detection	DA-2375	N	N	Y	N	Y	
	Suppression	WP-2375	N	N	Y	N	Y	
T7A	Detection	DA-2425	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	
T8A	Detection	D-2445	N	N	Y	N	Y	
	Detection	D-2485	N	N	Y	N	Y	
	Suppression	SP-2465	N	N	Y	N	Y	
T8B	Detection	D-2485	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA5 of a Fire Originating In FA5
The activation of a suppression system would not be expected to adversely impact equipment/components credited by the NSCA in the Turbine Building.

Scenario 2: Suppression Affects in FA5 of a Fire Originating Outside of FA5
Based on the limited impact a fire can have on plant suppression systems, the operation of fire suppression systems outside of FA5 would not be expected to cause activation of a fire protection system within FA5 that could have an impact on the nuclear safety performance criteria.

Water spray systems in FA5 are designed to spray on specific hazards. Water runoff from fire suppression activities in these areas would drain through floor drains, floor grating, or down stairwells to floor drain sumps. A floor drain sump, having a capacity of 1,600 gallons is provided in the Off-Gas Building. Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). The associated sump pumps automatically transfer to the UTILITY COLLECTOR TANK (normal line-up), the WASTE NEUTRALIZER TANK, or the FLOOR DRAIN COLLECTOR TANK where it is stored prior to processing by the liquid waste system.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

Reference: Document No.
N1-FSS-F001, Rev.1 Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis
P&ID C18045C,009 Rev. 29

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk and safety margin criteria were satisfied. Note that VFDR-05-011, VFDR-05-026, VFDR-05-027, VFDR-05-042 were completely resolved by crediting modifications to the plant. VFDR-05-040 and VFDR-05-043 were partially resolved by crediting modifications to the plant.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks. See additional discussion related to this fire area in N1-FRE-F001.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems for partial coverage, sprinkler systems for safety related cable trays stacked more than two deep and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-001	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR–05-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-39-13R and IV-39-14R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-117 can maintain EC system isolation valves IV-39-13R and IV-39-14R open resulting in an inventory loss to Main Steam drain line.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-39-11R and IV-39-12R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-111 can maintain EC system isolation valves IV-39-11R and IV-39-12R open resulting in an inventory loss to Main Steam drain line.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-05-04R, IV-05-12, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-157 valve can maintain both EC system isolation valves IV-05-04R and IV-05-12 open. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-31 can maintain both EC system isolation valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area for the MCR ventilating flow dampers BV-210-08 and BV-210-39 which are required to be closed. An external cable to cable short on cable 1671-95 maintains open or opens BV-210-08. An external cable to cable short on cable 1671-103 maintains open or opens BV-210-39. Maintaining both dampers open short circuits the ventilation flow to the MCR.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area for the damper BV-210-41 which is credited to open to provide ventilation flow to MCR. An internal wire to wire short on cable 1N-32 or an external cable to cable short on cable 1671-96 may cause the damper BV-210-41 to close. Closure of the damper BV-210-41 blocks all credited path-B (Train-12) ventilation flow to MCR.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for the ventilation fan FN-210-02 (Train-12) which is credited to provide ventilation flow to MCR. Fire damage to cable 1671-99 or cable 1N-33 can cause loss of ventilation supply fan FN-210-02 (Train-12) resulting in the loss of ventilation flow to MCR.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for the chilled water valves (CHW) BV-210.1-01 and BV-201.1-02 which are required to be open to provide CHW supply to the Control Room Coolers. An internal wire-to-wire short on either cable 1671-100, 1671-152 or 1N-28 can close CHW valve BV-210.1-01. An internal wire-to-wire short on either cable 1671-100 or 1671-153 can close CHW valve BV-210.1-02. The result is a loss of CHW supply to the control room coolers.
An open or ground of cable 161-46 can disable credited Chilled Water Pump PMP-210.1-36.
An internal wire-to-wire short on cable 1671-44 can close RBCLC valve BV-70-26 resulting in loss of heat sink for credited MCR ventilation chiller compressors CMPR-210.1-120-A and CMPR-210.1-120-B.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-009	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting Inboard MSIV's IV-01-01 and IV-01-02, and outboard MSIV's IV-01-03 and IV-01-04 which are required closed for HSD and CSD to support inventory control. Inboard MSIV IV-01-01 and IV-01-02 may remain open due to a loss of Train 11 and Train 12 AC electrical systems below 4kV. Outboard MSIV IV-01-03 may remain open due to a cable-to-cable short on any of the following 125V DC cables: 1F-39, 1F-39A, 1F-51, 1F-52 and 1F-52A. Outboard MSIV IV-01-04 may remain open due to either an internal wire-to-wire short on cable 1F-44 or cable-to-cable short on any of the following 125V DC cables: 1f-45, 1F-45A, 1F-57 and 1F-57A.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-010	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for postulated fire in this area for the Main Feed Water. Main Feed Water is required for inventory control. Feed Water Isolation Valves IV-31-07 and IV-31-08 may remain open due to loss of power due to fire damage to the following cables:

IV-31-07: 101-5, 101-6, 101-66, 1A-147, 102-43, 102-49, 1A-124, 102-1, 102-2, 16-3, 102-47, 102-66
IV-31-08: 101-64, 1A-161, 103-29, 103-49, 12B-28, 1A-134, 103-1, 103-2, 17-3, 101-65, 101-87

Main Feed Water Pump PMP-29-02 may spuriously start due to an internal wire-to-wire short on cable 11-78. Main Feed Water Pump PMP-29-03 may spuriously start due to an internal wire-to-wire short on cable 12-78. The clutch of the turbine shaft driven pump 29-01 may remain engaged due to a short on cable 1F-110 or fire damage to any of the following cables: 141-298, 1F-110, 1F-4, 1F-4A and 1S-1730. The result is reactor vessel overfill that has the potential to compromise the ability of the credited EC train to achieve and maintain HSD. The clutch is disengaged by a procedure driven operator action.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	**TURBINE BUILDING EL 240-0 THRU 369-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-011	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for postulated fire in this area impacting series RWCU isolation valves IV-33-02R, IV-33-04 and BV-33-41 which are required to be closed to support inventory control. Isolation valves IV-33-02R and IV-33-04 may remain open due to loss of AC power. Isolation valves IV-33-02R, IV-33-04 and BV-33-41 may remain open due to fire damage to the any of the cables listed:
Isolation valve IV-33-02R may remain open due a ground fault on cable 167-7. Isolation valve IV-33-04 may remain open due to an open circuit on cables 12B-24, 12DV-26, 12DV-44 or wire-to-wire short on cable 12DV-27. Isolation Valve BV-33-41 may remain open due to a cable-to-cable short on cable 11-99. The result is that the RCS remains connected to the Clean-Up system.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been brought into compliance with NFPA 805 section 4.2.3, deterministic approach, via a plant modification as described in Attachment S.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-012	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for postulated fire for the control circuit of the EC Valve IV-39-05. This valve is required to be open to support pressure control and decay heat removal functions. Both EC paths may be adversely affected by the fire and the credited EC's 111 and 112 may be impacted as follows:
IV-39-05 may remain closed due to: A cable-to-cable short on cable 1K-48 can apply a potential to solenoid 39-05H. A ground on the neutral side of 39-05H on cable 1K-48 coupled with any other ground from a Battery 12 circuit completes the path to battery neutral through interposing contact R37D. Contact R37D may remain closed due to a wire-to-wire short in cable 12B-56 in the master logic RPS-EMER-COOL-12. The result is that 39-05H remains energized. This scenario defeats the auto initiation feature provided for EC's 111 and 112. Operation from the Remote Shutdown Panels (RSP's) may not be possible due to a loss of power to each panel from fire damage to power cables.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	**TURBINE BUILDING EL 240-0 THRU 369-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-013	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for postulated fire for the control circuit of EC Valve IV-39-08R. This valve is required to be open. The non-credited EC's 121 and 122 may be impacted as follows: IV-39-08R may close due to: Internal wire-to-wire shorts (2) in conjunction with an open on cable 12DV-22, an internal wire-to-wire short (2) in conjunction with an open on cable 12DV-23, an internal wire-to-wire short (2) on cable RSP-83, an internal wire-to-wire short on cable RSP-76 in conjunction with an internal wire-to-wire short on cable RSP-83, an internal wire-to-wire short on cable 12DV-23 in conjunction with an internal wire-to-wire short on cable RSP-83, an internal wire-to-wire short on cable 12DV-41 in conjunction with an internal wire-to-wire short on cable RSP-83.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-014	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for postulated fire in this area for the control circuit of EC Valve IV-39-10R. This valve is required to be open. The non-credited EC's 121 and 122 may be impacted as follows The EC valve may be closed due to an internal wire-to-wire short in conjunction with an open on cable 161-192, and an internal wire-to-wire short in conjunction with an open on cable 161-193, isolating EC 121 and 122.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-015	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for postulated fire in this area for process monitoring associated with Reactor Coolant Temperature indication. Reactor coolant temperature indication is a required indication to support transition from the credited Path A hot shutdown method to the credited Path B cold shutdown method and maintaining cold shutdown.
All channels of Reactor Coolant Temperature Indication may be lost due to cable damage as follows:
TI-32-03B (Train 11): Fire damage to cable RSP-25
TI-32-03D (Train 12): Fire damage to cable 11-60 or 11-65 or 1S-1907 or 1S-1909 or 1S-1910 or 1S-1911 or 1S-1912 or 1S-1913
TI-32-04B (Train 12): Fire damage to cable RSP-30 or RSP-54
TI-32-04D (Train 12): Fire damage to cable 12-40 or 12-45 or 1S-1907 or 1S-1909 or 1S-1910 or 1S-1911 or 1S-1912 or 1S-1913.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-016	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for postulated fire in this area for process monitoring associated with Reactor Vessel Level indication. Reactor vessel level is a required indication to assure adequate level is maintained in the reactor vessel during the shutdown process.
All channels of Reactor Level indication may be lost due to cable damage or loss of power supply as follows.
LI-36-09 (Train 11): Loss of Power Supply to PB-RPS-11 from UPS-UPS-162A and UPS-UPS162B. Cables from UPS-UPS-162A affected are 11B-100, 11B-101, 11B-97, 11B-98, 11B-99, 16-125, 16-130, 16-131, 16-132, 16-133, 16-134, 16-135, 16-136, 16-137, 16-138, 16-139, 16-140, 16-141. Cables from UPS-UPS162B affected are 11B-100, 11B-102, 11B-97, 11B-98, 11B-99, 16-125, 16-130, 16-137, 16-138, 16-139, 16-140, 16-141, 16-142, 16-143.
LI-36-19 (Train 11): Fire damage to cable 1F-118 or 1S-1750
LI-36-28 (Train 11): Fire damage to cable 1S-1750
LI-36-10 (Train 12): Fire damage to cable 1F-156 or 1F-188 or 1S-1748
LI-36-20 (Train 12): Fire damage to cable 1F-117 or 1F-156 or 1S-1748 or RSP-55
LI-36-26 (Train 12): Fire Damage to cable 1F-156 or 1S-1748 or RSP-37 or RSP-55
LI-36-43 (Train 11): Fire damage to cable 1S-1932
LI-36-44 (Train 12): Loss of Power Supply to PB-RPS-12 from UPS-UPS-172A and UPS-UPS172B. Cables from UPS-UPS-172A affected are 12B-100, 12B-102, 12B-68, 12B-69, 12B-70, 12B-71, 12B-97, 12B-98, 12B-99, 17-100, 17-101, 17-102, 17-103, 17-85, 17-90, 17-91, 17-92, 17-97, 17-98, 17-99. Cables from UPS-UPS172B affected are 12B-100, 12B-101, 12B-68, 12B-69, 12B-70, 12B-71, 12B-97, 12B-98, 12B-99, 17-85, 17-90, 17-91, 17-92, 17-93, 17-94, 17-95, 17-96, 17-97, 17-98, 17-99.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-017	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for postulated fire in this area for process monitoring associated with Reactor Pressure indication. Reactor pressure indication is a required indication to assure reactor vessel parameters are maintained within acceptable bands during the shutdown process.
All channels of Reactor Pressure indication may be lost due to cable damage or loss of power supply as follows:
PI-36-25 (Train 11): Fire damage to cable RSP-25
PI-36-31A (Train 11): Loss of Power Supply to PB-RPS-11 from UPS-UPS-162A and UPS-UPS162B. Cables from UPS-UPS-162A affected are 11B-100, 11B-101, 11B-97, 11B-98, 11B-99, 16-125, 16-130, 16-131, 16-132, 16-133, 16-134, 16-135, 16-136, 16-137, 16-138, 16-139, 16-140, 16-141. Cables from UPS-UPS162B affected are 11B-100, 11B-102, 11B-97, 11B-98, 11B-99, 16-125, 16-130, 16-137, 16-138, 16-139, 16-140, 16-141, 16-142, 16-143.
PI-36-27 (Train 12): Fire damage to cable1F-156 or 1S-1748 or RSP-34 or RSP-54 or RSP-56
PI-36-32A (Train 12): Fire damage to cable 1S-669.

Ref: 51-9133191, Appendix A.8, Section A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-018	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. A loss of air fails EC Level Control Valves, LCV-60-17 & LCV-60-18, open. This could deplete the inventory in the Emergency Condenser Make-Up Water Tanks prior to the eight hours required for the EC's to remove decay heat.
Local operator action to manually throttle VLV-60-12, close VLV-60-11, and open/close BV-60-13 as required ensures availability of make-up inventory for EC's 111 & 112.

Ref: 51-9133191, Appendix A.8, Section A.8.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-019	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. Control Rod Drive Flow control valve FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

Ref: 51-9133191, Appendix A.8, Section A.8.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-020	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. RBCLC to SDC Flow control valve BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

Ref: 51-9133191, Appendix A.8, Section A.8.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-021	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. SDC heat exchanger outlet flow control valve FCV-38-10 may fail closed on loss of instrument air. Valve is required open for cold shutdown.

Ref: 51-9133191, Appendix A.8, Section A.8.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-022	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area for CTSRW pump PMP-93-01 and CTS cross connect valve FCV-93-72. CTSRW is required off for a fire in this area but has the potential to spuriously start and cross tie to the CTS system. If Train 11 power is available, an internal wire-to-wire short on cable 102-14 can spuriously start CTSRW pump PMP-93-01. An internal wire-to-wire short on cable 161-160 can spuriously open CTSRW to CTS cross connect valve FCV-93-72. The spurious start of the Path C (Train 11) CTSRW pump PMP-93-01 in conjunction with spurious opening of Path C (Train 11) FCV-93-72 causes an in inadvertent CTS actuation which results in an increasing Torus water level. The increase in Torus water level will lead the Plant operators to enter the EOPs and will cause a deviation from the credited safe shutdown strategy.

Ref: 51-9133191, Appendix A.8.5, Section A.8.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within EIR 51-9156521-000.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-023	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for postulated fire in the area for the ERV's with regards to pressure control. ERV's PSV-01-102C, PSV-01-102D, and PSV-01-102F may spuriously open resulting in blowdown to Torus

- The PSV-01-102C may spuriously open due to a cable to cable short on cable 1F-156 in conjunction with an open on cable 1F-179
- The PSV-01-102D may spuriously open due to a cable to cable short on cable 1F-156 in conjunction with an open on cable 1F-179
- The PSV-01-102F may spuriously open due to a cable to cable short on cable 1F-156 in conjunction with an open on cable 1F-179

Spurious opening of ERV's can undermine the credited HSD flow path.

Ref: 51-9133191, Appendix A.8, Section A.8.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-024	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in this area that affects the inventory control function. Cable-to-cable shorts on cables 1S-1042 and 1S-1892 cause IV-44.2-15, IV-44.2-16, IV-44.2-17 and IV-44.2-18 to spuriously open (post scram) resulting in an inventory loss.

Ref: 51-9133191, Appendix A.8, Section A.8.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-025	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for postulated fire in this area for the loss of PB-167 due to an uncoordinated associated emergency lighting circuit. Non SSD Emergency Lighting 11 is supplied by cable 167-101 from 600V Power Board 167, Breaker H01. The emergency lighting power cable 167-101 supply breaker H01 does not coordinate with PB-167 supply breaker. Fire damage to cable 167-101 could cause the loss of PB-167 preventing operation of Reactor Shutdown Cooling Isolation Valves, IV-38-01 and IV-38-13 as directed in repair procedure N1-DRP-005.

Ref: 51-9133191, Appendix A.8, Section A.8.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-026	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: The UFSAR states, "Administrative procedures have been implemented to prohibit bulk storage of combustible materials inside or adjacent to safety-related buildings or systems during operation or maintenance." The UFSAR also states, "Bulk gas storage is not permitted within structures housing safety-related equipment."

Contrary to the UFSAR statements, bulk compressed hydrogen gas, the Hydrogen Stand-By Supply, is installed on the 261'-0" elevation of the Turbine Building. No prior NRC approval of the Hydrogen Stand-By Supply location was identified.

UFSAR, Appendix 10A, Rev. 21, Sec. 2.2.2, 2.4.2.2

Disposition: This VFDR has been brought into compliance with NFPA 805 section 4.2.3, deterministic approach, via a plant modification as described in Attachment S.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-027	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: The UFSAR states, "Bulk gas storage is not permitted within structures housing safety-related equipment. Bulk Hydrogen and nitrogen storage tanks are located outside."

The hydrogen storage configurations that expose power block structures are the hydrogen tube rack located north of the Turbine Building and west of the Reactor Building, and the Hydrogen Stand-By Supply located on the 261'-0" elevation of the Turbine Building. These hydrogen storage configurations meet the applicable requirements of NFPA 50A.

The outdoor containers are pressurized horizontal cylinders and the Hydrogen Stand-By Supply containers are pressurized vertical cylinders that meet the ASME Boiler and Pressure Vessel Code. They are permanently installed on a solid concrete foundation and provided with substantial steel supports.

The outdoor containers are provided with pressure relief devices arranged to discharge above the containers, unobstructed to the open air in a manner to prevent impingement of escaping gas upon the containers, adjacent structures, or personnel. The pressure relief devices and vent piping are oriented or capped so that moisture cannot collect and freeze in a manner that would interfere with proper operation of the devices.

The Hydrogen Stand-By Supply cylinders have the pressure relief device designed and manufactured into the cylinder valve to meet NFPA 50A requirements.

The small amount of piping, tubing, and fittings that comprise the manifold systems are a combination of flanged, threaded, or welded construction suitable for hydrogen service and the pressures and temperatures involved. Cast-iron pipe and fittings are not used.

Valves, gauges, regulators and other accessories are provided by the equipment manufacturer and/or hydrogen supplier. Periodic refilling of the outdoor tube rack is performed by trained personnel familiar with proper practices and the storage installation. Outdoor storage containers, piping, valves, regulating equipment and other accessories are readily accessible and are protected by chain link fencing. Neither hydrogen supply is located near electrical hazards, near flammable or combustible gaseous or liquids, or other hazardous materials. The outdoor tube rack is readily accessible for emergency vehicles. There is no dry vegetation or other combustibles in the vicinity of the tube rack.

The outdoor tube rack and its unloading connections are located greater than the 5-ft. minimum distance allowed by NFPA 50A from the Turbine and Reactor Buildings. The six (6) storage cylinders that comprise the Hydrogen Stand-By Supply are located on the 261'-0" elevation of the Turbine Building.

There is no requirement for operation of equipment by the user that would warrant special operating instructions. The equipment is inspected and maintained by qualified personnel during system replacement and when inspection and maintenance is warranted.

Although the Hydrogen Stand-By Supply is located within the Turbine Building, the maximum total contained volume of hydrogen is less than 3500 scf. On this basis, Section 3-2.1 and Table 3-2.1 of NFPA 50A-1999 permit storage "inside buildings not in a special room or exposed to other occupancies."

UFSAR Appendix 10A, Rev. 21, Sec. 2.4.2.2

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

NFPA 50A-1999, "Standard for Gaseous Hydrogen Storage at Consumer Sites"

Disposition: This VFDR has been brought into compliance with NFPA 805 section 4.2.3, deterministic approach, via a plant modification as described in Attachment S.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-028	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of AC power and battery charging capability due to damage to the following cables:

Train 11: 101-66, 101-6, 1A-147, 102-43, 102-49, 11B-28, 1A-124, 1A-147, 102-1, 102-2, 102-29, 102-42, 102-47, 102-51, 102-52, 102-56, 11B-28, 11B-82, 11B-83, 1A-116, 101-5, 16-3, 16-119, 16-120, 16-123, 16-124, 16-125, 16-20, 16-21, 16-27

Train 12: 101-64, 1A-161, 103-29, 103-49, 12B-28, 1A-134, 1A-161, 103-1, 103-2, 1A-130, 101-65, 101-87, 17-3, 17-10, 17-12, 17-35, 17-79, 17-80, 17-83, 17-84, 17-85

As a result, battery loads are shed per procedure N1-SOP-21.1 to extend battery capability.

Ref: 51-9133191, Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-029	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Fire damage to control power cables 12B-68, 12B-69, 12B-70 or 12B-71 can cause a loss of control power to credited power board PB 17. PB 17 control power is restored via N1-DRP-GEN-005, Attachment 3 or 4 or 5.

Ref: 51-9133191, Appendix A.8.7 Report of Procedure Directed Operator Actions and Table A.8.2 Fire Area Cable Assessment Report

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-030	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Fire damage to credited cooling water pump PMP-79-54 cable 171-63 or 171-66 adversely affects credited EG-EDG103 operation. EG-EDG103 cooling water is restored using the DFP via N1-DRP-GEN-005, Attachment 5.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-031	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of credited Train 12 EG-EDG103 ventilation fans BLWROT-209-5 and BLWROT-209-6 and prevent door DOOR-D-034 from opening. Affected cables are:

BLWROT-209-5: 11DV-34, 11DV-57, 11DV-57B, 11DV-58, 11DV-69, 171-71, 171-72, FP-584, FP-588, FP-598
BLWROT-209-6: 11DV-34, 11DV-57, 11DV-57B, 11DV-58, 11DV-69, 171-73, 171-74, FP-584, FP-588, FP-598
DOOR-D-034: 11DV-34, 11DV-57, 11DV-57B, 11DV-58, 171-81, 171-74, FP-584, FP-588, FP-598
CARDOX/HALON-PURGE: FP-545, FP-585, FP-590, FP-591, FP-599, FP-600

EG-EDG103 roof exhaust fans BLWROT-209-5 and BLWROT-209-6 are repaired and operated per N1-DRP-GEN-005, Attachment 6. As recommended by NEI 04-02, this action shall be classified as a recovery action and subsequently evaluated in the Fire Risk Evaluation for this fire area.

DOOR-D-034 is manually opened per N1-SOP-21.1.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a plant modification, as described in Attachment S, to reduce risk. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-032	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Fire damage to credited ESW pump PMP-72-03 (Cable 12B-68, 12B-69, 12B-70, 12B-71 or 17-37) adversely affects decay heat removal using SDC. ESW is restored using the DFP via N1-DRP-GEN-005, Attachment 5 or Attachment 7.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-033	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of credited EG-EDG103 cooling water pump PMP-79-54 and/or ESW pump PMP-72-03. The DFP is credited to supply EG-EDG103 cooling water and/or ESW (See VFDR-05-030 and VFDR-05-032). The DFP control and control power supply cables are located in the fire area of concern.

Ref: 51-9133191, Appendix A.8.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-034	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause isolation of the normal reactor inventory makeup line from CRD due to fire damage to cable 1S-2586 (FCV-44-149 and E/P-113-301), or cable 155-71 (PCV-44-04) or cable 155-74 (PCV-44-05). To support the reactor vessel inventory makeup function, CRD flow is controlled manually by operating VLV-28-18 locally per Procedure N1-SOP-21.1.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-035	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of AC power and battery charging capability due to damage to the following cables:

Train 11: 101-66, 101-6, 1A-147, 102-43, 102-49, 11B-28, 1A-124, 1A-147, 102-1, 102-2, 102-29, 102-42, 102-47, 102-51, 102-52, 102-56, 11B-28, 11B-82, 11B-83, 1A-116, 101-5, 16-3, 16-119, 16-120, 16-123, 16-124, 16-125, 16-20, 16-21, 16-27

Train 12: 101-64, 1A-161, 103-29, 103-49, 12B-28, 1A-134, 1A-161, 103-1, 103-2, 1A-130, 101-65, 101-87, 17-3, 17-10, 17-12, 17-35, 17-79, 17-80, 17-83, 17-84, 17-85

Credited battery BAT-B12 charging capability is restored via N1-DRP-GEN-005, Attachment 1 or 2 or 3 or 4 or 5 using Computer MG set 167 or Attachment 7 using battery chargers BC-B11-1, BC-B11-2, BC-B12-1 or BC-B12-2. However, MG Set 167 control panel PNL-CPB167 and associated cables are located in this fire area.

Ref: 51-9133191, Appendix A.8.4 Separation Concerns and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 5
Fire Area Description TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-036	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Restoration of control power to PB 17 per N1-DRP-GEN-005 coupled with control cable fire damage to breakers R1052 (Cable 17-2) & R1051 (Cable 17-1) and offsite power available may result in tying PB 17A & PB 17B together out of phase affecting PB 17 availability.

Ref: 51-9133191, Appendix A.8.4 Separation Concerns and Table A.8.2 Fire Area Cable Assessment Report

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-037	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may adversely affect credited EG-EDG103 Breaker R1032 due to damage to the following cables: 101-64, 103-49, 12B-28, 1A-134 or 1A-161. EG-EDG103 is required to support various performance goals. The cable damage is repaired via N1-DRP-GEN-005, Attachment 1 or 2 or 4 or 6 and the breaker operated per N1-OP-45, Section H.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-038	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may adversely affect credited EG-EDG103 due to fire damage to cable 103-49, 103-29, 12B-28, or 1A-134. EG-EDG103 is required to support various performance goals. EG-EDG103 is repaired via N1-DRP-GEN-005, Attachment 1 and operated per N1-OP-45, Section H.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-039	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may adversely affect credited EG-EDG103 due to damage to its neutral breaker control cable 1A-130. EG-EDG103 is required to support various performance goals. The EG-EDG103 neutral breaker is repaired and closed per N1-DRP-GEN-005, Attachment 1.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-040	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may adversely affect the credited Train 12 power system which is required to support various performance goals. Fire damage is related to the following components and cables:

BKR-(101/2A-1)R013/161: 1A-161 and 101-64
BKR-(103/1-9)R1031/181: 103-2
BKR-(17/007B)R1052/612: 17-2
BKR-(17/002B)R1053/613: 17-3

The breakers are repaired and operated per N1-DRP-GEN-005, Attachment 1 or 2 or 4 or 6.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-041	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A fire in this area may cause a loss of normal AC power and a loss of instrument air. To support various shutdown performance requirements, AC power is required. In order to facilitate starting EG-EDG103, EG-EDG103 starting air is restored via N1-DRP-GEN-005, Attachment 1 and EG-EDG103 is operated per N1-OP-45, Section H.

Ref: 51-9133191, Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-042	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may damage credited Train 12 control power cable 12B-28 adversely affecting various PB 12 components. Electrical Train 12 is required to support various performance goals. Train 12 control power is isolated via N1-DRP-GEN-005, Attachment 2.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-043	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may adversely affect credited Train 12 power availability via damage (fault) to cable 101-87 that feeds PB 103. Train 12 power is required to support various performance goals. Cable 101-87 is disconnected from PB 103 per procedure N1-DRP-GEN-005, Attachment 6 to allow EG-EDG103 to supply PB 103.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-044	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may damage cable 171-41 adversely affecting credited SDC valve BV-38-04. SDC is required to support the decay heat removal function. Local-Manual operation of BV-38-04 per N1-SOP-21.1 may be required.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
5	TURBINE BUILDING EL 240-0 THRU 369-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-045	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may damage cable 12DV-10, 12DV-11, 12DV-29, 12DV-9 or 167-11 adversely affecting credited SDC valve IV-38-02. SDC is required to support the decay heat removal function. Local-Manual operation of IV-38-02 per N1-SOP-21.1 may be required.

Ref: 51-9133191, Table A.8.2 Fire Area Cable Assessment Report and Table A.8.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-05-046	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in this area that affects the reactivity control performance goal for reactor scram. Cable-to-cable shorts on cables 1S-1042 and 1S-1892 impact Channel 12 scram solenoids SOV-113-271, SOV-113-274 and SOV-113-276 associated with IV-44.2-15, IV-44.2-16, IV-44.2-17 and IV-44.2-18. The SOV's remaining energized prevents venting the scram air header which, in turn, prevents reactor scram from the control room.

Ref: 51-9133191, Appendix A.8, Section A.8.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within EIR 51-9156521-000.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 6
Fire Area Description TURBINE BUILDING NORTH EL 250-0

Fire Zone T2A
Fire Zone Description TURBINE BUILDING EL 250-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	The Train 11 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path B HSD is credited via use of EC's 121 & 122 for Decay Heat Removal and Train 12 support systems. Path A is credited to achieve CSD. Path A CSD is accomplished via the use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

(f) Vital Auxiliaries	1) AC and DC Distribution System The Train 11 emergency diesel generator is credited for supplying power for the shutdown process when AC power is required to operate shutdown equipment. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
6 | **TURBINE BUILDING NORTH EL 250-0**

Path C – Train 11 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-03-002 Rev. 0

Eng Eval Summary: FPEE 0-03-002 Rev. 0 - Evaluation of Interim Action To Prevent Personnel Injury From CO2
Evaluation justifies use of temporarily placing automatic CO2 systems in alarm-only mode due to adverse trend of unplanned system actuations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
T2A	Detection	DA-2013N	N	N	Y	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
	Detection	DA-2013S	N	N	Y	Y	N	
	Suppression	WP-2013	N	N	N	Y	N	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA6 of a Fire Originating In FA6
The activation of a water-based suppression system would not adversely impact equipment/components credited by the NSCA in the Turbine Building.

Scenario 2: Suppression Affects in FA6 of a Fire Originating Outside of FA6
Based on the limited impact a fire can have on plant suppression systems, the operation of fire suppression systems outside of FA6 would not be expected to cause activation of a fire protection system within FA6 that could have an impact on the nuclear safety performance criteria.

Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

ΔLERF: Redacted

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems for partial coverage and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between EC system isolation valves IV-39-11R and IV-39-12R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-114 for each valve can maintain EC system isolation valves IV-39-11R and IV-39-12R open

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

resulting in an inventory loss to the High Pressure Section of the Condenser at penetration 15.

51-9133191 Appendix A.9, Section A.9.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between EC system isolation valves IV-39-13R and IV-39-14R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-120 for each valve can maintain EC system isolation valves IV-39-13R and IV-39-14R open resulting in an inventory loss to the High Pressure Section of the Condenser at penetration 15.

51-9133191 Appendix A.9, Section A.9.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between EC system isolation valves IV-05-04R, IV-05-12, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-159 for each valve can maintain EC system vent valves IV-05-04R and IV-05-12 open. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-22 for each valve can maintain EC system isolation valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line.

51-9133191 Appendix A.9, Section A.9.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between EC system isolation valves IV-05-01R, IV-05-11, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-158 for each valve can maintain EC system isolation valves IV-05-01R and IV-05-11 open. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-22 for each valve can maintain EC system isolation valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line.

51-9133191 Appendix A.9, Section A.9.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between EC system isolation valves IV-05-04R, IV-05-12, BV-05-05, and BV-05-07. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-159 for each valve can maintain EC system isolation valves IV-05-04R and IV-05-12 open. An internal wire-to-wire hot short on cables 167-187, for valve BV-05-05, and 167-190, for valve BV-05-07, can maintain EC system isolation valves BV-05-05 and BV-05-07 open. The multiple spurious combination results in an inventory loss to the torus.

51-9133191 Appendix A.9, Section A.9.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between EC system isolation valves IV-05-01R, IV-05-11, BV-05-5, and BV-05-7. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-158 for each valve can maintain EC system isolation valves IV-05-01R and IV-05-11 open. An internal wire-to-wire hot short on cables 167-187, for valve BV-05-05, and 167-190, for valve BV-05-07, can maintain EC system isolation valves BV-05-05 and BV-05-07 open. The multiple spurious combination results in an inventory loss to the torus.

51-9133191 Appendix A.9, Section A.9.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between Inboard MSIV IV-01-01 and Outboard MSIV IV-01-03. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the power supply or control circuits of Inboard MSIV IV-01-01 and Outboard MSIV IV-01-03 leaves the MS line open. Inboard MSIV IV-01-01 may remain open due to a loss of the Train 11 AC electrical system below 4 kV. An external cable-to-cable short to cable 1F-38 or 1F-51 (125 VDC) results in preventing closure of series Outboard MSIV IV-01-03.

51-9133191 Appendix A.9, Section A.9.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between Inboard MSIV IV-01-02 and Outboard MSIV IV-01-04. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the power supply or control circuits of Inboard MSIV IV-01-02 and Outboard MSIV IV-01-04 leaves the MS line open. Inboard MSIV IV-01-02 may remain open due to a loss of the Train 12 AC electrical system below 4 kV. An external cable-to-cable short to cable1F-44, 1F45, 1F-56 or 1F-57 (125 VDC) results in preventing closure of series Outboard MSIV IV-01-04.

51-9133191 Appendix A.9, Section A.9.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-009	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-010	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between Train 11 and 12 power systems. AC and DC Power is a Vital Auxiliary system required to support the nuclear safety performance goals for CSD. The Power cables associated with the credited Train 11 power supply are impacted in this area and can fail the credited 600 Vac system. In addition, Train 12 power cables are also impacted in this area. Power is restored to Power Board 16B per procedure N1-DRP-GEN-001, "Fire Zones T2A and T2D Turbine Building Elevation 250' Detectors DA-2013S, DA-2013N and DA-2031."

51-9133191 Appendix A.9, Section A.9.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-011	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. A loss of instrument air could fail the decay heat removal capability by limiting the availability of the emergency condenser make up tanks from the required eight hours to some time frame less than 8 hours. LCV-60-17 and LCV-60-18 fail open on loss of instrument air. This failure causes a run out concern for the inventory contained in the emergency condenser make up tanks prior to the eight hours required for EC's to remove decay heat. In addition, BV-60-13, which provides a cross tie between the EC Make up Tanks, fails closed on loss of air. Based on NMP1 design bases document, SDBD-204 and NMP1 UFSAR Chapter V, Section E, Dated October 2001, the level control valves and the cross tie valve are required to be functional to meet the required eight hours.

51-9133191 Appendix A.9, Section A.9.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-012	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. Flow control valve FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

51-9133191 Appendix A.9, Section A.9.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-013	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. Flow control valve BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

51-9133191 Appendix A.9, Section A.9.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-014	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. SDC heat exchanger outlet flow control valve FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

51-9133191 Appendix A.9, Section A.9.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-015	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. SDC heat exchanger outlet flow control valve FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

51-9133191 Appendix A.9, Section A.9.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-016	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area for the credited Train 12 EC Loop Makeup capability. EC Makeup is required to ensure the emergency condensers are available for eight hours. Fire damage to cable 1S-272 may cause a false signal from level transmitter LT-60-29 that closes level control valve LCV-60-18. Closure of level control valve LCV-60-18 will isolate the EC Makeup Tank from the emergency condensers. An operator action is required to control makeup to EC's 121 & 122.Fire damage to cable 1S-272 results in a loss of EC 121 & 122 level indication in the MCR. EC 121 & 122 level indication is available at the RSP.

51-9133191 Appendix A.9, Section A.9.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-017	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of AC power and battery charging capability. Battery loads are shed per procedure N1-SOP-21.1 to extend battery capability.

Ref: 51-9133191, Table A.9.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
6	TURBINE BUILDING NORTH EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-06-018	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may result in the loss of Train 12 power adversely affecting credited SDC valve IV-38-02. SDC is required to support the decay heat removal function. Local-Manual operation of IV-38-02 per N1-SOP-21.1 may be required.

Ref: 51-9133191, Table A.9.3 Fire Area Cascading Power Loss Report - SSD and Table A.9.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
7	**TURBINE BUILDING SOUTH & WEST EL 250-0**

Fire Zone	Fire Zone Description
T2B	TURBINE BUILDING SOUTH AND WEST EL 250-0
T2E	UPS BATTERY ROOM EL 250

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(b) Inventory Control Function	The Train 12 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(d) Decay Heat Removal Function	Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 support systems. Path B is credited to achieve CSD. Path B CSD is accomplished via use of the Train 12 SDC System supported by the Train 12 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

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Fire Area	Fire Area Description
7	TURBINE BUILDING SOUTH & WEST EL 250-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure o Emergency Condenser Level	No comments.

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Fire Area	Fire Area Description
7	TURBINE BUILDING SOUTH & WEST EL 250-0

(f) Vital Auxiliaries	1) AC and DC Distribution System The Train 12 emergency diesel generator is credited for supplying power for the shutdown process when AC power is required to operate shutdown equipment. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

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Fire Area 7
Fire Area Description TURBINE BUILDING SOUTH & WEST EL 250-0

Path D – Train 12 positive pressure flow path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
T2B	Detection	DA-2013N	N	N	N	Y	N	
	Detection	DA-2022N	N	N	N	Y	N	
	Detection	DA-2022S	N	N	N	Y	N	
	Detection	DA-2051E	N	N	N	Y	N	
	Detection	DA-2051W	N	N	N	Y	N	
	Suppression	WP-2022	N	N	N	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
7	TURBINE BUILDING SOUTH & WEST EL 250-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
	Suppression	WP-2051	N	N	N	Y	N	
T2E	Detection	DA-2022S	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA7 of a Fire Originating In FA7
The activation of a water-based suppression system would not adversely impact equipment/components credited by the NSCA in the Turbine Building.

Scenario 2: Suppression Affects in FA7 of a Fire Originating Outside of FA7
Based on the limited impact a fire can have on plant suppression systems,the operation of fire suppression systems outside of FA7 would not be expected to cause activation of a fire protection system within FA7 that could have an impact on the nuclear safety performance criteria.

Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
7	TURBINE BUILDING SOUTH & WEST EL 250-0

ΔLERF: Redacted

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems for partial coverage and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the Inboard MSIV IV-01-01 and Outboard MSIV IV-01-03. These valves are required closed for HSD and CSD to

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
7	TURBINE BUILDING SOUTH & WEST EL 250-0

support inventory control. Fire damage to the control circuits of Inboard MSIV IV-01-01 and Outboard MSIV IV-01-03 leaves the MS line open. Inboard MSIV IV-01-01 can suffer an internal wire-to-wire short on cable 161-50 maintaining the valve open. An external cable-to-cable short to cable 1F-38, 1F-39, 1F-51, or 1F-52 (125 VDC) results in preventing closure of series Outboard MSIV IV-01-03.

Ref: 51-9133191000 Appendix A.10, Section A.10.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the Inboard MSIV IV-01-02 and Outboard MSIV IV-01-04. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the power supply or control circuits of Inboard MSIV IV-01-02 and Outboard MSIV IV-01-04 leaves the MS line open. Inboard MSIV IV-01-02 may remain open due to a loss of the Train 12 AC electrical system below 4 kV. An external cable-to-cable short to cable 1F-45 or 1F-56 (125 VDC) results in preventing closure of series Outboard MSIV IV-01-04.

Ref: 51-9133191000 Appendix A.10, Section A.10.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for shutdown cooling. Credited pump, PMP-38-152, is required to operate to support decay heat removal. Fire damage to cable 17-62 can prevent remote start of the credited SDC pump due to loss of RCS Temperature Switch permissive. Local breaker operation is required.

Ref: 51-9133191000 Appendix A.10, Section A.10.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
7	TURBINE BUILDING SOUTH & WEST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting Train 11 and 12 power systems. AC and DC Power is a Vital Auxiliary system required to support the nuclear safety performance goals for CSD. Control power cables, 12B-83 and 12B-84, associated with the credited Train 12 power supply (EG-EDG103) are impacted in this area and can fail the credited 4 kV AC system. In addition, Train 11 major power cables 101-5 feeding PB-102 and 102-1 feeding PB-16B are also impacted in this area and can fail 4KV and 600VAC system.

Ref: 51-9133191000 Appendix A.10, Section A.10.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in this area that affects the inventory control function. Cable-to-cable shorts on cables 1S-1041 and 1S-1891 cause IV-44.2-15, IV-44.2-16, IV-44.2-17, and IV-44.2-18 to spuriously open (post scram) resulting in an inventory loss.

Ref: 51-9133191000 Appendix A.10, Section A.10.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
7	TURBINE BUILDING SOUTH & WEST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in the area for the ERV's with regards to pressure control. ERV's PSV-01-102A, PSV-01-102B and PSV-01-102E may spuriously open. PSV-01-102A may spuriously open due to a cable-to-cable short on cable 1F-151 in conjunction with an open in cable 1F-178. PSV-01-102B may spuriously open due to a cable-to-cable short on cable 1F-151 in conjunction with an open in cable 1F-178. PSV-01-102E may spuriously open due to a cable-to-cable short on cable 1F-151 in conjunction with an open in cable 1F-178. The alternate cooling path using CS may not be available due to the loss of both electrical trains.

Ref: 51-9133191000 Appendix A.10, Section A.10.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any area of the plant. A loss of instrument air could fail the decay heat removal capability by limiting the availability of the emergency condenser make up tanks from the required eight hours to some time frame less than 8 hours. LCV-60-17 and LCV-60-18 fail open on loss of instrument air. This failure causes a run out concern for the inventory contained in the emergency condenser make up tanks. In addition, BV-60-13, which provides a cross tie between the EC Make up Tanks, fails closed on loss of air. Based on NMP1 design bases document, SDBD-204 and NMP1 UFSAR Chapter V, Section E, Dated October 2001, the level control valves and the cross tie valve are required to be functional to meet the required eight hours.

Ref: 51-9133191000 Appendix A.10, Section A.10.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
7	TURBINE BUILDING SOUTH & WEST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any area of the plant. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open Manual Bypass Valve VLV-28-18 locally to provide reactor makeup from the CRD system.

Ref: 51-9133191000 Appendix A.10, Section A10.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-009	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any area of the plant. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53 locally to supply RBCLC water to the SDC heat exchanger.

Ref: 51-9133191000 Appendix A.10, Section A10.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-010	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any area of the plant. FCV-38-10 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-10 locally to control SDC cooldown.

Ref: 51-9133191000 Appendix A.10, Section A.10.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
7	TURBINE BUILDING SOUTH & WEST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-011	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-012	Closed		N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of AC power and battery charging capability. Battery loads are shed per procedure N1-SOP-21.1 to extend battery capability.

Ref: 51-9133191, Table A.10.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-07-013	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in this area that affects the reactivity control performance goal for reactor scram. Cable-to-cable shorts on cables 1S-1041 and 1S-1891 impact Channel 11 scram solenoids SOV-113-272, SOV-113-273 and SOV-113-275 associated with IV-44.2-15, IV-44.2-16, IV-44.2-17, and IV-44.2-18. The SOV's remaining energized prevents venting the scram air header which, in turn, prevents reactor scram from the control room.

Ref: 51-9133191000 Appendix A.10, Section A.10.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within EIR 51-9156521-000.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	**TURBINE BUILDING EAST EL 250-0**

Fire Zone	Fire Zone Description
T2C	TURBINE BUILDING OFFGAS TUNNEL EL 250-0
T2D	TURBINE BUILDING GENERAL AREA EAST EL 250-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	The Path A (Train 11) CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 support systems. Path A is credited to achieve CSD. Path A CSD is accomplished via the use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems. Path C Control Room Ventilation is credited to support shutdown	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

(f) Vital Auxiliaries	1) AC and DC Distribution System The Train 11 emergency diesel generator is credited for supplying power for the shutdown process when AC power is required to operate shutdown equipment. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 9

Fire Area Description TURBINE BUILDING EAST EL 250-0

Path C – Train 11 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
T2C	Detection	DA-2031	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
T2D	Detection	DA-2013	N	N	N	Y	N	
	Detection	DA-2031	N	N	N	Y	N	
	Suppression	WP-2031	N	N	N	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA9 of a Fire Originating In FA9
The activation of a water-based suppression system would not adversely impact equipment/components credited by the NSCA in the Turbine Building.

Scenario 2: Suppression Affects in FA9 of a Fire Originating Outside of FA9
Based on the limited impact a fire can have on plant suppression systems, operation of fire suppression systems outside of FA9 would not be expected to cause activation of a fire protection system within FA9 that could have an impact on the nuclear safety performance criteria.

Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks. See additional discussion related to this fire area in N1-FRE-F001.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems for partial coverage and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-001	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	**TURBINE BUILDING EAST EL 250-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-39-11R and IV-39-12R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-114 for each valve can maintain EC system isolation valves IV-39-11R and IV-39-12R open resulting in an inventory loss to the High Pressure Section of the Condenser at penetration 15.

Ref: 51-9133191 Appendix A.11, Section A.11.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-39-13R and IV-39-14R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-120 can maintain EC system isolation valves IV-39-13R and IV-39-14R open resulting in an inventory loss to the High Pressure Section of the Condenser at penetration 15.

Ref: 51-9133191 Appendix A.11, Section A.11.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-05-04R, IV-05-12, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-159 can maintain EC system isolation valves IV-05-04R and IV-05-12 open. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-22 can maintain EC system isolation valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line. All cables are 125VDC.

Ref: 51-9133191 Appendix A.11, Section A.11.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-05-01R, IV-05-11, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-158 for each valve can maintain EC system isolation valves IV-05-01R and IV-05-11 open. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-22 can maintain EC system isolation valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line. All cables are 125VDC.

Ref: 51-9133191 Appendix A.11, Section A.11.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-05-04R, IV-05-12, BV-05-05, and BV-05-07. These valves are required closed for HSD to support inventory control. A cable-to-cable and an internal wire-to-wire hot short on cable 1K-159 can maintain EC system isolation valves IV-05-04R and IV-05-12 open. An internal wire-to-wire hot short on cables 167-187, for valve BV-05-05, and 167-190, for valve BV-05-07, can maintain EC system isolation valves BV-05-05 and BV-05-07 open. The multiple spurious combination results in an inventory loss to the torus. All cables are 125VDC.

Ref: 51-9133191 Appendix A.11, Section A.11.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-05-01R, IV-05-11, BV-05-5, and BV-05-7. These valves are required closed for HSD to support inventory control. A cable-to-cable or an internal wire-to-wire hot short on cable 1K-158 can maintain EC system isolation valves IV-05-01R and IV-05-11 open. An internal wire-to-wire hot short on cables 167-187, for valve BV-05-05, and 167-190, for valve BV-05-07, can maintain EC system isolation valves BV-05-05 and BV-05-07 open. The multiple spurious combination results in an inventory loss to the torus. All cables are 125VDC.

Ref: 51-9133191 Appendix A.11, Section A.11.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	**TURBINE BUILDING EAST EL 250-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting Inboard MSIV IV-01-02 and Outboard MSIV IV-01-04. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the power supply or control circuits of Inboard MSIV IV-01-02 and Outboard MSIV IV-01-04 leaves the MS line open. Inboard MSIV IV-01-02 may remain open due to a loss of the Train 12 AC electrical system below 4 kV. An external cable-to-cable short to cable 1F-56 (125 VDC) results in preventing closure of series Outboard MSIV IV-01-04.

Ref: 51-9133191 Appendix A.11, Section A.11.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-009	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-010	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-011	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. A loss of instrument air could fail the decay heat removal capability by limiting the availability of the Emergency Condenser make-up tanks from the eight hours to some time frame less than 8 hours. A loss of air fails EC Level Control Valves, LCV-60-17 & LCV-60-18, open. This could deplete the inventory in the Emergency Condenser Make-Up Water Tanks prior to the eight hours required for the EC's to remove decay heat. Local operator action to manually throttle VLV-60-11, close VLV-60-12, and open/close BV-60-13 as required ensures availability of make-up inventory for EC's 121 & 122.

Ref: 51-9133191 Appendix A.11, Section A.11.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-012	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. Flow control valve FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

Ref: 51-9133191 Appendix A.11, Section A.11.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-013	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. Flow control valve BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A recovery action may be required to open valve BV-70-53 locally to supply RBCLC water to SDC heat exchanger.

Ref: 51-9133191 Appendix A.11, Section A.11.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-014	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. SDC heat exchanger outlet flow control valve FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

Ref: 51-9133191 Appendix A.11, Section A.11.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-015	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. SDC heat exchanger outlet flow control valve FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

Ref: 51-9133191 Appendix A.11, Section A.11.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-016	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Separation concern exists for postulated fire in this area for the credited Train 12 EC Loop Make-up capability. EC make-up capability is required to ensure the Emergency Condensers are available for eight hours. Fire damage to cable 1S-272 will result in a loss of EC 121 & 122 level indication in the MCR. EC 121 and 122 level indication, however, is available at the Remote Shutdown Panel (RSP).

Ref: 51-9133191 Appendix A.11, Section A.11.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-017	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Separation concern exists for postulated fire in this area for the credited Train 12 EC's 121&122 Loop Make-up capability. EC make-up capability is required to ensure the Emergency Condensers are available for eight hours. Fire damage to cable 1S-272 may cause a false signal that closes LCV-60-18. Operator action is required to control makeup to EC's 121& 122.

Ref: 51-9133191 Appendix A.11, Section A.11.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-018	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of AC power and battery charging capability. Battery loads are shed per procedure N1-SOP-21.1 to extend battery capability.

Ref: 51-9133191, Table A.11.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-019	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may adversely affect credited Train 11 power availability via damage (fault) to cable 101-5 that feeds PB 102. Cable 101-5 is disconnected from PB 102 per procedure N1-DRP-GEN-001 to allow EG-EDG102 to supply PB 102.

Ref: 51-9133191, Table A.11.2 Cable Assessment Report and Table A.11.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
9	TURBINE BUILDING EAST EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-020	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Power cable 102-1 associated with the credited Train 11 power supply to PB 16B is impacted in this area and can fail the credited 600 VAC system. Power is restored to PB 16B per procedure N1-DRP-GEN-001, "Fire Zones T2A and T2D Turbine Building Elevation 250' Detectors DA-2013S, DA-2013N and DA-2031".

Ref: 51-9133191, Table A.11.2 Cable Assessment Report and Table A.11.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-09-021	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may result in the loss of Train 12 power adversely affecting credited SDC valve IV-38-02. SDC is required to support the decay heat removal function. Local-Manual operation of IV-38-02 per N1-SOP-21.1 may be required.

Ref: 51-9133191, Table A.11.3 Fire Area Cascading Power Loss Report - SSD and Table A.11.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	**CABLE SPREADING ROOM EL 250-0**

Fire Zone	Fire Zone Description
C1	CABLE SPREADING ROOM EL 250-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(b) Inventory Control Function	The Train 11 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(d) Decay Heat Removal Function	Path A HSD is credited via use of EC's 111 & 112 for Decay Heat Removal and Train 11 support systems. Alternatively, Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 Support Systems. Path A is credited to achieve CSD. Path A CSD is accomplished via the use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 10

Fire Area Description CABLE SPREADING ROOM EL 250-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Path A HSD is generally supported by the Train 11 DC power systems. Path B HSD is generally supported by the Train 12 DC power systems. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. Path A CSD is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by Train 12 AC and/or DC power system. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC SystemsHVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

Path C – Train 11 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-03-002 Rev. 0

Eng Eval Summary: FPEE 0-03-002 Rev. 0 - Evaluation of Interim Action To Prevent Personnel Injury From CO2
Evaluation justifies use of temporarily placing automatic CO2 systems in alarm-only mode due to adverse trend of unplanned system actuations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
C1	Detection	DX-3011A/B	N	N	Y	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
10 | **CABLE SPREADING ROOM EL 250-0**

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
	Suppression	C-3011	N	N	N	N	Y	
	Suppression	WP-3011	N	N	Y	Y	N	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA10 of a Fire Originating In FA10
The activation of a water-based suppression system would not adversely impact equipment/components credited by the NSCA in the Cable Spreading Room. Additionally, the CO2 system installed in this area is isolated by a manual valve and set to "Alarm Only", so a CO2 system actuation which could adversely impact equipment/components credited by the NSCA would not occur.

Scenario 2: Suppression Affects in FA10 of a Fire Originating Outside of FA10
Based on the limited impact a fire can have on plant suppression systems, operation of fire suppression systems outside of FA10 would not be expected to cause activation of the preaction sprinkler system within FA10 that could have an impact on the nuclear safety performance criteria. Additionally, based on the fact that the CO2 system in FA10 is manually locked out, operation of fire suppression systems outside of FA10 would not be expected to cause activation of the CO2 system within FA10 that could have an impact on the nuclear safety performance criteria.

Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied. Note that VFDR-10-021 was partially resolved by crediting a modification to the plant.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks. See additional discussion related to this fire area in N1-FRE-F001.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems for full coverage (sprinkler system) and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR-10-001	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for Main Feedwater. Main Feedwater is required to be isolated for inventory control. Main Feedwater valves IV-31-07 and IV-31-08 may remain open due to fire damage to the following cables:

IV-31-07: 101-5, 161-57
IV-31-08: 103-18, 103-22, 103-28, 103-43, 103-44, 103-47, 103-66, 171-57

Main Feedwater pumps may spuriously start. Pump 29-02 may start due to an internal wire-to-wire short on cable 11-78. Pump 29-03 may spuriously start due to an internal wire-to-wire short on cable 12-78. The clutch of turbine shaft driven pump 29-01 may remain engaged due to shorts on cable 1F-110 or fire damage to any of the following cables 1F-110, 1S-1610, 1S-1730, 1S-1735, or 1S-1736. The result is reactor vessel overfill that has the potential to compromise the ability of the credited EC to achieve and maintain HSD.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-05-04R, IV-05-12, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-159 can maintain EC system isolation valve IV-05-12 open. A cable-to-cable hot short on cable 1K-159 or 1K-34 can maintain EC system isolation valve IV-05-04R open. A cable-to-cable hot short on cable 1K-22 for each valve can maintain EC system isolation valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-05-04R, IV-05-12, BV-05-05, and BV-05-07. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-159 can maintain EC system isolation valve IV-05-12 open. A cable-to-cable hot short on cable 1K-159 or 1K-34 can maintain EC system isolation valve IV-05-04R open. An internal wire-to-wire hot short on cable 167-187 can maintain EC system isolation valve BV-05-05 open. An internal wire-to-wire hot short on cable 167-190 can maintain EC system isolation valve BV-05-07 open. The multiple spurious combination results in an inventory loss to the torus.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-05-01R, IV-05-11, BV-05-05, and BV-05-07. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-158 or 1S-2017 or an internal wire-to-wire hot short on cable 1K-18 can maintain EC system isolation valve IV-05-01R open. Either an internal wire-to-wire short on cable 1K-18 or a cable-to-cable short on cable 1K-158 can maintain EC system isolation valve IV-05-11 open. An internal wire-to-wire hot short on cable 167-187 can maintain EC system isolation valve BV-05-05 open. An internal wire-to-wire hot short on cable 167-190 can maintain EC system isolation valve BV-05-07 open. The multiple spurious combination results in an inventory loss to the torus.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-05-01R, IV-05-11, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-158 or 1S-2017 or an internal wire-to-wire hot short on cable 1K-18 can maintain EC system isolation valve IV-05-01R open. Either an internal wire-to-wire short on cable 1K-18 or a cable-to-cable short on cable 1K-158 can maintain EC system isolation valve IV-05-11 open. A cable-to-cable hot short on cable 1K-22 for each valve can maintain EC system isolation valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-39-11R and IV-39-12R. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-114 for each valve can maintain EC system isolation valves IV-39-11R and IV-39-12R open resulting in an inventory loss to the Main Steam drain line.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-39-13R and IV-39-14R. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-120 for each valve can maintain EC system isolation valves IV-39-13R and IV-39-14R open resulting in an inventory loss to the Main Steam drain line.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-009	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting Inboard MSIV IV-01-01 and Outboard MSIV IV-01-03. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the power supply or control circuits of Inboard MSIV IV-01-01 and Outboard MSIV IV-01-03 leaves the MS line open. Inboard MSIV IV-01-01 may remain open due to a loss of the Train 11 AC electrical system below 4 kV. An internal wire-to-wire short to cable 1F-37 or 1F-38 (125 VDC) results in preventing closure of series Outboard MSIV IV-01-03.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-010	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting Inboard MSIV IV-01-02 and Outboard MSIV IV-01-04. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the power supply or control circuits of Inboard MSIV IV-01-02 and Outboard MSIV IV-01-04 leaves the MS line open. Inboard MSIV IV-01-02 may remain open due to a loss of the Train 12 AC electrical system below 4 kV. An internal wire-to-wire short to cable 1F-55 or 1F-56 (125 VDC) results in preventing closure of series Outboard MSIV IV-01-04.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-011	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area for the Shutdown Cooling system. SDC valve IV-38-01 is required open for CSD to support decay heat removal. Fire damage (ground) to cable 12DV-29 prevents SDC IV-38-01 from opening. When power is restored to the valve and the control switch operated, the control circuit fuse will blow due to a dead short across the CPT.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-012	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the 11 and 12 Emergency Condenser Loops. EC Loop 11 or 12 is required to ensure decay heat removal.

Path A - EC's 111 & 112 are not available due to:
An internal wire-to-wire short in conjunction with an open wire on either cable 171-194 or 171-195 closes IV-39-09R isolating EC's 111 & 112.
Internal wire-to-wire shorts (2) on cable RSP-82, or an internal wire-to-wire short on cable 11DV-92B in conjunction with an internal wire-to-wire short on cable RSP-82 will close IV-39-07R.
Fire damage to cable 1S-352 may cause a false signal that closes LCV-60-17. Operator action is required to control makeup to EC's 111 & 112.

Path B - EC's 121 & 122 are not available due to:
Internal wire-to-wire shorts (2) in conjunction with an open wire on cable 12DV-23 (125VDC) will close IV-39-08R.
Fire damage to cable 1S-272 may cause a false signal that closes LCV-60-18. Operator action is required to control makeup to EC's 121 & 122.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-013	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for process monitoring associated with Reactor Level and Reactor Pressure. Fire damage to cable 16-146 results in a loss of credited Train 11 MCR instruments. The corresponding Train 12 MCR instruments are also adversely affected due to the loss of EDG-103. Reactor vessel level is a required indication to assure adequate level is maintained in the reactor vessel during the shutdown process. Reactor pressure indication is a required indication to assure reactor vessel parameters are maintained within acceptable bands during the shutdown process.

Components affected are:
LI-36-09
LI-36-19
LI-36-43
PI-36-31A

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-014	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for battery charging capability to the credited Battery B-11. Battery B-11 is associated with vital auxiliaries to support the performance goals. A cable-to-cable short on cable 16-124 (125 VDC) can cause the feeder breaker to Battery Chargers BC-B11-1 and BC-B11-2 to open resulting in the loss of battery charging capability for credited Battery B-11. Also, an internal wire-to-wire short on cable 11B-95 (125 VDC) can cause the feeder breaker from Battery Chargers BC-B11-1 and BC-B11-2 to open resulting in the loss of battery charging capability for credited Battery B-11.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-015	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in the area for the ERV's with regards to pressure control. ERV's PSV-01-102A, PSV-01-102B, and PSV-01-102E may spuriously open. PSV-01-102A may spuriously open due to a cable-to-cable short on cable 1F-151 in conjunction with an open in cable 1F-178. PSV-01-102B may spuriously open due to a cable-to-cable short on cable 1F-151 in conjunction with an open in cable 1F-178. PSV-01-102E may spuriously open due to a cable-to-cable short on cable 1F-151 in conjunction with an open in cable 1F-178. The alternate cooling path using CS may not be available due to the loss of both electrical trains.

Ref: 51-9133191 Appendix A.12, Section A12.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-016	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in this area that affects the inventory control function. Cable-to-cable shorts on cables 1S-1041 and 1S-1891 cause IV-44.2-15, IV-44.2-16, IV-44.2-17, and IV-44.2-18 to spuriously open (post scram) resulting in an inventory loss.

Ref: 51-9133191 Appendix A.12, Section A12.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-017	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. A loss of instrument air could fail the decay heat removal capability by limiting the availability of the emergency condenser make up tanks from the required eight hours to some time frame less than 8 hours. EC level control valves LCV-60-17 and LCV-60-18 fail open on loss of instrument air. This failure causes a run out concern for the inventory contained in the emergency condenser make up tanks. In addition, BV-60-13, which provides a cross tie between the EC Make up Tanks, fails closed on loss of air. Based on NMP1 design bases document, SDBD-204 and NMP1 UFSAR Chapter V, Section E, Dated October 2001, the level control valves and the cross tie valve are required to be functional to meet the required eight hours.

Ref: 51-9133191 Appendix A.12, Section A.12.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-018	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open Manual Bypass Valve VLV-28-18 to provide reactor makeup from CRD system.

Ref: 51-9133191 Appendix A.12, Section A.12.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-019	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53 locally to supply RBCLC water to SDC heat exchangers.

Ref: 51-9133191 Appendix A.12, Section A.12.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-020	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09 locally to control SDC cooldown.

Ref: 51-9133191 Appendix A.12, Section A.12.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-021	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for the AC Power System. The AC Power system is required to ensure success of various performance goals. The power cable from PB 101 to PB 102 (Cable 101-5) can be damaged by fire in this area preventing EG-EDG102 from supplying PB 102. A repair is required to ensure Train 11 power for CSD.

Ref: 51-9133191 Appendix A.12, Section A.12.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-022	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-023	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may damage cable 16-49 adversely impacting credited CRD pump PMP-28-15. PMP-28-15 is required to provide inventory to the reactor vessel. CRD PMP-28-15 is repaired and operated locally at PB 16 per N1-DRP-GEN-003, Attachment 3.

Ref: 51-9133191 Table A.12.2 Fire Area Cable Assessment Report and Table A.12.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-024	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may damage cable 16-55 adversely impacting credited ESW pump PMP-72-04. PMP-72-04 is required to support decay heat removal from SDC. ESW PMP-72-04 is repaired and operated locally at PB 16 per N1-DRP-GEN-003, Attachment 4.

Ref: 51-9133191 Table A.12.2 Fire Area Cable Assessment Report and Table A.12.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-025	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may damage cable 16-43 or 16-76 adversely impacting credited RBCLC pump PMP-70-01. PMP-70-01 is required to support decay heat removal from SDC. RBCLC PMP-70-01 is repaired and operated locally at PB 16 per N1-DRP-GEN-003, Attachment 5.

Ref: 51-9133191 Table A.12.2 Fire Area Cable Assessment Report and Table A.12.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-026	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may damage cable 16-34 adversely impacting credited SDC pump PMP-38-149. PMP-38-149 is required to support decay heat removal. SDC PMP-38-149 is repaired and operated locally at PB 16 per N1-DRP-GEN-003, Attachment 6.

Ref: 51-9133191 Table A.12.2 Fire Area Cable Assessment Report and Table A.12.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-027	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may damage cable 12DV-11 or 12DV-29 adversely affecting credited SDC valve IV-38-02. SDC is required to support the decay heat removal function. Local-Manual operation of IV-38-02 per N1-SOP-21.1 may be required.

Ref: 51-9133191, Table A.12.2 Fire Area Cable Assessment Report and Table A.12.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
10	CABLE SPREADING ROOM EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-10-028	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in this area that affects the reactivity control performance goal for reactor scram. Cable-to-cable shorts on cables 1S-1041 and 1S-1891 impact Channel 11 scram solenoids SOV-113-272, SOV-113-273 and SOV-113-275 associated with IV-44.2-15, IV-44.2-16, IV-44.2-17, and IV-44.2-18. The SOV's remaining energized prevents venting the scram air header which, in turn, prevents reactor scram from the control room.

Ref: 51-9133191 Appendix A.12, Section A.12.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within EIR 51-9156521-000.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	**CONTROL COMPLEX EL 261-0 AND EL 277-0**

Fire Zone	Fire Zone Description
C2	AUXILIARY CONTROL ROOM, COMPUTER ROOM 261-0
C3	CONTROL ROOM EL 277-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(b) Inventory Control Function	The Train 12 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(d) Decay Heat Removal Function	Path A HSD is credited via use of EC's 111 & 112 for Decay Heat Removal and Train 11 support systems. Alternatively, Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 Support Systems. Path B is credited to achieve CSD. Path B CSD is accomplished via the use of the Train 12 SDC System supported by the Train 12 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure o Emergency Condenser Level	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Train 12 emergency diesel generator is credited for supplying power for the shutdown process when AC power is required to operate shutdown equipment. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	**CONTROL COMPLEX EL 261-0 AND EL 277-0**

Path C - Train 11 Positive Pressure Flow Path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-00-004 Rev. 0

Eng Eval Summary: FPEE 0-00-004 Rev. 0 - Engineering Evaluation of Alternate Fire Protection Systems for the Nine Mile Point Halon Protected Areas
Evaluation conducted for the existing Halon protected areas to determine if alternate types of fixed fire suppression systems can be utilized to replace Halon 1301 extinguishing systems. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-03-002 Rev. 0

Eng Eval Summary: FPEE 0-03-002 Rev. 0 - Evaluation of Interim Action To Prevent Personnel Injury From CO2
Evaluation justifies use of temporarily placing automatic CO2 systems in alarm-only mode due to adverse trend of unplanned system actuations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-04-002 Rev. 0

Eng Eval Summary: FPEE 1-04-002 Rev. 0 - Evaluation of Fire Effects on Current Transformers and Instrument Sensing Lines and The Plant Safe Shutdown Capability
The analysis provided in this FPEE demonstrates that a postulated fire at NMP1 and the resultant fire effects on Current Transformers and instrument tubing lines will not affect the existing assumptions in the NMP1's present Appendix R Safe Shutdown Analysis. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

Engineering Eval Id: FPEE 1-12-001 Rev. 0

Eng Eval Summary: This Engineering Evaluation provides analysis of the existing NMP1 approved exemptions from 10CFR50, Appendix R and input to NFPA-805 Licensing Amendment Request (LAR) submittal with respect to previously-approved exemptions to Appendix R to 10 CFR Part 50. This document provides the evaluation, analysis and basis to demonstrate that current plant configuration and NMP1 Fire Protection Program complies with the requirements of 10CFR50.48(c), NFPA 805. The evaluation documents the similar requirements of 10CFR50, Appendix R and NFPA 805. This evaluation demonstrates that the current plant configuration and fire protection features comply with the 10CFR50.48(c) licensing basis.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-95-001 Rev. 0

Eng Eval Summary: FPEE 1-95-001 Rev. 0 - Fire Damper 211-54 Installation Deviation
Evaluation justifies lack of retaining angles on all four sides of the sleeve of Damper 211-54. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
C2	Detection	D-3031PL	N	N	Y	Y	N	
	Detection	DX-3031A/B	N	N	Y	Y	N	
	Suppression	C-3031	N	N	N	N	Y	
	Suppression	H-3031	N	N	N	Y	N	
C3	Detection	D-3054	N	N	Y	Y	N	
	Suppression	None	N	N	N	N	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA11 of a Fire Originating In FA11
Activation of a Halon 1301 suppression system would not adversely impact equipment/components credited by the NSCA. The CO2 system installed in this area is manually actuated and is a back-up to the total flooding Halon 1301 system. It is not likely that activation as a result of a fire in the area could have an impact on the nuclear safety performance criteria.

Scenario 2: Suppression Affects in FA11 of a Fire Originating Outside of FA11
Based on the limited impact a fire can have on plant suppression systems, operation of fire suppression systems outside of FA11 would not be expected to cause activation of a system within FA11 that could have an impact on the nuclear safety performance criteria. Additionally, based on the fact that the CO2 system in FA11 is mechanically isolated, operation of fire suppression systems outside of FA11 would not be expected to cause activation of the CO2 system within FA11 that could have an impact on the nuclear safety performance criteria.

Since there are no installed water-based fire suppression systems in FA11, there is no flooding potential resulting from the fire suppression systems.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied. Note that VFDR-11-011 was completely resolved by crediting a modification to the plant.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks. See additional discussion related to this fire area in N1-FRE-F001.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems for the aux control room, C2, (Automatic Halon system) and manual suppression from the fire brigade. The control room, C3 has no automatic suppression. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-001	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for Main Feedwater. Main Feedwater is required to be isolated for inventory control. If offsite power is available, Main Feedwater may continue to supply the reactor vessel. Valve IV-31-07 may remain open due to fire damage or an internal wire-to-wire short on either cable 161-57 or 161-58. IV-31-08 may remain open due to fire damage or an internal wire-to-wire short on either cable 171-57 or 171-58. Main Feedwater pumps may spuriously start. Pump 29-02 may start due to an internal wire-to-wire short on cable 11-78. Pump 29-03 may spuriously start due to an internal wire-to-wire short on either cable 12-78 or 12-79. The result is reactor vessel overfill that has the potential to compromise the ability of the credited EC to achieve and maintain HSD. The clutch of turbine shaft driven pump 29-01 may remain engaged due to cable shorts or fire damage to cables 1F-110, 1F-111, 1F-4, 1F-5, 1S-1610, 1S-1730, 1S-1735, 1S-1736, 1S-1766. The clutch is disengaged by a procedure driven operator action.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the Inboard MSIV IV-01-01 and Outboard MSIV IV-01-03. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the power supply or control circuits of Inboard MSIV IV-01-01 and Outboard MSIV IV-01-03 leaves the MS line open. Inboard MSIV IV-01-01 may remain open due to a loss of the Train 11 AC electrical system below 4 kV or a cable-to-cable short on either cable 161-50 or 161-61. An internal wire-to-wire short to cable 1F-37, 1F-38, or 1F-50 (125 VDC) or a cable-to-cable short on cable 1F-51 (125 Vdc) prevents closure of series Outboard MSIV IV-01-03 and results in an inventory loss until manual action is taken to close the outboard MSIV IV-01-03.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the Inboard MSIV IV-01-02 and Outboard MSIV IV-01-04. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the power supply or control circuits of Inboard MSIV IV-01-02 and Outboard MSIV IV-01-04 leaves the MS line open. Inboard MSIV IV-01-02 may remain open due to a loss of the Train 12 AC electrical system below 4 kV or a cable-to-cable short on either cable 171-50 or 171-61. An internal wire-to-wire short to cable 1F-43, 1F-44, or 1F-55, or a cable-to-cable short on cable 1F-56 (125 Vdc) prevents closure of series Outboard MSIV IV-01-04 and results in an inventory loss until a manual action is taken to close the outboard MSIV IV-01-04.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system isolation valves IV-05-01R, IV-05-11, BV-05-05, and BV-05-07. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-158 or 1S-2017 or an internal wire-to-wire hot short on cable 1K-18 can maintain EC system isolation valve IV-05-01R open. Either an internal wire-to-wire short on cable 1K-18, 1K-156, or 1K-154 or a cable-to-cable short on cable 1K-158 can maintain EC system isolation valve IV-05-11 open. An internal wire-to-wire hot short on cable 167-187 can maintain EC system isolation valve BV-05-05 open. An internal wire-to-wire hot short on cable 167-190 can maintain EC system isolation valve BV-05-07 open. The multiple spurious combination results in an inventory loss to the torus.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the EC system isolation valves IV-05-04R, IV-05-12, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-159 or an internal wire-to-wire short on cable 1K-155, 1K-157, or 1K-25 can maintain the EC system isolation valves IV-05-04R and IV-05-12 open. A cable-to-cable hot short on cable 1K-22 or 1K-31; or an internal wire-to-wire short on either cable 1K-21 or 1K-30 can maintain the EC system isolation valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the EC system isolation valves IV-05-01R, IV-05-11, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-158 or 1S-2017 or an internal wire-to-wire hot short on cable 1K-18 can maintain EC system isolation valve IV-05-01R open. Either an internal wire-to-wire short on cable 1K-18, 1K-156, or 1K-154 or a cable-to-cable short on cable 1K-158 can maintain EC system isolation valve IV-05-11 open. A cable-to-cable hot short on cable 1K-22 or 1K-31; or an internal cable wire-to-wire short on either cable 1K-21 or 1K-30 can maintain the EC system isolation valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the EC system isolation valves IV-05-04R, IV-05-12, BV-05-05, and BV-05-07. These valves are required closed for HSD to support inventory control. An internal wire-to-wire short on cable 1K-155 or 1K-157 or 1K-25; or a cable-to-cable short on cable 1K-159 can maintain the EC system isolation valves IV-05-04R and IV-05-12 open. An internal wire-to-wire hot short on cable 167-187 can maintain EC system isolation valve BV-05-05 open. An internal wire-to-wire hot short on cable 167-190 can maintain EC system isolation valve BV-05-07 open. The multiple spurious combination results in an inventory loss to the Torus.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-009	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the EC system isolation valves IV-39-11R and IV-39-12R. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-111, 1K-114, 1K-139, or 1K-145 can maintain the EC system isolation valves IV-39-11R and IV-39-12R open resulting in an inventory loss to the Main Steam drain line.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-010	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the EC system isolation valves IV-39-13R and IV-39-14R. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-117, 1K-120, 1K-139, or 1K-145 can maintain the EC system isolation valves IV-39-13R and IV-39-14R open resulting in an inventory loss to the Main Steam drain line.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-011	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the RWCU isolation valves IV-33-02R, IV-33-04, and BV-33-41. One of these valves is required closed to support inventory control. IV-33-02R may remain open due to a cable-to-cable short on cable 167-7, or fire damage to cable 167-7 (ground) or 167-8 (ground or open). IV-33-04 may remain open due to an internal wire-to-wire short on cable 12DV-27 or 12DV-28. BV-33-41 may remain open due to a cable-to-cable short on cable 11-99 or 167-8. The result is that the RCS remains connected to the Clean-Up System.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been brought into compliance with NFPA 805 section 4.2.3, deterministic approach, via a plant modification as described in Attachment S.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-012	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area for shutdown cooling. Credited SDC pump, PMP-38-152, is required to support decay heat removal for CSD. The credited SDC pump, PMP-38-152, may spuriously start and run with no suction source. SDC pump PMP-38-152 may spuriously start due to an internal wire-to-wire short on cable 17-23. SDC isolation valve IV-38-01 has power removed to prevent spurious opening for a fire in this fire area. SDC valves IV-38-02 and BV-38-04 are normally closed. Min-flow recirc valve FCV-38-131 may remain closed due to an internal wire-to-wire short on cable 1K-4. In the event the credited SDC pump is not available or other equipment operation causes a vessel overfill rendering EC's unavailable, various circuit failures in Train 11 and 12 CS valves could render the CS system unavailable for vessel injection and heat removal.

Ref: 51-9133191, Appendix A.13, Section A.13.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-013	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in this area for the ERV's with regards to pressure control. ERV's PSV-01-102A, PSV-01-102B, PSV-01-102C, PSV-01-102D, PSV-01-102E, and PSV-01-102F may spuriously open, resulting in blowdown to Torus.

- The master ADS logic may actuate, or:
- PSV-01-102A may spuriously open due to a cable-to-cable short on cable 1F-151 in conjunction with an open in cable 1F-178.
- PSV-01-102B may spuriously open due to a cable-to-cable short on cable 1F-151 in conjunction with an open in cable 1F-178.
- PSV-01-102C may spuriously open due to a cable-to-cable short on cable 1F-156 in conjunction with an open in cable 1F-179.
- PSV-01-102D may spuriously open due to a cable-to-cable short on cable 1F-156 in conjunction with an open in cable 1F-179.
- PSV-01-102E may spuriously open due to a cable-to-cable short on cable 1F-151in conjunction with an open in cable 1F-178.
- PSV-01-102F may spuriously open due to a cable-to-cable short on cable 1F-156 in conjunction with an open in cable 1F-179

The alternate cooling path using CS may not be available due to the loss of both electrical trains.

Ref: 51-9133191, Appendix A.13, Section A.13.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-014	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in this area that affects the inventory control function. Shorts on various combinations of the following cables can cause IV-44.2-15, IV-44.2-16, IV-44.2-17 and IV-44.2-18 to spuriously open (post scram) resulting in an inventory loss.

•Cable-to-cable short on cable 1F-163
•Cable-to-cable short on cable 1F-164
•Cable-to-cable short on cable 1S-1041
•Cable-to-cable short on cable 1S-1042
•Cable-to-cable short on cable 1S-1891
•Cable-to-cable short on cable 1S-1892
•An internal wire-to-wire short on cable 1S-1020 in conjunction with an external cable-to-cable short on cable 1S-1078
•An internal wire-to-wire short on cable 1S-1020 in conjunction with external cable-to-cable shorts (2) on cable 1S-1083
•An internal wire-to-wire short on cable 1S-1025 in conjunction with an external cable-to-cable short on cable 1S-1093
•An internal wire-to-wire short on cable 1S-1025 in conjunction with external cable-to-cable shorts (2) on cable 1S-1094.

Ref: 51-9133191, Appendix A.13, Section A.13.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-015	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. A loss of air fails EC Level Control Valves, LCV-60-17 & LCV-60-18, open. This could deplete the inventory in the Emergency Condenser Make-Up Water Tanks prior to the eight hours required for the EC's to remove decay heat. Local operator action to manually close or throttle VLV-60-11 and VLV-60-12, and open/close BV-60-13 as required ensures availability of make-up inventory for the EC's chosen by the operator for HSD.

Ref: 51-9133191, Appendix A.13, Section A.13.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-016	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. Block Valve, BV-70-53 may fail closed on loss of instrument air. This Valve is required to be open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70- 53.

Ref: 51-9133191, Appendix A.13, Section A.13.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-017	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for CTSRW. If Train 11 power is available, an internal wire-to-wire short on cable 102-14 or 102-15 can spuriously start CTSRW pump PMP-93-01. An internal wire-to-wire short on cable 161-160 can spuriously open CTSRW to CTS cross connect valve FCV-93-72. The spurious start of the Path C (Train 11) CTSRW pump, PMP-93-01, in conjunction with the spurious opening of Path C (Train 11) valve FCV-93-72 causes an inadvertent CTS actuation which results in an increasing Torus water level. The increase in Torus water level will lead the Plant operators to enter the EOPs and will cause a deviation from the credited safe shutdown strategy.

Ref: 51-9133191, Appendix A.13, Section A.13.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within EIR 51-9156521-000.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-018	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for CTSRW. If Train 12 power is available, an internal wire-to-wire short on cable 103-22 or 103-23 can spuriously start CTSRW pump PMP-93-04. An internal wire-to wire short on cable 171-160 spuriously opens CTSRW to CTS cross connect valve FCV-93-73. The spurious start of Path D (Train 12) CTSRW pump, PMP-93-04, in conjunction with the spurious opening of Path D (Train 12) valve FCV-93-73 causes an inadvertent CTS actuation which results in an increasing Torus water level. The increase in Torus water level will lead the Plant operators to enter the EOPs and will cause a deviation from the credited safe shutdown strategy.

Ref: 51-9133191, Appendix A.13, Section A.13.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within EIR 51-9156521-000.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-019	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for the AC Power System. The AC Power system is required to ensure success of various performance goals. If offsite power is available, combinations of internal wire-to-wire shorts on cables 17-1, 17-2 and 17-3 can result in out of phase closure of Train 12 600 VAC breakers 1051, 1052 and 1053. The result is a loss of all credited Train 12 power below 4 kV.

Ref: 51-9133191, Appendix A.13, Section A.13.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-020	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of AC power and battery charging capability due to damage to the following cables:

Train 11: 101-6, 1A-147, 1A-60, 102-22, 102-28, 102-31, 102-43, 102-44, 102-49, 11B-28, 1A-124, 1B-126, 1S-2386, 102-2, 102-29, 102-42, 102-47, 102-51, 102-52, 102-55, 102-56, 1A-116, 16-124, 16-2, 16-3, 16-82

Train 12: 101-64, 1A-161, 103-18, 103-22, 103-28, 103-29, 103-30, 103-31, 103-43, 103-44, 103-49, 12B-28, 1A-134, 1B-121, 1S-2388, 103-2, 103-42, 103-47, 103-51, 103-52, 103-56, 1A-130, 1A-61, 17-12, 103-2, 17-2, 17-3, 17-67

As a result, battery loads are shed per procedure N1-SOP-21.1 to extend battery capability until the power system restored via N1-DRP-GEN-001.

Ref: 51-9133191, Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-021	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of credited Train 12 power board PB 17B due to fire damage to cable 103-2, 17-2, 17-3 or 17-67. Power restored to PB 17B per N1-DRP-GEN-004, Attachment 2.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-022	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of credited RBCLC pump PMP-70-02 due to cable damage to either cable 17-29, 17-30, 17-60 or 17-65. To support the decay heat removal function, PMP-70-02 is operated locally per N1-DRP-GEN-004, Attachment 9.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-023	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of PB 171B due to damage to the following power supply component cables:

BKR-(17/007B)R1052/612: Cables 17-2, 17-67
BKR-(17B/002B)R1053/613: Cable 17-3
BKR-(103/1-9)R1031/181: Cables 101-64, 103-2
BKR-(103/1-1)R1032/581: Cables 101-64, 103-18, 103-22, 103-28, 103-29, 103-30, 103-31, 103-43, 103-44, 103-45, 103-46, 103-49, 103-67, 12B-28, 1A-134, 1A-161, 1B-121, 1S-2388
BKR-(101/2A-1)R1013/161: Cables 101-64, 12B-28, 1A-161

This loss of power to PB-171B affects the following credited shutdown loads:

PMP-79-54: EG-EDG103 Cooling Water Pump
BLWROT-209-05: EG-EDG103 Roof Exhaust Fan
BLWROT-209-06: EG-EDG103 Roof Exhaust Fan
DOOR-D-034: EG-EDG103 Roll-Up Door
BV-38-04: SDC Pump PMP-38-152 Suction Blocking Valve

In support of shutdown from outside the control room, power is restored to PB 171B via local breaker operations per N1-DRP-GEN-004, Attachment 3.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-024	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of PB 167 due to damage to the following Train 11 component cables:

BKR-(16B/013B)R1043/603: Cable 16-3
BKR-(102/2-9)R1021/171: Cables 101-6, 102-2, 11B-10
BKR-(102/2-1)R1022/571: Cables 101-6, 102-22, 102-28, 102-29, 102-30, 102-31, 102-43, 102-44, 102-45, 102-46, 102-49, 102-67, 11B-28, 1A-124, 1A-147, 1B-126, 1S-2386
BKR-(101/2B-1)R1012/151: Cables 101-6, 11B-28, 1A-147, 1A-60

In support of shutdown from outside the control room, power is realigned from Train 12 to PB 167 per N1-DRP-GEN-004, Attachment 4.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-025	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of CRD pump PMP-28-17 due to fire damage to cable 17-32, 17-33, 17-66 or 1S-1057. In support of shutdown from outside the control room and to provide reactor vessel inventory makeup, CRD pump PMP-28-17 operated locally per N1-DRP-GEN-004, Attachment 5.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-026	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause isolation of the normal reactor inventory makeup line from CRD due to fire damage to cables for the following valves:

FCV-44-149: 1F-123, 1F-90, 1S-1765, 1S-2586, 1S-325, 1S-405, 1S-407, 1S-560, 1S-645, 1S-646
PCV-44-04: 155-71
PCV-44-05: 155-74

To support the reactor vessel inventory makeup function, CRD flow is controlled manually by operating VLV-28-18 locally per N1-DRP-GEN-004, Attachment 6.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-027	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause a malfunction of credited ESW pump PMP-72-03 due to fire damage to cable 17-38, 17-39 or 17-65 resulting in the need for the DFP to supply ESW per N1-DRP-GEN-004, Attachment 8 to accomplish the decay heat removal function.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-028	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may result in misoperation of credited SDC isolation valve IV-38-01 due to fire damage to cable 167-11 or 167-12. To support meeting the decay heat removal function, SDC valve IV-38-01 is repaired and operated from PB 167 per N1-DRP-GEN-004, Attachment 15.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-029	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may result in misoperation of credited SDC isolation valve IV-38-13 due to fire damage to cable 167-15 or 167-16. To support meeting the decay heat removal function, SDC valve IV-38-13 is repaired and operated from PB 167 per N1-DRP-GEN-004, Attachment 16.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-030	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may cause the loss of credited Train 12 battery chargers BC-B12-1 and BC-B12-2 due to fire damage to cable 17-12. The train 12 battery is required to support various shutdown performance goals. Computer MG Set 167 is used to charge BAT12 per N1-DRP-GEN-004, Attachment 17.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area — **Fire Area Description**
11 — **CONTROL COMPLEX EL 261-0 AND EL 277-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-031	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A fire in this area may cause a loss of normal AC power and a loss of starting air. To support various shutdown performance requirements, AC power is required. In order to facilitate starting EG-EDG103, EG-EDG103 starting air is restored per N1-DRP-GEN-004, Attachment 18.

Ref: 51-9133191, Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-032	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may adversely affect credited EG-EDG103 due to fire damage to numerous cables. EG-EDG103 is required to support various performance goals. EG-EDG103 local control restored via repair action associated with cables 103-28, 103-29, 103-49, 103-51 and 103-52 and output breaker R1032 is operated locally per N1-DRP-GEN-004, Attachment 1.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-033	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A postulated fire in this area may adversely affect the credited Train 12 electrical system due to damage to the following power supply and component cables:

BKR-(101/2A-1)R1013/161: Cables 101-64, 12B-28, 1A-161
BKR-(103/1-1)R1032/581: Cables 101-64, 103-18, 103-22, 103-28, 103-29, 103-30, 103-31, 103-43, 103-44, 103-45, 103-46, 103-49, 103-67, 12B-28, 1A-134, 1A-161, 1B-121, 1S-2388
BKR-(103/1-9)R1031/181: Cables 101-64, 103-2
PB-103: Cables 101-64, 103-22, 103-28, 103-31, 103-43, 103-44, 103-49, 12B-28, 1A-134, 1A-161, 1A-61, 1B-121, 1S-2388, 103-47 (UV Protection), 103-66 (UV Protection)
PMP-80-03: Cables 103-18, 103-19, 103-59
PMP-80-23: Cables 103-10, 103-11
PMP-81-04: Cables 103-6, 103-7
PMP-81-24: Cables 103-26, 103-27
PMP-81-49: Cables 103-61, 103-62
PMP-81-52: Cables 103-58, 103-64, 103-65
PMP-93-03: Cables 103-14, 103-15
PMP-93-04: Cables 103-22, 103-23

To facilitate shutdown from outside the control room, the identified power supply and component breakers are opened locally at PB101 and PB 103 per N1-DRP-GEN-004, Attachment 1.

Ref: 51-9133191, Table A.13.2 Fire Area Cable Assessment Report and Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	**CONTROL COMPLEX EL 261-0 AND EL 277-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-034	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

Ref: 51-9133191 Appendix A.13, Section A.13.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-035	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: In support of shutdown from outside the control room and the decay heat removal performance function, SDC valves BV-38-04, FCV-38-10 and IV-38-02 are operated locally per N1-DRP-GEN-004, Attachment 12, Attachment 13 or Attachment 14.

Ref: 51-9133191, Table A.13.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
11	CONTROL COMPLEX EL 261-0 AND EL 277-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-11-036	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for a postulated fire in this area that affects the reactivity control performance goal for reactor scram. Shorts on various combinations of the following cables impact Channel 11 scram solenoids SOV-113-272, SOV-113-273 and SOV-113-275; and Channel 12 scram solenoids SOV-113-271, SOV-113-274 and SOV-113-276 associated with IV-44.2-15, IV-44.2-16, IV-44.2-17 and IV-44.2-18. The SOV's remaining energized prevents venting the scram air header which, in turn, prevents reactor scram from the control room.

•Cable-to-cable short on cable 1F-163
•Cable-to-cable short on cable 1F-164
•Cable-to-cable short on cable 1S-1041
•Cable-to-cable short on cable 1S-1042
•Cable-to-cable short on cable 1S-1891
•Cable-to-cable short on cable 1S-1892
•An internal wire-to-wire short on cable 1S-1020 in conjunction with an external cable-to-cable short on cable 1S-1078
•An internal wire-to-wire short on cable 1S-1020 in conjunction with external cable-to-cable shorts (2) on cable 1S-1083
•An internal wire-to-wire short on cable 1S-1025 in conjunction with an external cable-to-cable short on cable 1S-1093
•An internal wire-to-wire short on cable 1S-1025 in conjunction with external cable-to-cable shorts (2) on cable 1S-1094.

Ref: 51-9133191, Appendix A.13, Section A.13.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within EIR 51-9156521-000.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
12	ADMINISTRATION BUILDING EL 250-0

Fire Zone	Fire Zone Description
AB1A	RECORDS STORAGE AREA EL 250--0
AB1B	SAS EQUIPMENT AREA EL 252-0
AB1C	CPU EQUIPMENT AREA EL 252-0
AB1D	GENERAL AREA EL 250-0
AB1E	LOCKER AREA, LUNCH ROOM, OFFICES EL 261-0
AB2A	ACCESS PASSAGEWAY EL 248-0
AB2B	TECHNICAL SUPPORT AREA EL 248-0
AB2C	RADIATION RECORDS AREA EL 248-0
AB2D	WAREHOUSE AREA EL 248-0
AB3A	WAREHOUSE AREA EL 261-0
AB3B	OIL STORAGE ROOM EL 261-0
AB3C	STOREROOM TRUCK DOCK EL 261-0
AB3D	ELECTRICAL/MECHANICAL SHOP AREA, OFFICE AREAS, LOCKER ROOMS EL 261-0
AB3E	TELEPHONE ROOM 1 EL 261-0
AB3F	TELEPHONE ROOM 2 EL 261-0
AB4A	GENERAL OFFICE AREA EL 277-0
AB4B	FILE ROOM EL 277-0
AB4C	RECORDS PROCESSING AREA EL 277-0
AB4D	GENERAL OFFICE AREA EL 277-0
AB5	PENTHOUSE VENTILATION ROOM EL 290-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	Both 11 and 12 CRD pumps are credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 12

Fire Area Description ADMINISTRATION BUILDING EL 250-0

(d) Decay Heat Removal Function	Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 Support Systems. Path A and Path B are credited to achieve CSD. Path A CSD is accomplished via use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems. Path B CSD is accomplished via use of the Train 12 SDC System supported by the Train 12 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
12	ADMINISTRATION BUILDING EL 250-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system. Path B is generally supported by the Train 12 AC and DC power systems. Some components in Path B systems are powered by the Train 11 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open)	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
12	ADMINISTRATION BUILDING EL 250-0

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:

Path C – Train 11 positive pressure flow path.
Path D – Train 12 positive pressure flow path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev.1

Eng Eval Summary: FPEE 1-90-020 Rev.1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
12	ADMINISTRATION BUILDING EL 250-0

Engineering Eval Id: FPEE 1-95-001 Rev. 0

Eng Eval Summary: FPEE 1-95-001 Rev. 0 - Fire Damper 211-54 Installation Deviation
Evaluation justifies lack of retaining angles on all four sides of the sleeve of Damper 211-54. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES								
Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
AB1A	Detection	DX-9209A/B	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
AB1B	Detection	DX-9289A/B	N	N	N	N	Y	
	Suppression	H-9289	N	N	N	N	Y	
AB1C	Detection	DX-9279A/B	N	N	N	N	Y	
	Suppression	H-9279	N	N	N	N	Y	
AB1D	Detection	DA-9219	N	N	N	N	Y	
	Suppression	SP-9219	N	N	N	N	Y	
AB1E	Detection	D-9169	N	N	N	N	Y	
	Suppression	SP-9069	N	N	N	N	Y	
AB2A	Detection	None	N	N	N	N	N	
	Suppression	SP-9079	N	N	N	N	Y	
AB2B	Detection	D-9299	N	N	N	N	Y	
	Detection	D-9309	N	N	N	N	Y	
	Detection	D-9499	N	N	N	N	Y	
	Suppression	SP-9079	N	N	N	N	Y	
AB2C	Detection	D-9199	N	N	N	N	Y	
	Detection	D-9409	N	N	N	N	Y	
	Suppression	SP-9079	N	N	N	N	Y	
AB2D	Detection	None	N	N	N	N	N	
	Suppression	SP-9079	N	N	N	N	Y	
AB3A	Detection	D-9159	N	N	N	N	N	
	Detection	D-9499	N	N	N	N	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
12	ADMINISTRATION BUILDING EL 250-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
	Suppression	SP-9089	N	N	N	N	Y	
AB3B	Detection	None	N	N	N	N	N	
	Suppression	SP-9089	N	N	N	N	Y	
AB3C	Detection	None	N	N	N	N	N	
	Suppression	SP-9119	N	N	N	N	Y	
AB3D	Detection	D-9179	N	N	N	N	Y	
	Detection	D-9189	N	N	N	N	Y	
	Suppression	SP-9069	N	N	N	N	Y	
	Suppression	SP-9089	N	N	N	N	Y	
AB3E	Detection	DX-9229A/B	N	N	N	N	Y	
	Suppression	H-9229	N	N	N	N	Y	
AB3F	Detection	DX-9259A/B	N	N	N	N	Y	
	Suppression	H-9259	N	N	N	N	Y	
AB4A	Detection	D-9049	N	N	N	N	Y	
	Detection	DA-9019	N	N	N	N	Y	
	Detection	DA-9039	N	N	N	N	Y	
	Detection	DA-9049	N	N	N	N	Y	
	Detection	DA-9149	N	N	N	N	Y	
	Suppression	SP-9019	N	N	N	N	Y	
	Suppression	SP-9039	N	N	N	N	Y	
AB4B	Detection	DA-9029	N	N	N	N	Y	
	Suppression	WP-9029	N	N	N	N	Y	
AB4C	Detection	D-9139	N	N	N	Y	N	
	Detection	DA-9029	N	N	N	Y	N	
	Suppression	WP-9029	N	N	N	Y	N	
AB4D	Detection	DA-9239	N	N	N	Y	N	
	Suppression	SP-9239	N	N	N	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
12	ADMINISTRATION BUILDING EL 250-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
AB5	Detection	D-9249	N	N	N	N	Y	
	Detection	D-9249FL	N	N	N	N	Y	
	Suppression	WD-9249FL	N	N	N	N	Y	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA12 of a Fire Originating In FA12
The activation of a water-based suppression system would not adversely impact equipment/components credited by the NSCA in the Administration Building.

Scenario 2: Suppression Affects in FA12 of a Fire Originating Outside of FA12
Based on the limited impact a fire can have on plant suppression systems, the operation of fire suppression systems outside of FA12 would not be expected to cause activation of a fire protection system within FA12 that could have an impact on the nuclear safety performance criteria.

Water-based fire suppression systems installed in FA12 are adequately separated from other plant areas, such that there is expected to be no flooding potential resulting from the fire suppression systems that could negatively impact the nuclear safety performance criteria.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
12	ADMINISTRATION BUILDING EL 250-0

ΔLERF: Redacted

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems for partial coverage and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-12-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
12	ADMINISTRATION BUILDING EL 250-0

51-9133191 Appendix A.14, Section A.14.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-12-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. BV-60-13 may fail closed on loss of instrument air. Valve is required open for hot shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-60-13.

51-9133191 Appendix A.14, Section A.14.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-12-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. LCV-60-17 may fail open on loss of instrument air. Valve is required to close for hot shutdown to support decay heat removal. A Recovery Action may be required to close valve VLV-60-12.

51-9133191 Appendix A.14, Section A.14.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
12	ADMINISTRATION BUILDING EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-12-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. LCV-60-18 may fail open on loss of instrument air. Valve is required to throttle for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle valve VLV-60-11.

51-9133191 Appendix A.14, Section A.14.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-12-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

51-9133191 Appendix A.14, Section A.14.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-12-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

51-9133191 Appendix A.14, Section A.14.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
12	ADMINISTRATION BUILDING EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-12-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-10 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-10.

51-9133191 Appendix A.14, Section A.14.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-12-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

51-9133191 Appendix A.14, Section A.14.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
13	SCREENHOUSE

Fire Zone	Fire Zone Description
S1	SCREENHOUSE EL 225-0 - 256-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Performance Goals section	No comments.
(b) Inventory Control Function	Both 11 and 12 CRD pumps are credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path A HSD is achieved via use of EC's 111 & 112 for Decay Heat Removal and Train 11 support systems. Also credited, Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 Support Systems. Path A and B are credited to achieve CSD. Path A CSD is accomplished via use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems. Path B CSD is accomplished via use of the Train 12 SDC System supported by the Train 12 RBCLC and ESW Systems. The DFP can be used to provide cooling to the RBCLC System.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
13	SCREENHOUSE

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: . o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure o Emergency Condenser Level	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
13	SCREENHOUSE

(f) Vital Auxiliaries	1) AC and DC Distribution System Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system. The DFP can be used to provide cooling to the EDG's. Path B is generally supported by the Train 12 AC and DC power systems. Some components in Path B systems are powered by the Train 11 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and DFP provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the DFP o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open)	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
13	SCREENHOUSE

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:

Path C – Train 11 positive pressure flow path.
Path D – Train 12 positive pressure flow path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
S1	Detection	D-5013	N	N	N	Y	N	
	Detection	D-5023	N	N	N	Y	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
13	SCREENHOUSE

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
	Suppression	SP-5033	Y	N	N	N	Y	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA13 of a Fire Originating In FA13
The activation of a wet-pipe sprinkler system would not adversely impact equipment/components credited by the NSCA.

Scenario 2: Suppression Affects in FA13 of a Fire Originating Outside of FA13
Based on the limited impact a fire can have on plant suppression systems, the operation of a fire suppression system outside of FA13 is not expected to impact fire suppression systems within FA13 that could have an impact on the nuclear safety performance criteria.

Actuation of a wet-pipe sprinkler system would only impact a small number of sprinklers. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
13	SCREENHOUSE

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-13-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

51-9133191 Appendix A.15, Section A.15.6 Instrument Air

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
13	SCREENHOUSE

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-13-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-60-13 may fail closed on loss of instrument air. Valve is required open for hot shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-60-13.

51-9133191 Appendix A.15, Section A.15.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-13-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-17 may fail open on loss of instrument air. Valve is required to throttle or close for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle or close valve VLV-60-12 depending on the chosen HSD path.

51-9133191 Appendix A.15, Section A.15.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
13	SCREENHOUSE

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-13-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-18 may fail open on loss of instrument air. Valve is required to throttle or close for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle or close valve VLV-60-11 depending on the chosen HSD path.

51-9133191 Appendix A.15, Section A.15.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-13-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

51-9133191 Appendix A.15, Section A.15.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-13-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-10 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-10.

51-9133191 Appendix A.15, Section A.15.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
13	SCREENHOUSE

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-13-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

51-9133191 Appendix A.15, Section A.15.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-13-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

51-9133191 Appendix A.15, Section A.15.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
13	SCREENHOUSE

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-13-009	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists between the ESW pumps. Also, normal SW is assumed not to be available in the deterministic analysis. A repair action is required to install the required piping spool pieces for the diesel fire pump (DFP) to support the RBCLC system with cooling water and to manually start the DFP. This action is in support of the decay heat removal function for CSD.

51-9133191 Appendix A.15, Section A.15.2, Primary Shutdown Method

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-13-010	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists between the diesel generator cooling water pumps. Pump PMP-79-53 is adversely affected by a fire in this area due to damage to its motor power cable 161-186. Pump PMP-79-54 is adversely affected by a fire in this area due to damage to its motor power cable 171-63. Also, normal SW is assumed not to be available in the deterministic analysis. A repair action is required to install the required piping spool pieces to for the diesel fire pump (DFP) to support the EDG's with cooling water and to manually start the DFP. This action is in support of the vital auxiliaries, AC and DC distribution systems.

51-9133191 Appendix A.15, Section A.15.2, Primary Shutdown Method

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
14	**DIESEL FIRE PUMP ROOM EL 261-0**

Fire Zone	Fire Zone Description
S2	DIESEL FIRE PUMP ROOM EL 256-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	Both 11 and 12 CRD pumps are credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path A HSD is achieved via use of EC's 111 & 112 for Decay Heat Removal and Train 11 support systems. Also credited, Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 Support Systems. Path A and B are credited to achieve CSD. Path A CSD is accomplished via use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems. Path B CSD is accomplished via use of the Train 12 SDC System supported by the Train 12 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
14	DIESEL FIRE PUMP ROOM EL 261-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure o Emergency Condenser Level	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
14	DIESEL FIRE PUMP ROOM EL 261-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system. Path B is generally supported by the Train 12 AC and DC power systems. Some components in Path B systems are powered by the Train 11 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open)	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
14	DIESEL FIRE PUMP ROOM EL 261-0

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:

Path C – Train 11 positive pressure flow path.
Path D – Train 12 positive pressure flow path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: EIR 51-9156616 Rev. 1

Eng Eval Summary: EIR 51-9156616 Rev. 1 - Nine Mile Point Unit 1 Code Compliance Evaluation for NFPA 30, Flammable and Combustible Liquids Code, 2000 Edition
Evaluation provides comparison of the plant combustible liquid storage configurations and methods to applicable NFPA 30 requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
14	DIESEL FIRE PUMP ROOM EL 261-0

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
S2	Detection	D-5023	Y	N	N	N	Y	
	Suppression	SP-5033	Y	N	N	N	Y	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA14 of a Fire Originating In FA14
The activation of a wet-pipe sprinkler system would not adversely impact equipment/components credited by the NSCA.

Scenario 2: Suppression Affects in FA14 of a Fire Originating Outside of FA14
Based on the limited impact a fire can have on plant suppression systems, the operation of a fire suppression system outside of FA14 is not expected to impact fire suppression systems within FA14 that could have an impact on the nuclear safety performance criteria.

Actuation of a wet-pipe sprinkler system would only impact a small number of sprinklers. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were

within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems (automatic sprinklers) and manual suppression from the fire brigade. Pre-fire plans are documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
14	DIESEL FIRE PUMP ROOM EL 261-0

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-14-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

51-9133191 Appendix A.16, Section A.16.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-14-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-60-13 may fail closed on loss of instrument air. Valve is required open for hot shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-60-13.

51-9133191 Appendix A.16, Section A.16.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
14	DIESEL FIRE PUMP ROOM EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-14-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-17 may fail open on loss of instrument air. Valve is required to throttle or close for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle or close valve VLV-60-12 depending on the chosen HSD path.

51-9133191 Appendix A.16, Section A.16.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-14-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-18 may fail open on loss of instrument air. Valve is required to throttle or close for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle or close valve VLV-60-11 depending on the chosen HSD path.

51-9133191 Appendix A.16, Section A.16.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-14-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

51-9133191 Appendix A.16, Section A.16.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
14	DIESEL FIRE PUMP ROOM EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-14-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

51-9133191 Appendix A.16, Section A.16.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-14-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-10 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-10.

51-9133191 Appendix A.16, Section A.16.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-14-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

51-9133191 Appendix A.16, Section A.16.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
15	**RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0**

Fire Zone	Fire Zone Description
RS1A	DRUM WASTE STORAGE VAULTS EL 252-0
RS1B	ELECTRICAL EQUIPMENT ROOM EL 252-0
RS1C	GENERAL FLOOR AREA SOUTH, DRUM STORAGE ROOM EL 252-0
RS2A	TRUCK LOADING AREA, NORTH EL 261-0
RS2B	TRUCK LOADING AREA, WEST EL 261-0
RS2C	GENERAL FLOOR AREA EL 261-0
RS2D	RADWASTE CONTROL ROOM, WEST EL 261-0
RS2E	GENERAL FLOOR AREA, SOUTH EL 261-0
RS3A	GENERAL FLOOR AREA, WEST EL 281-0
RS4A	GENERAL FLOOR AREA, NORTHWEST EL 292-0
RS5B	GENERAL FLOOR AREA, SOUTHWEST EL 292-0
WD1	GENERAL AREA El 225-0 & 229-0
WD2	GENERAL AREA EL 247-0
WD3A	GENERAL AREA EL 261-0
WD3B	RADWASTE CONTROL ROOM EL 261-0
WD3C	BALER ROOM EL 261-0
WD3D	DOW SOLIDIFICATION AREA EL 261-0
WD3E	TRUCK BAY EL 261-0
WD4	WASTE BUILDING VENTILATION AREA EL 277-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	Both 11 and 12 CRD pumps are credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
15	**RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0**

(d) Decay Heat Removal Function	Path A HSD is achieved via use of EC's 111 & 112 for Decay Heat Removal and Train 11 support systems. Also credited, Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 Support Systems. Path A and B are credited to achieve CSD. Path A CSD is accomplished via use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems. Path B CSD is accomplished via use of the Train 12 SDC System supported by the Train 12 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure o Emergency Condenser Level	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
15	**RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0**

(f) Vital Auxiliaries	1) AC and DC Distribution System Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system. Path B is generally supported by the Train 12 AC and DC power systems. Some components in Path B systems are powered by the Train 11 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open)	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
15	RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:

Path C – Train 11 positive pressure flow path.
Path D – Train 12 positive pressure flow path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-85-007 Rev. 0

Eng Eval Summary: FPEE 1-85-007 Rev. 0 - Waste Building Truck Bay West Wall
Evaluation justifies method to seal conduit penetrations through the west wall of the Waste Building Truck Bay and the north wall of the Turbine Building at the 261-ft. elevation. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
15 | **RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0**

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
RS1A	Detection	None	N	N	N	N	N	
	Suppression	None	N	N	N	N	N	
RS1B	Detection	DX-6188A/B	N	N	N	N	Y	
	Suppression	H-6188	N	N	N	N	Y	
RS1C	Detection	D-6198	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
RS2A	Detection	None	N	N	N	N	N	
	Suppression	SP-6168	N	N	N	N	Y	
RS2B	Detection	None	N	N	N	N	N	
	Suppression	SP-6168	N	N	N	N	Y	
RS2C	Detection	D-6188	N	N	N	N	Y	
	Suppression	SP-6168	N	N	N	N	Y	
RS2D	Detection	DX-6178A/B	N	N	N	N	Y	
	Suppression	H-6178	N	N	N	N	Y	
RS2E	Detection	None	N	N	N	N	N	
	Suppression	None	N	N	N	N	N	
RS3A	Detection	D-6148	N	N	N	N	Y	
	Detection	D-6158	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
RS4A	Detection	DA-6128	N	N	N	N	Y	
	Detection	DA-6128FL	N	N	N	N	Y	
	Suppression	WD-6128FL	N	N	N	N	Y	
RS5B	Detection	DA-6138	N	N	N	N	Y	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
15 | RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
	Suppression	None	N	N	N	N	N	
WD1	Detection	D-6013	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
WD2	Detection	D-6013	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
WD3A	Detection	D-6043	N	N	N	N	Y	
	Suppression	SP-6043	N	N	N	N	Y	
	Suppression	WP-6063	N	N	N	N	Y	
WD3B	Detection	D-6043	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	
WD3C	Detection	D-6043	N	N	N	N	Y	
	Suppression	SP-6043	N	N	N	N	Y	
WD3D	Detection	D-6063	N	N	N	N	Y	
	Detection	DA-6063	N	N	N	N	Y	
	Suppression	WP-6063	N	N	N	N	Y	
WD3E	Detection	D-6043	N	N	Y	N	Y	
	Suppression	SP-6073	N	N	Y	N	Y	
WD4	Detection	D-6103	N	N	N	N	Y	
	Detection	D-6104	N	N	N	N	Y	
	Suppression	None	N	N	N	N	N	

Legend: **Table Field: "Required ?"**

S - Required for Chapter 4 Separation Criteria
L - Required for NRC Approved Licensing Action
E - Required for Existing Engineering Equivalency Evaluation
R - Required for Risk Significance
D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
15	**RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0**

Scenario 1: Suppression Affects in FA15 of a Fire Originating In FA15
The activation of a water-based or Halon 1301 suppression system would not adversely impact equipment/components credited by the NSCA.

Scenario 2: Suppression Affects in FA15 of a Fire Originating Outside of FA15
Based on the limited impact a fire can have on plant suppression systems, the operation of a fire suppression system outside of FA15 is not expected to impact fire suppression systems within FA15 that could have an impact on the nuclear safety performance criteria.

Actuation of a dry-pipe or preaction sprinkler system would only impact a small number of sprinklers. The water spray system in FA15 is designed to spray on a specific charcoal filter hazard. The Radwaste Solidification Storage Building (RSSB) is equipped with floor drains throughout the building. Water runoff from fire suppression activities in the RSSB would drain through the floor drains or down stairwells to floor drain sumps. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. The floor drain sump pumps will automatically transfer the potentially contaminated water to the Waste Collector Tank where it is stored prior to processing by the liquid waste system.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
15	RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0

consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems for partial coverage in the area credited for DID only and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-15-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

Ref: 51-9133191 Appendix A.17, Section A.17.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
15	RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-15-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. BV-60-13 may fail closed on loss of instrument air. Valve is required open for hot shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-60-13.

Ref: 51-9133191 Appendix A.17, Section A.17.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-15-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. LCV-60-17 may fail open on loss of instrument air. Valve is required to throttle or close for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle or close valve VLV-60-12 depending on the chosen HSD path.

Ref: 51-9133191 Appendix A.17, Section A.17.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-15-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. LCV-60-18 may fail open on loss of instrument air. Valve is required to throttle or close for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle or close valve VLV-60-11 depending on the chosen HSD path.

Ref: 51-9133191 Appendix A.17, Section A.17.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
15	RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-15-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

Ref: 51-9133191 Appendix A.17, Section A.17.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-15-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

Ref: 51-9133191 Appendix A.17, Section A.17.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-15-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-10 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-10.

Ref: 51-9133191 Appendix A.17, Section A.17.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
15	RADWASTE AND WASTE DISPOSAL BUILDINGS EL 252-0 THRU 292-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-15-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

Ref: 51-9133191 Appendix A.17, Section A.17.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
16A | **BATTERY BOARD ROOM 12 EL 261-0**

Fire Zone | **Fire Zone Description**
B1A | BATTERY BOARD ROOM 12 EL 261-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	The Train 11 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path A HSD is achieved via use of EC's 111 & 112 for Decay Heat Removal and Train 11 support systems. Path A is credited to achieve CSD. Path A CSD is accomplished via the use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glas, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure o Emergency Condenser Level	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16A	BATTERY BOARD ROOM 12 EL 261-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	No comments

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16A	BATTERY BOARD ROOM 12 EL 261-0

Path C – Train 11 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-12-001 Rev. 0

Eng Eval Summary: This Engineering Evaluation provides analysis of the existing NMP1 approved exemptions from 10CFR50, Appendix R and input to NFPA-805 Licensing Amendment Request (LAR) submittal with respect to previously-approved exemptions to Appendix R to 10 CFR Part 50. This document provides the evaluation, analysis and basis to demonstrate that current plant configuration and NMP1 Fire Protection Program complies with the requirements of 10CFR50.48(c), NFPA 805. The evaluation documents the similar requirements of 10CFR50, Appendix R and NFPA 805. This evaluation demonstrates that the current plant configuration and fire protection features comply with the 10CFR50.48(c) licensing basis.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16A	BATTERY BOARD ROOM 12 EL 261-0

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
B1A	Detection	DA-2161E	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA16A of a Fire Originating In FA16A
There are no installed fire suppression systems in FA16A that could be actuated as a result of a fire originating within the fire area.

Scenario 2: Suppression Affects in FA16A of a Fire Originating Outside of FA16A
There are no installed fire suppression systems in FA16A that could be actuated as a result of a fire originating outside of the fire area.

Since there are no installed fire suppression systems in FA16A, there is no flooding potential resulting from fire suppression system actuation.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16A	**BATTERY BOARD ROOM 12 EL 261-0**

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16A-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

Ref: 51-9133191 Appendix A18, Section A18.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16A	BATTERY BOARD ROOM 12 EL 261-0

action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16A-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-60-13 may fail closed on loss of instrument air. Valve is required open for hot shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-60-13.

Ref: 51-9133191 Appendix A18, Section A18.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16A-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-17 may fail open on loss of instrument air. Valve is required to throttle for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle valve VLV-60-12.

Ref: 51-9133191 Appendix A18, Section A18.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16A	BATTERY BOARD ROOM 12 EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16A-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-18 may fail open on loss of instrument air. Valve is required to close for hot shutdown to support decay heat removal. A Recovery Action may be required to close valve VLV-60-11.

Ref: 51-9133191 Appendix A18, Section A18.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16A-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

Ref: 51-9133191 Appendix A18, Section A18.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16A-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

Ref: 51-9133191 Appendix A18, Section A18.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16A	BATTERY BOARD ROOM 12 EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16A-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

Ref: 51-9133191 Appendix A18, Section A18.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16A-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Due to the unavailability of EDG 103 (Train 12, Path B), power is not available to motor operated valve IV-38-02. IV-38-02 is a normally closed valve that fails as-is on loss of power. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve IV-38-02.

Ref: 51-9133191, Table A.18.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16B	**BATTERY BOARD ROOM 11 EL 261-0**

Fire Zone	Fire Zone Description
B1B	BATTERY BOARD ROOM 11 EL 261-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	The Train 12 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 support systems. Path B is credited to achieve CSD. Path B CSD is accomplished via the use of the Train 12 SDC System supported by the Train 12 RBCLC and ESW Systems. Train 11 SDC valves IV-38-01 and IV-38-13 are powered from PB-167 which will automatically transfer to Train 12.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16B	BATTERY BOARD ROOM 11 EL 261-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glas, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16B	BATTERY BOARD ROOM 11 EL 261-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Path B is generally supported by the Train 12 AC and DC power systems. Some components in Path B systems are powered by the Train 11 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16B	BATTERY BOARD ROOM 11 EL 261-0

Path D – Train 12 positive pressure flow path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-12-001 Rev. 0

Eng Eval Summary: This Engineering Evaluation provides analysis of the existing NMP1 approved exemptions from 10CFR50, Appendix R and input to NFPA-805 Licensing Amendment Request (LAR) submittal with respect to previously-approved exemptions to Appendix R to 10 CFR Part 50. This document provides the evaluation, analysis and basis to demonstrate that current plant configuration and NMP1 Fire Protection Program complies with the requirements of 10CFR50.48(c), NFPA 805. The evaluation documents the similar requirements of 10CFR50, Appendix R and NFPA 805. This evaluation demonstrates that the current plant configuration and fire protection features comply with the 10CFR50.48(c) licensing basis.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16B	BATTERY BOARD ROOM 11 EL 261-0

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
B1B	Detection	DA-2161E	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria

Scenario 1: Suppression Affects in FA16B of a Fire Originating In FA16B
There are no installed fire suppression systems in FA16B that could be actuated as a result of a fire originating within the fire area.

Scenario 2: Suppression Affects in FA16B of a Fire Originating Outside of FA16B
There are no installed fire suppression systems in FA16B that could be actuated as a result of a fire originating outside of the fire area.

Since there are no installed fire suppression systems in FA16B, there is no flooding potential resulting from fire suppression system actuation.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16B	**BATTERY BOARD ROOM 11 EL 261-0**

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16B-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

Ref: 51-9133191 Appendix A.19, Section A.19.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 16B

Fire Area Description BATTERY BOARD ROOM 11 EL 261-0

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16B-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-60-13 may fail closed on loss of instrument air. Valve is required open for hot shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-60-13.

Ref: 51-9133191 Appendix A.19, Section A.19.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16B-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-17 may fail open on loss of instrument air. Valve is required to close for hot shutdown to support decay heat removal. A Recovery Action may be required to close valve VLV-60-12.

Ref: 51-9133191 Appendix A.19, Section A.19.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
16B	BATTERY BOARD ROOM 11 EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16B-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-18 may fail open on loss of instrument air. Valve is required to throttle for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle valve VLV-60-11.

Ref: 51-9133191 Appendix A.19, Section A.19.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16B-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-10 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-10.

Ref: 51-9133191 Appendix A.19, Section A.19.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-16B-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

Ref: 51-9133191 Appendix A.19, Section A.19.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17A	BATTERY ROOM 12 EL 277-291

Fire Zone	Fire Zone Description
B2A	BATTERY ROOM 12 EL 277-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	The Train 11 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path A HSD is achieved via use of EC's 111 & 112 for Decay Heat Removal and Train 11 support systems. Path A is credited to achieve CSD. Path A CSD is accomplished via the use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17A	BATTERY ROOM 12 EL 277-291

(f) Vital Auxiliaries

1) AC and DC Distribution System

No Comments

Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system.
The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers.

2) Cooling Systems

Various cooling water systems to support safe shutdown system operation:

o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers
o The EDG's are cooled by the EDG Cooling Water pumps
o Chilled Water for control room ventilation

3) HVAC Systems

HVAC Systems required to support post-fire shutdown are:

o Main control room ventilation (except for control room evacuation)
o Diesel generator rooms (fans operating and roll-up doors required open)

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17A	BATTERY ROOM 12 EL 277-291

Path C – Train 11 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-12-001 Rev. 0

Eng Eval Summary: This Engineering Evaluation provides analysis of the existing NMP1 approved exemptions from 10CFR50, Appendix R and input to NFPA-805 Licensing Amendment Request (LAR) submittal with respect to previously-approved exemptions to Appendix R to 10 CFR Part 50. This document provides the evaluation, analysis and basis to demonstrate that current plant configuration and NMP1 Fire Protection Program complies with the requirements of 10CFR50.48(c), NFPA 805. The evaluation documents the similar requirements of 10CFR50, Appendix R and NFPA 805. This evaluation demonstrates that the current plant configuration and fire protection features comply with the 10CFR50.48(c) licensing basis.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17A	BATTERY ROOM 12 EL 277-291

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
B2A	Detection	D-2224	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA17A of a Fire Originating In FA17A
There are no installed fire suppression systems in FA17A that could be actuated as a result of a fire originating within the fire area.

Scenario 2: Suppression Affects in FA17A of a Fire Originating Outside of FA17A
There are no installed fire suppression systems in FA17A that could be actuated as a result of a fire originating outside of the fire area.

Since there are no installed fire suppression systems in FA17A, there is no flooding potential resulting from fire suppression system actuation.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17A	BATTERY ROOM 12 EL 277-291

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17A-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17A	BATTERY ROOM 12 EL 277-291

Ref: 51-9133191 Appendix A.20, Section A.20.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17A-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-60-13 may fail closed on loss of instrument air. Valve is required open for hot shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-60-13.

Ref: 51-9133191 Appendix A.20, Section A.20.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17A-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-17 may fail open on loss of instrument air. Valve is required to throttle for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle valve VLV-60-12.

Ref: 51-9133191 Appendix A.20, Section A.20.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17A	BATTERY ROOM 12 EL 277-291

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17A-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-18 may fail open on loss of instrument air. Valve is required to close for hot shutdown to support decay heat removal. A Recovery Action may be required to close valve VLV-60-11.

Ref: 51-9133191 Appendix A.20, Section A.20.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17A-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

Ref: 51-9133191 Appendix A.20, Section A.20.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17A-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

Ref: 51-9133191 Appendix A.20, Section A.20.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17A	BATTERY ROOM 12 EL 277-291

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17A-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

Ref: 51-9133191 Appendix A.20, Section A.20.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17A-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Due to the unavailability of EDG 103 (Train 12, Path B), power is not available to motor operated valve IV-38-02. IV-38-02 is a normally closed valve that fails as-is on loss of power. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve IV-38-02.

Ref: 51-9133191, Table A.18.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17B	BATTERY ROOM 11 EL 277-291

Fire Zone	Fire Zone Description
B2B	BATTERY ROOM 11 EL 277-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	The Train 12 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 support systems. Path B is credited to achieve CSD. Path B CSD is accomplished via the use of the Train 12 SDC System supported by the Train 12 RBCLC and ESW Systems. Train 11 SDC valves IV-38-01 and IV-38-13 are powered from PB-167 which will automatically transfer to Train 12.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17B	BATTERY ROOM 11 EL 277-291

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17B	BATTERY ROOM 11 EL 277-291

(f) Vital Auxiliaries	1) AC and DC Distribution System Path B is generally supported by the Train 12 AC and DC power systems. Some components in Path B systems are powered by the Train 11 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17B	**BATTERY ROOM 11 EL 277-291**

Path D – Train 12 positive pressure flow path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-12-001 Rev. 0

Eng Eval Summary: This Engineering Evaluation provides analysis of the existing NMP1 approved exemptions from 10CFR50, Appendix R and input to NFPA-805 Licensing Amendment Request (LAR) submittal with respect to previously-approved exemptions to Appendix R to 10 CFR Part 50. This document provides the evaluation, analysis and basis to demonstrate that current plant configuration and NMP1 Fire Protection Program complies with the requirements of 10CFR50.48(c), NFPA 805. The evaluation documents the similar requirements of 10CFR50, Appendix R and NFPA 805. This evaluation demonstrates that the current plant configuration and fire protection features comply with the 10CFR50.48(c) licensing basis.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17B	BATTERY ROOM 11 EL 277-291

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
B2B	Detection	D-2224	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	

Legend:

Table Field: "Required ?"

- S - Required for Chapter 4 Separation Criteria
- L - Required for NRC Approved Licensing Action
- E - Required for Existing Engineering Equivalency Evaluation
- R - Required for Risk Significance
- D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA17B of a Fire Originating In FA17B
There are no installed fire suppression systems in FA17B that could be actuated as a result of a fire originating within the fire area.

Scenario 2: Suppression Affects in FA17B of a Fire Originating Outside of FA17B
There are no installed fire suppression systems in FA17B that could be actuated as a result of a fire originating outside of the fire area.

Since there are no installed fire suppression systems in FA17B, there is no flooding potential resulting from fire suppression system actuation.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17B	BATTERY ROOM 11 EL 277-291

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17B-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

Ref: 51-9133191 Appendix A.21, Section A.21.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17B	BATTERY ROOM 11 EL 277-291

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17B-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-60-13 may fail closed on loss of instrument air. Valve is required open for hot shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-60-13.

Ref: 51-9133191 Appendix A.21, Section A.21.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17B-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-17 may fail open on loss of instrument air. Valve is required to close for hot shutdown to support decay heat removal. A Recovery Action may be required to close valve VLV-60-12.

Ref: 51-9133191 Appendix A.21, Section A.21.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
17B	BATTERY ROOM 11 EL 277-291

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17B-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. LCV-60-18 may fail open on loss of instrument air. Valve is required to throttle for hot shutdown to support decay heat removal. A Recovery Action may be required to throttle valve VLV-60-11.

Ref: 51-9133191 Appendix A.21, Section A.21.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17B-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. FCV-38-10 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-10.

Ref: 51-9133191 Appendix A.21, Section A.21.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-17B-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air for a fire in any fire area of the plant. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

Ref: 51-9133191 Appendix A.21, Section A.21.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
18	**EMERGENCY DIESEL GENERATOR 102 MISSLE ENCLOSURE EL 271**

Fire Zone	Fire Zone Description
D3	EDG 102 CONTROL CABLE MISSILE ENCLOSURE EL 271--0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	Reactor vessel make-up is required 8 hours after operation of the EC's ensues. Path D is credited via the use of the Train 12 Systems. Train 12 ERV's are opened and CS is used to flood the vessel.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD. Pressure Control for CSD is accomplished by ensuring the Train 12 ERVs are functional and Train 11 ERVs maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for CSD.	No comments.
(d) Decay Heat Removal Function	Path B HSD is credited via use of EC's 121 & 122 for Decay Heat Removal and Train 12 support systems. Path D is credited to achieve CSD. Path D CSD is accomplished via the use of the Train 12 Systems. The ERV's are opened, CS is used to flood the vessel, CTS and CTSRW are used to remove decay heat.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
18 | **EMERGENCY DIESEL GENERATOR 102 MISSLE ENCLOSURE EL 271**

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser level o Torus level o Torus temperature o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
18	EMERGENCY DIESEL GENERATOR 102 MISSLE ENCLOSURE EL 271

(f) Vital Auxiliaries	1) AC and DC Distribution System The Train 12 emergency diesel generator is credited for supplying power for the shutdown process when AC power is required to operate shutdown equipment. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
18	EMERGENCY DIESEL GENERATOR 102 MISSLE ENCLOSURE EL 271

Path D – Train 12 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-95-002 Rev. 1

Eng Eval Summary: FPEE 1-95-002 Rev. 1 - Deviations to Appendix R 3-Hour Fire Barriers
Evaluation justifies not testing 3-hour rated Promat-H ERFBS in accordance with GL 86-10, Supplement 1 fire test methodology and acceptance criteria and minor differences between installed ERFBS and HVAC duct fire barriers from tested configurations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: Nine Mile Point Unit 1 USFAR Safety Evaluation 83-02

Eng Eval Summary: Nine Mile Point Unit 1 USFAR Safety Evaluation 83-02 - Diesel Generator 103 Control Cable Protection Modification
This modification provided a 3-hr rated fire barrier around the conduits associated with DG 103. The evaluation concludes that the modification brings the area into compliance with the separation requirements of Appendix R, Section III.G.2. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 18

Fire Area Description EMERGENCY DIESEL GENERATOR 102 MISSLE ENCLOSURE EL 271

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
D3	Detection	D-2151	N	N	Y	N	Y	
	Suppression	None	N	N	N	N	N	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA18 of a Fire Originating In FA18
There are no installed fire suppression systems in FA18 that could be actuated as a result of a fire originating within the fire area.

Scenario 2: Suppression Affects in FA18 of a Fire Originating Outside of FA18
There are no installed fire suppression systems in FA18 that could be actuated as a result of a fire originating outside of the fire area.

Since there are no installed water-based fire suppression systems in FA18, there is no flooding potential resulting from fire suppression systems.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
B-40144-C, Sheet 3, Rev. 1

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
18	EMERGENCY DIESEL GENERATOR 102 MISSLE ENCLOSURE EL 271

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks. See additional discussion related to this fire area in N1-FRE-F001.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, and manual suppression from the fire brigade. A manually operated CO2 system is available but credited only for DID. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-18-001	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
18	EMERGENCY DIESEL GENERATOR 102 MISSLE ENCLOSURE EL 271

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-18-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting MSIV's IV-01-02 and IV-01-04. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the control circuits of inboard MSIV IV-01-02 and outboard MSIV IV-01-04 leaves the Main Steam line open. Inboard MSIV IV-01-02 may remain open due to an internal wire-to-wire short or a ground of cable 171-50. An internal wire-to-wire short to cable 1F-44 results in preventing closure of series outboard MSIV IV-01-04.

Ref: 51-9133191 Appendix A.22, Section A.22.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-18-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system vent valves IV-05-04R, IV-05-12, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable or an internal wire-to-wire hot short on cable 1K-157 can maintain EC system vent valves IV-05-04R and IV-05-12 open. A cable-to-cable or an internal wire-to-wire hot short on cable 1K-31 can maintain EC system vent valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line.

Ref: 51-9133191 Appendix A.22, Section A.22.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
18	EMERGENCY DIESEL GENERATOR 102 MISSLE ENCLOSURE EL 271

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-18-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between EC system drain valves IV-39-13R and IV-39-14R. These valves are required closed for HSD to support inventory control. A single cable-to-cable hot short on cable 1K-117 can maintain EC system drain valves IV-39-13R and IV-39-14R open resulting in an inventory loss to the High Pressure Section of the Condenser at penetration 15 until local action is taken to close manual valve FCV-39-16.

Ref: 51-9133191 Appendix A.22, Section A.22.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-18-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system drain valves IV-39-11R and IV-39-12R. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-111 can maintain EC system drain valves IV-39-11R and IV-39-12R open resulting in an inventory loss to the High Pressure Section of the Condenser at penetration 15 until local action is taken to close manual valve FCV 39-15.

Ref: 51-9133191 Appendix A.22, Section A.22.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-18-006	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
18	EMERGENCY DIESEL GENERATOR 102 MISSLE ENCLOSURE EL 271

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-18-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists impacting valves FCV-93-74 and FCV-93-73 by diverting flow due to wire-to-wire shorts on the following cables.
An internal wire-to-wire short on cable 171-163 spuriously opens FCV-93-74 diverting CTSRW flow to the CS system.
An internal wire-to-wire short on cable 171-160 spuriously opens FCV-93-73 diverting CTSRW flow to the CTS system.

Ref: 51-9133191 Appendix A.22, Section A.22.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-18-008	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption assumes a loss of instrument air. A loss of air fails EC Level Control Valves, LCV-60-17 & LCV-60-18, open. This could deplete the inventory in the Emergency Condenser Make-Up Water Tanks prior to the eight hours required for the EC's to remove decay heat. Local operator action to manually throttle VLV-60-11, close VLV60-12, and open/close BV-60-13 as required ensures availability of make-up inventory to EC's 121 & 122. A loss of instrument air prevents closure of CTS discharge header isolation valves IV-80-16 and IV 80-36.

Ref: 51-9133191 Appendix A.22, Section A.22.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-18-009	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
18	**EMERGENCY DIESEL GENERATOR 102 MISSLE ENCLOSURE EL 271**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-18-010	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for the credited EG-EDG103 ventilating fans BLWROT-209-05 and BLRWOT-06 and the Door 34 from proper operation due to a cable-to-cable short on the following cables:
Fire damage to cable 171-75; or, an external cable-to-cable short on cable 11DV-57 or 11DV-57B or 11DV-58 stops credited EG-EDG103 ventilation fans BLWROT-209-05 and BLWROT-209-06 and closes Door 34.

Fire damage to cable 171-71 or 171-72 stops BLWROT-209-05. Fire damage to cable 171-73 or 171-74 stops BLWROT-209-06. Fire damage to cable 171-75 can stop both BLWROT-209-05 and BLWROT-209-06.

Fire damage to cable 171-81 prevents opening Door 34.

Ref: 51-9133191 Appendix A.22, Section A.22.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a plant modification, as described in Attachment S, to reduce risk. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
19	**EMERGENCY DIESEL GENERATOR 103 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0**

Fire Zone	Fire Zone Description
D1A	EDG 103 FOUNDATION ROOM EL 250-0
D2A	EDG 103 ROOM EL 261-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	The Train 11 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path A HSD is credited via use of EC's 111 & 112 for Decay Heat Removal and Train 11 support systems. Alternatively, Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal and Train 12 Support Systems. Path A is credited to achieve CSD. Path A CSD is accomplished via the use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
19	**EMERGENCY DIESEL GENERATOR 103 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0**

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
19	EMERGENCY DIESEL GENERATOR 103 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
19	**EMERGENCY DIESEL GENERATOR 103 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0**

Path C – Train 11 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: EIR 51-9156616 Rev. 1

Eng Eval Summary: EIR 51-9156616 Rev. 1 - Nine Mile Point Unit 1 Code Compliance Evaluation for NFPA 30, Flammable and Combustible Liquids Code, 2000 Edition
Evaluation provides comparison of the plant combustible liquid storage configurations and methods to applicable NFPA 30 requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-03-002 Rev. 0

Eng Eval Summary: FPEE 0-03-002 Rev. 0 - Evaluation of Interim Action To Prevent Personnel Injury From CO2
Evaluation justifies use of temporarily placing automatic CO2 systems in alarm-only mode due to adverse trend of unplanned system actuations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-04-002 Rev. 0

Eng Eval Summary: FPEE 1-04-002 Rev. 0 - Evaluation of Fire Effects on Current Transformers and Instrument Sensing Lines and The Plant Safe Shutdown Capability
The analysis provided in this FPEE demonstrates that a postulated fire at NMP1 and the resultant fire effects on Current Transformers and instrument tubing lines will not affect the existing assumptions in the NMP1's present Appendix R Safe Shutdown Analysis. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
19	EMERGENCY DIESEL GENERATOR 103 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-003 Rev. 1

Eng Eval Summary: FPEE 1-91-003 Rev. 1 - Excessive Door Gap at Floor for Fire Doors D-84, D-108, D-111, D-245, and D-246
Evaluation determines adequacy of gaps under fire doors that exceed those allowed by NFPA 80. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
D1A	Detection	DA-2041S	N	N	Y	N	Y	
	Suppression	WP-2041	N	N	Y	N	Y	
D2A	Detection	DA-2151	N	N	Y	Y	N	
	Detection	DX-2151A/B	N	N	Y	Y	N	
	Suppression	C-2151	N	N	N	Y	N	Manual actuation of CO2

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA19 of a Fire Originating In FA19
It is not likely that activation as a result of a fire in the area could have an impact on the nuclear safety performance criteria.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
19	**EMERGENCY DIESEL GENERATOR 103 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0**

Scenario 2: Suppression Affects in FA19 of a Fire Originating Outside of FA19
Based on the limited impact a fire can have on plant suppression systems, operation of fire suppression systems outside of FA19 would not be expected to cause activation of a system within FA19 that could have an impact on the nuclear safety performance criteria. Additionally, based on the fact that the CO2 system in FA19 is mechanically isolated, operation of fire suppression systems outside of FA19 would not be expected to cause activation of the CO2 system within FA19 that could have an impact on the nuclear safety performance criteria.

Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
19	**EMERGENCY DIESEL GENERATOR 103 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0**

Based on the above assessment, safety margin is maintained.

II) Defense-in Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, and manual suppression from the fire brigade. A manually operated CO2 system is available but credited only for DID. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-19-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. A loss of instrument air could fail the decay heat removal capability by limiting the availability of the emergency condenser make up tanks from the required eight hours to some time frame less than 8 hours. LCV-60-17 and LCV-60-18 fail open on loss of instrument air. This failure causes a run out concern for the inventory contained in the emergency condenser make up tanks. In addition, BV-60-13, which provides a cross tie between the EC Make up Tanks, fails closed on loss of air. Based on NMP1 design bases document, SDBD-204 and NMP1 UFSAR Chapter V, Section E, Dated October 2001, the level control valves and the cross tie valve are required to be functional to meet the required eight hours.

Ref: 51-9133191 Appendix A.23, Section A.23.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
19	**EMERGENCY DIESEL GENERATOR 103 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-19-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

Ref: 51-9133191 Appendix A.23, Section A.23.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-19-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

Ref: 51-9133191 Appendix A.23, Section A.23.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-19-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

Ref: 51-9133191 Appendix A.23, Section A.23.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
19	**EMERGENCY DIESEL GENERATOR 103 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-19-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

Ref: 51-9133191 Appendix A.23, Section A.23.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-19-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Due to the unavailability of EG-EDG103 (Train 12, Path B), power is not available to motor operated valve IV-38-02. IV-38-02 is a normally closed valve that fails as-is on loss of power. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve IV-38-02.

Ref: 51-9133191 Table A.23.3 Fire Area Cascading Power Loss Report - SSD and Table A.23.7 Report of Procedure Directed Operator Actions

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
20	**DIESEL GENERATOR ENCLOSED CABLEWAY EL 250-0**

Fire Zone	Fire Zone Description
D1C	EDG 103 CABLE ROUTING AREA EL 250-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	Reactor vessel make-up is required 8 hours after operation of the EC's ensues. The Train 11 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path A HSD is achieved via use of EC's 111 & 112 for Decay Heat Removal. The Train 11, 125VDC System is credited to support shutdown for the duration of Battery 11. Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal. The Train 12, 125VDC System is credited to support shutdown for the duration of Battery 12 Path A is credited to achieve CSD. Path A CSD is accomplished via the use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
20	**DIESEL GENERATOR ENCLOSED CABLEWAY EL 250-0**

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser Level o Torus level o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 20
Fire Area Description DIESEL GENERATOR ENCLOSED CABLEWAY EL 250-0

(f) Vital Auxiliaries

1) AC and DC Distribution System

Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system.

The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers.

2) Cooling Systems

Various cooling water systems to support safe shutdown system operation:

o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers
o The EDG's are cooled by the EDG Cooling Water pumps
o Chilled Water for control room ventilation

3) HVAC Systems

HVAC Systems required to support post-fire shutdown are:

o Main control room ventilation (except for control room evacuation)
o Diesel generator rooms (fans operating and roll-up doors required open)

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:

Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
20	DIESEL GENERATOR ENCLOSED CABLEWAY EL 250-0

Path C – Train 11 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
D1C	Detection	DA-2041N	N	N	N	N	Y	
	Suppression	WP-2041	N	N	N	N	Y	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
20	DIESEL GENERATOR ENCLOSED CABLEWAY EL 250-0

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA20 of a Fire Originating In FA20
The activation of a suppression system would not adversely impact equipment/components credited by the NSCA.

Scenario 2: Suppression Affects in FA20 of a Fire Originating Outside of FA20
Based on the limited impact a fire can have on plant suppression systems, the operation of fire suppression systems outside of FA20 would not be expected to cause activation of a fire protection system within FA20 that could have an impact on the nuclear safety performance criteria.

Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
20	DIESEL GENERATOR ENCLOSED CABLEWAY EL 250-0

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-In-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, fixed suppression credited for DID only and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-20-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Due to the unavailability of EG-EDG103 (Train 12, Path B), power is not available to motor operated valve IV-38-02. IV-38-02 is a normally closed valve that fails as-is on loss of power. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve IV-38-02.

51-9133191 Appendix A.24, Section A.24.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
20	DIESEL GENERATOR ENCLOSED CABLEWAY EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-20-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. A loss of instrument air could fail the decay heat removal capability by limiting the availability of the emergency condenser make up tanks from the required eight hours to some time frame less than 8 hours. LCV-60-17 and LCV-60-18 fail open on loss of instrument air. This failure causes a run out concern for the inventory contained in the emergency condenser make up tanks. In addition, BV-60-13, which provides a cross tie between the EC Make up Tanks, fails closed on loss of air. Based on NMP1 design bases document, SDBD-204 and NMP1 UFSAR Chapter V, Section E, Dated October 2001, the level control valves and the cross tie valve are required to be functional to meet the required eight hours.

51-9133191 Appendix A.24, Section A.24.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-20-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

51-9133191 Appendix A.24, Section A.24.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
20	DIESEL GENERATOR ENCLOSED CABLEWAY EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-20-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

51-9133191 Appendix A.24, Section A.24.6, Subsection A.24.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-20-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

51-9133191 Appendix A.24, Section A.24.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-20-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

51-9133191 Appendix A.24, Section A.24.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
20	DIESEL GENERATOR ENCLOSED CABLEWAY EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-20-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for the Power Board 102 from the Emergency Diesel Generator EG-EDG102
The power feed cable 102-33 from EG-EDG102 to PB 102 is protected in this area by an underrated fire barrier. Therefore, the availability of EG-EDG102 cannot be assumed. The power feed cable 103-33 from EG-EDG103 to PB 103 is also in this fire area and unprotected. The result is no power available to either vital PB.

51-9133191 Appendix A.24, Section A.24.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
21	BELOW POWERBOARDS 102 & 103 EL 250-0

Fire Zone	Fire Zone Description
D1D	ROOM BELOW PB'S 102 & 103 EL 250-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	Reactor vessel make-up is required 8 hours after operation of the EC's ensues. The Train 11 CRD pump is credited to provide makeup to the RPV.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD and CSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD and CSD.	No comments.
(d) Decay Heat Removal Function	Path A HSD is achieved via use of EC's 111 & 112 for Decay Heat Removal. Train 11, 125VDC system is credited to support shutdown for the duration of Battery 11. Path B HSD is achieved via use of EC's 121 & 122 for Decay Heat Removal. The Train 12, 125VDC system is credited to support shutdown for the duration of Battery 12. Path A is credited to achieve CSD. Path A CSD is accomplished via the use of the Train 11 SDC System supported by the Train 11 RBCLC and ESW Systems.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
21	BELOW POWERBOARDS 102 & 103 EL 250-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Torus temperature o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure o Emergency Condenser Level	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
21	BELOW POWERBOARDS 102 & 103 EL 250-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Path A is generally supported by the Train 11 AC and DC power systems. Some components in Path A systems are powered by the Train 12 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area 21

Fire Area Description BELOW POWERBOARDS 102 & 103 EL 250-0

Path C – Train 11 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
D1D	Detection	DA-2041N	N	N	N	N	Y	
	Suppression	WP-2041	N	N	N	N	Y	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
21	BELOW POWERBOARDS 102 & 103 EL 250-0

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA21 of a Fire Originating In FA21
The activation of a suppression system would not adversely impact equipment/components credited by the NSCA.

Scenario 2: Suppression Affects in FA21 of a Fire Originating Outside of FA21
Based on the limited impact a fire can have on plant suppression systems, the operation of fire suppression systems outside of FA21 would not be expected to cause activation of a fire protection system within FA21 that could have an impact on the nuclear safety performance criteria.

Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
21	BELOW POWERBOARDS 102 & 103 EL 250-0

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, fixed fire suppression systems (credited for DID only) and manual suppression from the fire brigade. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-21-001	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: Due to the unavailability of EG-EDG103 (Train 12, Path B), power is not available to motor operated valve IV-38-02. IV-38-02 is a normally closed valve that fails as-is on loss of power. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve IV-38-02.

51-9133191 Appendix A.25, Section A.25.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
21	BELOW POWERBOARDS 102 & 103 EL 250-0

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-21-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. A loss of instrument air could fail the decay heat removal capability by limiting the availability of the emergency condenser make up tanks from the required eight hours to some time frame less than 8 hours. LCV-60-17 and LCV-60-18 fail open on loss of instrument air. This failure causes a run out concern for the inventory contained in the emergency condenser make up tanks. In addition, BV-60-13, which provides a cross tie between the EC Make up Tanks, fails closed on loss of air. Based on NMP1 design bases document, SDBD-204 and NMP1 UFSAR Chapter V, Section E, Dated October 2001, the level control valves and the cross tie valve are required to be functional to meet the required eight hours.

51-9133191 Appendix A.25, Section A.25.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-21-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-44-149 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support inventory control. A Recovery Action may be required to open valve VLV-28-18.

51-9133191 Appendix A.25, Section A.25.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required. No recovery action is credited for this VFDR.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
21	**BELOW POWERBOARDS 102 & 103 EL 250-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-21-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. BV-70-53 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve BV-70-53.

51-9133191, Appendix A.25, Section A.25.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-21-005	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-11 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-11.

51-9133191, Appendix A.25, Section A.25.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-21-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. FCV-38-09 may fail closed on loss of instrument air. Valve is required open for cold shutdown to support decay heat removal. A Recovery Action may be required to open valve FCV-38-09.

51-9133191 Appendix A.25, Section A.25.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
21	BELOW POWERBOARDS 102 & 103 EL 250-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-21-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for the Power Board 102. The power feeds from Emergency Diesel Generator EG-EDG102 (cable 102-33) and Power Board 101 (cable 101-5) are protected in this area by an underrated fire barrier. Therefore, the availability of EG-EDG102 cannot be assumed. The power feed cable 103-33 from EG-EDG103 to PB 103 is also in this fire area and unprotected. The result is no power available to either vital PB.

51-9133191 Appendix A.25, Section A.25.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
22	**EMERGENCY DIESEL GENERATOR 102 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0**

Fire Zone	Fire Zone Description
D1B	EDG 102 FOUNDATION ROOM EL 250-0
D2B	EDG 102 ROOM EL 261-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	Reactor vessel make-up is required 8 hours after operation of the EC's ensues. Path D is credited via the use of the Train 12 Systems. Train 12 ERV's are opened and CS is used to flood the vessel.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD. Pressure Control for CSD is accomplished by ensuring the Train 12 ERVs are functional and Train 11 ERVs maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for CSD.	No comments.
(d) Decay Heat Removal Function	Path B HSD is credited via use of EC's 121 & 122 for Decay Heat Removal. Path D is credited to achieve CSD. Path D CSD is accomplished via the use of Train 12 Systems. The ERV's are opened, CS is used to flood the vessel, and CTS and CTSRW are used to remove decay heat from the Torus.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
22	EMERGENCY DIESEL GENERATOR 102 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Emergency Condenser level o Torus level o Torus temperature o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
22	EMERGENCY DIESEL GENERATOR 102 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Path B for HSD is generally supported by Train 12 DC power systems. Path D for CSD is generally supported by the Train 12 AC and DC power systems. Some components in Path B systems are powered by the Train 11 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system and SDC Heat Exchangers o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open)	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
22	**EMERGENCY DIESEL GENERATOR 102 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0**

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:

Path D – Train 12 positive pressure flow path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-03-002 Rev. 0

Eng Eval Summary: FPEE 0-03-002 Rev. 0 - Evaluation of Interim Action To Prevent Personnel Injury From CO2
Evaluation justifies use of temporarily placing automatic CO2 systems in alarm-only mode due to adverse trend of unplanned system actuations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-04-002 Rev. 0

Eng Eval Summary: FPEE 1-04-002 Rev. 0 - Evaluation of Fire Effects on Current Transformers and Instrument Sensing Lines and The Plant Safe Shutdown Capability
The analysis provided in this FPEE demonstrates that a postulated fire at NMP1 and the resultant fire effects on Current Transformers and instrument tubing lines will not affect the existing assumptions in the NMP1's present Appendix R Safe Shutdown Analysis. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
22	EMERGENCY DIESEL GENERATOR 102 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0

Engineering Eval Id: FPEE 1-91-003 Rev. 1

Eng Eval Summary: FPEE 1-91-003 Rev. 1 - Excessive Door Gap at Floor for Fire Doors D-84, D-108, D-111, D-245, and D-246
Evaluation determines adequacy of gaps under fire doors that exceed those allowed by NFPA 80. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
D1B	Detection	DA-2041N	N	N	Y	N	Y	
	Suppression	WP-2041	N	N	Y	N	Y	
D2B	Detection	DA-2141	N	N	Y	Y	N	
	Detection	DX-2141A/B	N	N	Y	Y	N	
	Suppression	C-2141	N	N	N	Y	N	Manual actuation of CO2

Legend: Table Field: "Required ?"

S - Required for Chapter 4 Separation Criteria
L - Required for NRC Approved Licensing Action
E - Required for Existing Engineering Equivalency Evaluation
R - Required for Risk Significance
D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA22 of a Fire Originating In FA22
It is not likely that activation as a result of a fire in the area could have an impact on the nuclear safety performance criteria.

Scenario 2: Suppression Affects in FA22 of a Fire Originating Outside of FA22
Based on the limited impact a fire can have on plant suppression systems, operation of fire suppression systems outside of FA22 would not be expected to cause activation of a system within FA22 that could have an impact on the nuclear safety performance criteria. Additionally, based on the fact that the CO2 system in FA22 is mechanically isolated, operation of fire suppression systems outside of FA22 would not be expected to cause activation of the CO2 system within FA22 that could have an impact on the nuclear safety performance criteria.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
22	EMERGENCY DIESEL GENERATOR 102 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0

Preaction sprinkler systems require both the fusing of a sprinkler head and the opening of the preaction valve as a result of actuation of a heat or smoke detector. Therefore, significant overflow or migration of fire suppression system water discharge would not be expected to occur and adversely affect components or equipment credited in the NSCA in an adjacent or otherwise unaffected area.

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks. See additional discussion related to this fire area in N1-FRE-F001.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
22	EMERGENCY DIESEL GENERATOR 102 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0

Results of the defense in DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, a manual CO2 system (credited for DID only) and manual suppression from the fire brigade fixed fire suppression systems for partial coverage. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-22-001	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-22-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting MSIV's IV-01-02 and IV-01-04. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the control circuits of MSIV's IV-01-02 and IV-01-04 leaves the MS line open. MSIV IV-01-02 may remain open due to an internal wire-to-wire short or a ground of cable 171-50. An internal wire-to-wire short on cable 1F-44 results in preventing closure of series MSIV IV-01-04.

51-9133191 Appendix A.26, Section A.26.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
22	EMERGENCY DIESEL GENERATOR 102 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-22-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the EC system vent valves IV-05-04R, IV-05-12, IV-05-02R, and IV-05-03R. These valves are required closed for HSD to support inventory control. A cable-to-cable or an internal wire-to-wire hot short on cable 1K-157 can maintain the EC system vent valves IV-05-04R and IV-05-12 open. A cable-to-cable or an internal wire-to-wire hot short on cable 1K-31 can maintain the EC system vent valves IV-05-02R and IV-05-03R open. The multiple spurious combination results in an inventory loss to the main steam line.

51-9133191 Appendix A.26, Section A.26.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-22-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting the EC system drain valves IV-39-13R and IV-39-14R. These valves are required closed for HSD to support inventory control. A cable-to-cable hot short on cable 1K-117 can maintain the EC system drain valves IV-39-13R and IV-39-14R open, resulting in an inventory loss to the High Pressure Section of the Condenser at penetration 15.

51-9133191 Appendix A.26, Section A.26.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-22-005	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
22	**EMERGENCY DIESEL GENERATOR 102 FOUNDATION ROOM EL 250-0 AND DIESEL GENERATOR ROOM EL 261-0**

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-22-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in the area for the Containment Spray Raw Water system. CTSRW is required to support the decay heat removal function. Both loops of the credited secondary decay heat removal function can be lost.
An internal wire-to-wire short on cable 171-163 spuriously opens FCV-93-74 diverting flow from pump PMP-93-03 to the CS system.
An internal wire-to-wire short on cable 171-160 spuriously opens FCV-93-73 diverting flow from pump PMP-93-04 to the CTS system.
Diversion of the CTSRW flow paths away from the CTSRW heat exchangers results in a loss of Decay Heat Removal.

51-9133191 Appendix A.26, Section A.26.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with no further action required.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-22-007	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. A loss of instrument air could fail the decay heat removal capability by limiting the availability of the emergency condenser make up tanks from the required eight hours to some time frame less than 8 hours. LCV-60-17 and LCV-60-18 fail open on loss of instrument air. This failure causes a run out concern for the inventory contained in the emergency condenser make up tanks. In addition, BV-60-13, which provides a cross tie between the EC Make up Tanks, fails closed on loss of air. Based on NMP1 design bases document, SDBD-204 and NMP1 UFSAR Chapter V, Section E, Dated October 2001, the level control valves and the cross tie valve are required to be functional to meet the required eight hours.

51-9133191 Appendix A.26, Section A.26.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
23	**POWER BOARD 102 ROOM EL 261-0**

Fire Zone	Fire Zone Description
D2C	POWER BOARD 102 ROOM EL 261-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No Comments.
(b) Inventory Control Function	Reactor vessel make-up is required 8 hours after operation of the EC's ensues. Path D is credited via the use of the Train 12 Systems. Train 12 ERV's are opened and CS is used to flood the vessel.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD. Pressure Control for CSD is accomplished by ensuring the Train 12 ERVs are functional and Train 11 ERVs maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for CSD.	No comments.
(d) Decay Heat Removal Function	Path B HSD is credited via use of EC's 121 & 122 for Decay Heat Removal and Train 12 support systems. Path D is credited to achieve CSD. Path D CSD is accomplished via the use of the Train 12 Systems. The ERV's are opened, CS is used to flood the vessel, CTS and CTSRW are used to remove decay heat..	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
23	POWER BOARD 102 ROOM EL 261-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Torus temperature o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure o Containment Spray pump discharge pressure o Emergency Condenser Level	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
23	POWER BOARD 102 ROOM EL 261-0

(f) Vital Auxiliaries	1) AC and DC Distribution System Path B HSD is generally supported by the Train 12 DC power systems. Path D CSD is generally supported by the Train 12 AC and DC power systems. Some components in Path D systems are powered by the Train 11 AC and/or DC power system. The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers. 2) Cooling Systems Various cooling water systems to support safe shutdown system operation: o RBCLC and ESW provide cooling to the control room ventilation system o The EDG's are cooled by the EDG Cooling Water pumps o Chilled Water for control room ventilation 3) HVAC Systems HVAC Systems required to support post-fire shutdown are: o Main control room ventilation (except for control room evacuation) o Diesel generator rooms (fans operating and roll-up doors required open) Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
23	POWER BOARD 102 ROOM EL 261-0

are as follows:

Path D – Train 12 positive pressure flow path.

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-03-002 Rev. 0

Eng Eval Summary: FPEE 0-03-002 Rev. 0 - Evaluation of Interim Action To Prevent Personnel Injury From CO2
Evaluation justifies use of temporarily placing automatic CO2 systems in alarm-only mode due to adverse trend of unplanned system actuations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-04-002 Rev. 0

Eng Eval Summary: FPEE 1-04-002 Rev. 0 - Evaluation of Fire Effects on Current Transformers and Instrument Sensing Lines and The Plant Safe Shutdown Capability
The analysis provided in this FPEE demonstrates that a postulated fire at NMP1 and the resultant fire effects on Current Transformers and instrument tubing lines will not affect the existing assumptions in the NMP1's present Appendix R Safe Shutdown Analysis. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
23	POWER BOARD 102 ROOM EL 261-0

Engineering Eval Id: FPEE 1-91-003 Rev. 1

Eng Eval Summary: FPEE 1-91-003 Rev. 1 - Excessive Door Gap at Floor for Fire Doors D-84, D-108, D-111, D-245, and D-246
Evaluation determines adequacy of gaps under fire doors that exceed those allowed by NFPA 80. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ?					Notes
			S	L	E	R	D	
D2C	Detection	DX-2123A/B	N	N	Y	Y	N	
	Suppression	C-2123	N	N	N	N	Y	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria
L - Required for NRC Approved Licensing Action
E - Required for Existing Engineering Equivalency Evaluation
R - Required for Risk Significance
D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA23 of a Fire Originating In FA23
It is not likely that activation as a result of a fire in the area could have an impact on the nuclear safety performance criteria.

Scenario 2: Suppression Affects in FA23 of a Fire Originating Outside of FA23
Based on the limited impact a fire can have on plant suppression systems and the fact that the CO2 system in FA23 is mechanically isolated, operation of fire suppression systems outside of FA23 would not be expected to cause activation of the CO2 system within FA23 that could have an impact on the nuclear safety performance criteria.

Since there are no installed water-based fire suppression systems in FA23, there is no flooding potential resulting from fire suppression system actuation.

Reference: Document No.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
23	**POWER BOARD 102 ROOM EL 261-0**

N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks. See additional discussion related to this fire area in N1-FRE-F001.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, a manual CO2 system (credited for DID only) and manual suppression from the fire brigade and fixed fire suppression systems for partial coverage. Pre-fire plans documented in N1-PFP-0101.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
23	POWER BOARD 102 ROOM EL 261-0

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-23-001	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-23-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area between MSIV's IV-01-02 and IV-01-04. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the control circuits of MSIV's IV-01-02 and IV-01-04 leaves the MS line open. MSIV IV-01-02 can suffer damage to its control circuit leaving the valve open. An external cable-to-cable short on cable 1F-44 results in SV 20-1/01-04 remaining energized preventing closure of series MSIV IV-01-04. The inventory loss continues until local manual action is taken to close IV-01-04.

Ref: 51-9133191 Appendix A.27 Section A.27.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
23	POWER BOARD 102 ROOM EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-23-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system drain valves IV-39-13R and IV-39-14R. These valves are required closed for HSD to support inventory control. A single cable-to-cable hot short on cable 1K-117 can maintain EC system drain valves IV-39-13R and IV-39-14R open resulting in an inventory loss to the High Pressure Section of the Condenser at penetration 15.

Ref: 51-9133191 Appendix A.27 Section A.27.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-23-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system vent valves IV-05-02R, IV-05-03R, IV-05-04R, and IV-05-12. These valves are required closed for HSD to support inventory control. Internal wire-to-wire shorts on cable 1K-31 can maintain EC system vent valves IV-05-02R and IV-05-03R open. Internal wire-to-wire shorts on cable 1K-157 can maintain EC system vent valves IV-05-04R and IV-05-12 open. The result is an open vent line from EC121 and EC122 to a main steam line.

Ref: 51-9133191 Appendix A.27 Section A.27.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-23-005	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
23	POWER BOARD 102 ROOM EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-23-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption assumes a potential loss of instrument air for a fire in any area of the plant. A loss of air fails EC Level Control Valves, LCV-60-17 & LCV-60-18, open. This could deplete the inventory in the Emergency Condenser Make-Up Water Tanks prior to the eight hours required for the EC's to remove decay heat. Local operator action to manually throttle VLV-60-11, close VLV-60-12, and open/close BV-60-13 as required ensures availability of make-up inventory for EC's 121 & 122.

Ref: 51-9133191 Appendix A.27 Section A.27.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-23-007	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
24	**POWER BOARD 103 ROOM EL 261-0**

Fire Zone	Fire Zone Description
D2D	POWER BOARD 103 ROOM EL 261-0

Regulatory Basis: NFPA 805 Section 4.2.4.2: Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method	Comment
(a) Reactivity Control Function	Manual Scram is credited via the push buttons in Control Room which actuate Backup Scram SOV's.	No comments.
(b) Inventory Control Function	Reactor vessel make-up is required 8 hours after operation of the EC's ensues. Path C is utilized via the use of the Train 11 Systems. Train 11 ERV's are opened and CS is used to flood the vessel.	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.
(c) Pressure Control	Pressure Control for HSD is accomplished by ensuring the ERVs are maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for HSD. Pressure Control for CSD is accomplished by ensuring the Train 11 ERVs are functional and Train 12 ERVs maintained closed while utilizing systems associated with the credited Inventory Control and Decay Heat Removal functions for CSD.	No comments.
(d) Decay Heat Removal Function	Path A HSD is achieved via use of EC's 111 & 112 for Decay Heat Removal and Train 11 support systems. Path C is credited to achieve CSD. Path C CSD is accomplished via the use of the Train 11 Systems. The ERV's are opened, CS is used to flood the vessel, CTS and CTSRW are used to remove decay heat..	Variance from the deterministic requirements of NFPA 805 exists for this performance goal; Fire Risk Evaluation required.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
24	POWER BOARD 103 ROOM EL 261-0

(e) Process Monitoring Function	The following process monitoring functions are provided to support post-fire shutdown: o Reactor coolant level o Reactor coolant pressure o Reactor coolant temperature o Emergency Diesel Generator parameters , a) Generator Frequency or Speed, b) Generator Voltage and Current, c) Bypass Fuel Flow Glass, d) Engine Coolant Inlet Temperature, e) Fuel Oil Pressure, f) Lube Oil Temperature, g) Lube Oil Pressure, h) Raw Water Strainer D/P, i) Fuel Oil Day Tank Level j) Fuel Oil Storage Tank Level o Torus level o Torus temperature o Drywell pressure o Drywell temperature o Containment Spray water temperature o Containment Spray pump discharge pressure o Emergency Condenser Level	No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
24	POWER BOARD 103 ROOM EL 261-0

(f) Vital Auxiliaries

1) AC and DC Distribution Systems

The Train 11 emergency diesel generator is credited for supplying power for the shutdown process when AC power is required to operate shutdown equipment.

The vital 125V DC distribution system is credited to support shutdown as it supplies control power to various 125V DC control panels including switchgear breaker controls. The 125V DC distribution panels may also supply power to the 120V AC reactor protection distribution panels via static inverters. These distribution panels supply power for instrumentation necessary to provide process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries supply any required control power during the interim time period required for the diesel generator to become operational. Once the diesel is operational, the 125V DC distribution system can be powered from the diesel through the battery chargers.

2) Cooling Systems

Various cooling water systems to support safe shutdown system operation:

o The EDG is cooled by the EDG Cooling Water pump
o CTS and CTSRW cool the torus for the secondary cooldown method
o Chilled Water for control room ventilation

3) HVAC Systems

HVAC Systems required to support post-fire shutdown are:

o Main control room ventilation
o Diesel generator rooms (fans and roll-up doors)

Paths developed for the control room ventilation system do not necessarily match the paths for process systems. The paths of the ventilation systems that are credited are as follows:

No comments.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
24	**POWER BOARD 103 ROOM EL 261-0**

Path C – Train 11 positive pressure flow path

Reference Documents: 51-9133191 (NMP1 Nuclear Safety Capability Assessment)

ENGINEERING EVALUATIONS

Engineering Eval Id: EIR 51-9077284 Rev. 0

Eng Eval Summary: EIR 51-9077284 Rev. 0 - Nine Mile Point Unit 1 Code Compliance Reviews
Evaluation provides comparison of the plant fire protection systems and design features to applicable NFPA 10, 12, 12A, 13, 14, 15, 20, 24, 72A, and 72E requirements. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-03-002 Rev. 0

Eng Eval Summary: FPEE 0-03-002 Rev. 0 - Evaluation of Interim Action To Prevent Personnel Injury From CO2
Evaluation justifies use of temporarily placing automatic CO2 systems in alarm-only mode due to adverse trend of unplanned system actuations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 0-92-002 Rev. 0

Eng Eval Summary: FPEE 0-92-002 Rev. 0 - Inaccessible Fire Damper Operability
Evaluation provides exception for testing fire dampers deemed inaccessible by extending testing to the next test interval provided previous test was satisfactory and damper has not been modified. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-04-002 Rev. 0

Eng Eval Summary: FPEE 1-04-002 Rev. 0 - Evaluation of Fire Effects on Current Transformers and Instrument Sensing Lines and The Plant Safe Shutdown Capability
The analysis provided in this FPEE demonstrates that a postulated fire at NMP1 and the resultant fire effects on Current Transformers and instrument tubing lines will not affect the existing assumptions in the NMP1's present Appendix R Safe Shutdown Analysis. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-90-020 Rev. 1

Eng Eval Summary: FPEE 1-90-020 Rev. 1 - NMP-1 Fire-Rated Penetration Seal Detail FS-8
Evaluation determines the acceptability of concrete penetration seals having a 6-1/2 in. depth versus the 8 in. depth tested by NMPC. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
24	POWER BOARD 103 ROOM EL 261-0

Engineering Eval Id: FPEE 1-91-003 Rev. 1

Eng Eval Summary: FPEE 1-91-003 Rev. 1 - Excessive Door Gap at Floor for Fire Doors D-84, D-108, D-111, D-245, and D-246
Evaluation determines adequacy of gaps under fire doors that exceed those allowed by NFPA 80. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

Engineering Eval Id: FPEE 1-91-006 Rev. 0

Eng Eval Summary: FPEE 1-91-006 Rev. 0 - Interior Conduit Sealing Walkdowns
Evaluation documents plant-wide walkdown conducted to determine adequacy of internal conduit seal installations. Evaluation is an appropriate use of the engineering evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
D2D	Detection	DX-2113A/B	N	N	Y	Y	N	
	Suppression	C-2113	N	N	N	N	Y	

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria
L - Required for NRC Approved Licensing Action
E - Required for Existing Engineering Equivalency Evaluation
R - Required for Risk Significance
D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

Fire suppression activities will have no impact on achieving the nuclear safety performance criteria.

Scenario 1: Suppression Affects in FA24 of a Fire Originating In FA24
It is not likely that activation as a result of a fire in the area could have an impact on the nuclear safety performance criteria.

Scenario 2: Suppression Affects in FA24 of a Fire Originating Outside of FA24
Based on the limited impact a fire can have on plant suppression systems and the fact that the CO2 system in FA24 is mechanically isolated, operation of fire suppression systems outside of FA24 would not be expected to cause activation of the CO2 system within FA24 that could have an impact on the nuclear safety performance criteria.

Since there are no installed water-based fire suppression systems in FA24, there is no flooding potential resulting from fire suppression system actuation.

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
24	**POWER BOARD 103 ROOM EL 261-0**

Reference: Document No.
N1-FSS-F001 Rev.1, Detailed Fire Modeling
51-9175332-000, NMP-1 NFPA 805 Fire Suppression Effects Analysis

FIRE RISK SUMMARY

A risk-informed, performance-based FRE was performed to address the VFDRs in this fire area. The acceptability of these VFDRs was based on 1) the measured change in CDF and LERF, which were within the acceptance guidelines given in Figures 3 and 4 of Regulatory Guide 1.174, and 2) the maintenance of DID and safety margin. The FRE determined that the applicable risk, DID and safety margin criteria were satisfied.

ΔCDF: Redacted

ΔLERF:

I) Safety Margin

Results of the safety margin assessment are summarized as follows:

Fire Modeling- Conservatisms in the analysis ensure safety margins are maintained based on the fire modeling process/approach, fire modeling inputs, fire modeling tools as described in N1-FRE-F001 and the corresponding Fire PRA notebooks.

Plant System Performance- The safety margins inherent in the analyses for the plant design basis events are captured in the internal event PRA notebooks. See additional discussion related to this fire area in N1-FRE-F001.

PRA Logic Model- The Fire PRA was developed based on the internal events model which includes a success criteria consistent with design basis. The Fire PRA follows industry guidelines consistent with the Fire PRA peer review process.

Based on the above assessment, safety margin is maintained.

II) Defense-in-Depth (DID)

Results of the DID assessment are summarized as follows:

Echelon 1- Combustible control implemented via Procedure GAP-INV-02. Hot work control implemented via Procedure GAP-FPP-02.

Echelon 2- The fire area includes fire detection systems, a manual CO2 system (credited for DID only) and manual suppression from the fire brigade and fixed fire suppression systems for partial

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
24	POWER BOARD 103 ROOM EL 261-0

coverage. Pre-fire plans documented in N1-PFP-0101.

Echelon 3- Walls, floors, ceilings, structural elements and penetration seals are rated or evaluated as adequate for the hazard.
Guidance provided to operations personnel detailing the required success path(s), for applicable recovery actions.

No improvements required for DID.

VARIANCE FROM DETERMINISTIC REQUIREMENTS

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-24-001	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-24-002	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting MSIV's IV-01-02 and IV-01-04. These valves are required closed for HSD and CSD to support inventory control. Fire damage to the control circuits of inboard MSIV IV-01-02 and outboard MSIV IV-01-04 leaves the Main Steam line open. Inboard MSIV IV-01-02 may remain open either due to fire damage to cable 171-50, an internal wire-to-wire short or ground, or a loss of Train 12 power. An internal wire-to-wire short on cable 1F-44 results in preventing closure of series outboard MSIV IV-01-04.

51-9133191 Appendix A.28, Section A.28.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action and a plant modification, as described in Attachment S, to reduce risk.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
24	POWER BOARD 103 ROOM EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-24-003	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A separation concern exists for a postulated fire in this area impacting EC system drain valves IV-39-11R and IV-39-12R. These valves are required closed for HSD to support inventory control. A single cable-to-cable hot short on cable 1K-111 can maintain EC system drain valves IV-39-11R and IV-39-12R open resulting in an inventory loss to the Main Steam Drain Line, until manual action is taken to close manual valve FCV-39-15.

51-9133191 Appendix A.28, Section A.28.4 Separation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-24-004	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A deterministic assumption exists for potential loss of instrument air. A loss of instrument air could fail the decay heat removal capability by limiting the availability of the emergency condenser make up tanks from the required eight hours to some time frame less than 8 hours. LCV-60-17 and LCV-60-18 fail open on loss of instrument air. This failure causes a run out concern for the inventory contained in the emergency condenser Make up Water Tanks. In addition, BV-60-13, which provides a cross tie between the EC Make up Tanks, fails closed on loss of air. Based on NMP1 design bases document, SDBD-204, the level control valves and the cross tie valve are required to be functional to meet the required eight hours.

51-9133191 Appendix A.28, Section A.28.6 Instrument Air

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within the Fire PRA using HRA analysis methods.

[Ref. N1-FRE-F001, Rev 0]

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-24-005	N/A	N/A	N/A

VFDR: Number intentionally left blank.

Disposition: N/A

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
24	POWER BOARD 103 ROOM EL 261-0

VFDR ID	Status	Corrective Action Reference	FRE / Change Evaluation / Mod Reference
VFDR-24-006	Closed	N/A	N1-FRE-F001, Rev 0 (Fire Risk Evaluation Report)

VFDR: A spurious actuation concern exists for potential fire in this area for CTSRW Pump 93-04. If Train 12 power is available, Path D CTSRW may spuriously actuate. CTSRW Pump PMP-93-04 can spuriously start due to an internal wire-to-wire short on cable 103-22. In addition, an internal wire-to-wire short on cable 103-106 opens valve FCV-93-73 to connect the CTSRW system to the CTS header resulting in an increasing Torus water level. The increase in Torus water level will lead the Plant operators to enter the EOPs and will cause a deviation from the credited safe shutdown strategy.

51-9133191 Appendix A.28 Section A.28.5 Spurious Actuation Concerns

Disposition: This VFDR has been evaluated and it has been determined that the risk, safety margin, and defense-in-depth meet the acceptance criteria of NFPA 805 section 4.2.4 with a Recovery Action credited. The Recovery Action has been evaluated for feasibility and reliability within EIR 51-9156521-000.

[Ref. N1-FRE-F001, Rev 0]

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area | **Fire Area Description**
EXT | **EXTERNAL TO PLANT**

Fire Zone | **Fire Zone Description**
EXT | EXTERNAL TO PLANT

Regulatory Basis: NFPA 805 Section 4.2.3.1 - Deterministic Approach

Reference Documents:

ENGINEERING EVALUATIONS

Engineering Eval Id: FPEE 1-90-019 Rev. 0

Eng Eval Summary: FPEE 1-90-019 Rev.0 - Hydrogen Storage Tank Explosion Potential
Evaluation provides analysis of impact of a potential explosion involving the bulk hydrogen storage system on west wall of Reactor Building and the north wall of the Turbine Building. The conclusion is that there is insufficient hydrogen gas material to develop an unconfined vapor cloud explosion. Evaluation is an appropriate use of the engineering
evaluation process, is judged to be of acceptable quality and reflects current plant conditions.

REQUIRED FIRE PROTECTION SYSTEMS/FEATURES

Fire Zone	Category	ID	Required ? S	L	E	R	D	Notes
EXT	Detection	DA-8042	N	N	Y	N	N	
	Detection	DA-8082	N	N	N	N	N	
	Detection	DA-8092	N	N	N	N	N	
	Detection	DA-8112	N	N	N	N	N	
	Detection	DA-8131	N	N	N	N	N	
	Detection	DA-8141	N	N	N	N	N	
	Suppression	WD-8042	N	N	Y	N	N	
	Suppression	WD-8072	N	N	N	N	N	
	Suppression	WD-8082	N	N	N	N	N	
	Suppression	WD-8092	N	N	N	N	N	
	Suppression	WD-8112	N	N	N	N	N	
	Suppression	WD-8121	N	N	N	N	N	
	Suppression	WD-8131	N	N	N	N	N	
	Suppression	WD-8141	N	N	N	N	N	
	Suppression	WD-8184	N	N	Y	N	N	

Table B-3 NFPA 805 Chapter 2 – Nuclear Safety Transition - Fire Area Assessment Worksheet

Fire Area	Fire Area Description
EXT	EXTERNAL TO PLANT

Legend:

Table Field: "Required ?"
S - Required for Chapter 4 Separation Criteria L - Required for NRC Approved Licensing Action E - Required for Existing Engineering Equivalency Evaluation R - Required for Risk Significance D - Required to maintain adequate balance of Defense-in-Depth in a Change Evaluation or Fire Risk Evaluation

Fire Suppression Activities Effect on Nucear Safety Performance Criteria

N/A

FIRE RISK SUMMARY

N/A - Yard Areas

D. NEI 04-02 Non-Power Operational Modes Transition

6 Pages Attached

NFPA 805 Section 1.3.1 Nuclear Safety Goal

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

Implementing Guidance F.1

Review Existing Outage Management Processes

Define Higher Risk Evolutions (HREs), if not already defined in plant outage management procedures. The HRE definition should consider the following:

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

Review

HREs are discussed in EIR 51-9171174-000, "Nine Mile Point 1 Nuclear Power Station – NFPA 805 Transition – Non Power Operations Component Pinch Point Analysis," Section 3.

"Time to boil" has been defined in CENG procedure CNG-OM-1.01-1001, Section 3.20, and is addressed in NMPNS procedure NIP-OUT-01.

NIP-OUT-01 defines Higher Risk Evolutions as: "Outage activities, plant configurations or conditions during shutdown where the plant is more susceptible to an event causing the loss of a key safety function or the number of key safety functions drops below the shutdown safety criteria." This procedure considers the HRE attributes delineated above and in FAQ-07-0040 as follows:

NIP-OUT-01 ensures that Higher Risk Evolutions be identified and communicated to plant personnel with applicable precautions and / or contingency plans clearly identified, e.g., on the Outage Schedule Shutdown Safety Review (SSR) reports. A SSR is conducted before all refueling, planned maintenance, and forced outages, and after significant changes are made to the final approved outage schedule.

Additionally, contingency plans are defined in NIP-OUT-01, Attachment 7, "Shutdown Safety Contingency Plan," which discuss:

- Calculated time to boil
- Reactor inventory control
- Spent fuel pool inventory control
- Core decay heat removal
- Spent fuel pool decay heat removal
- Secondary Containment
- Primary Containment

NIP-OUT-01, Attachment 9, "Unit 1 Shift Outage Safety Assessment Equal to Or Less than 212°F," further addresses the Time to Boil with regards to the Reactor Cavity, the time to reach 140°F in the Spent Fuel Pool, and the time for Secondary Containment closure, as well as required contingency controls for Key Safety Functions. These considerations address those stipulated in FAQ-07-0040.

Reference Documents

- EIR 51-9171174-000, Nine Mile Point 1 Nuclear Power Station – NFPA 805 Transition – Non Power Operations Component Pinch Point Analysis, Section 3
- CNG-OM-1.01-1001, Shutdown Safety Management Program
- NIP-OUT-01, Shutdown Safety
- FAQ 07-0040, Rev. 4, Non-Power Operations Clarifications

Implementing Guidance F.2

Identify Components and Cables

The identification of systems and components to be included in this NPO review begins with the identification of the plant operational states (POSs) that need to be considered.

Identify the various operational states that a plant goes through during NPO, and which ones are the most risk significant.

Review

KSFs are identified and discussed in EIR 51-9137629, “Nine Mile Point 1 Non- Power Operations KSF Equipment List,” Section 3.2.

NPO systems/paths are discussed in:

- EIR 51-9137629, Sections 4.0 through 4.4.
- EIR 51-9137629, Appendix A: NPO Safe Shutdown KSF Success Path Codes.
- EIR 51-9137629, Appendix B: Non Power Operations Component List.
- EIR 51-9137629, Appendix C: Non Power Operation Component Report by KSF.
- EIR 51-9137629, Appendix D: Key Safety Function Diagrams.

Components required for NPO (different than SSD analysis) are discussed in EIR 51-9137629, Section 3.4, Table 3-4: SSD vs. NPO Position Differences.

Reference Documents

- EIR 51-9137629-000, Nine Mile Point 1 Non- Power Operations KSF Equipment List

Implementing Guidance F.3

Perform Fire Area Assessments (Identify pinch points – plant locations where a single fire may damage all success paths of a KSF).

Identify locations where:

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

Fire modeling may be used to determine if postulated fires in a fire area are expected to damage equipment (and cabling) thereby eliminating a pinch point. Fire modeling (if used) should include a treatment of safety margin (MEFS/LFS or other treatment) to account for uncertainties/accuracy of the fire model used.

Review

Loss of KSFs are evaluated in EIR 51-9171174, Appendix B: NMP1 Pinch Point Assessment.

Note: The loss of the KSFs were performed on a Fire Zone basis, rather than a Fire Area basis. Appendix B documents the loss of the KSFs for each Fire Zone.

Compliance Methodology can be found in EIR 51-9171174, Section 4.0.

Documentation of Results can be found in EIR 51-9171174, Appendix B: NMP1 Pinch Point Assessment.

Supporting Appendices include Appendix A: NMP1 NPO Matrix, and Appendix C: Non-Power Operations FZ Pinch Point Impact Report.

Fire modeling was not used to determine if a postulated fire would be expected to damage required equipment.

Reference Documents

EIR 51-9171174, Nine Mile Point 1 Nuclear Power Station – NFPA 805 Transition – Non-Power Operations Component Pinch Point Analysis

Implementing Guidance F.4

Manage Risks Associated with Fire-Induced Vulnerabilities During the Outage

- During those NPO evolutions where risk is relatively low.

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

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

- During those NPO evolutions that are defined as HREs,

  Additional fire protection defense in depth measures will be taken during HREs by:

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

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

  - Prohibition or limitation of hot work in fire areas during periods of increased vulnerability.
  - Verification of operable detection and /or suppression in the vulnerable areas.
  - Prohibition or limitation of combustible materials in fire areas during periods of increased vulnerability.

- Plant configuration changes (e.g., removing power from equipment once it is placed in its desired position).
- Provision of additional fire patrols at periodic intervals or other appropriate compensatory measures (such as surveillance cameras) during increased vulnerability.
- Use of recovery actions to mitigate potential losses of key safety functions.
- 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 DID.

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

Review

Recommendations to mitigate potential fire damage are documented in EIR 51-9171174, Appendix B, NMP1 Pinch Point Assessment.

Recommended changes to Outage Procedure NIP-OUT-01 will be made to reflect the evaluations performed. See Implementation Items in Attachment S.

Reference Documents

EIR 51-9171174, Nine Mile Point 1 Nuclear Power Station – NFPA 805 Transition – Non Power Operations Component Pinch Point Analysis
NIP-OUT-01, Shutdown Safety

E. NEI 04-02 Radioactive Release Transition

55 Pages Attached

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA1	R1A	Containment Spray Pump Room (112, 122), General Floor Area, and East on elevations 198'/237'	PFP-RX198-02	Northeast Reactor Building - 198'/218' Elevation	Y	The floor drain sump, having a capacity of 58 gallons, automatically transfers potentially contaminated water runoff from fire suppression activities inside the Reactor Building to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system	Pre-Fire Plans specify normal building exhaust in the Cleanup (CU) Sludge Drain Tank Room and Equipment and Personnel Lock Room at 237' elevation will remove smoke from this area. Approximately 5200 CFM of exhaust is available in this area.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.
			PFP-RX237-01	East Reactor Building - 237' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal	Pre-Fire Plans specify smoke from the general area can be exhausted utilizing the normal Reactor Building exhaust system. Exhaust registers are located in the North portion of the building. Approximately 6800 CFM of exhaust capacity is available in this area. Smoke from the Southeast stairwell and emergency escape lock areas can be removed through the normal building exhaust in the Emergency Escape Lock Room.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Approximately 1200 CFM of exhaust capacity is available in this area.		
FA1	R1C	Access Stairwell, Southeast on elevations 237'/261'	PFP-RX237-01	East Reactor Building - 237' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify smoke from the general area can be exhausted utilizing the normal Reactor Building exhaust system. Exhaust registers are located in the North portion of the building. Approximately 6800 CFM of exhaust capacity is available in this area. Smoke from the Southeast stairwell and emergency escape lock areas can be removed through the normal building exhaust in the Emergency Escape Lock Room. Approximately 1200 CFM of exhaust capacity is available in this area.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
			PFP-RX261-01	East Reactor Building - 261' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Reactor Building in this area include the air lock at column line K-11 and air lock at the Track Bay. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 3000 CFM of exhaust capacity is available from this area. Exhaust registers are located in the CU filter rooms, CU heat exchanger room and the CU aux. pump room. Flow paths from the Reactor Building in this area include the air lock at column line K-11 and air lock at the Track Bay. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA1	R1D	Core Spray Pump Room (121, 122), Containment Spray Pump Room, and Protective Clothing Change Area on elevations 198'/237'	PFP-RX198-04	Southeast Reactor Building - 198'/218' Elevation	N	N/A	N/A	N/A	N/A
FA1	R2A	General Floor Area, East on elevation 261'	PFP-RX261-01	East Reactor Building - 261' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Reactor Building in this area include the air lock at column line K-11 and air lock at the Track	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 3000 CFM of exhaust capacity is available from this area. Exhaust registers are located in the CU filter rooms, CU heat exchanger room and the CU aux. pump room. Flow paths from the Reactor Building in this area include the air lock at column line K-11 and air lock at the Track Bay. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						Bay. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.			
FA1	R3A	General Floor Area, East on elevation 281'	PFP-RX281-01	East Reactor Building - 281' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation system may be used to remove smoke from this area. An exhaust grill located at the exhaust trunk near column line P-12 has approximately 2200 CFM of capacity. The exhaust trunk has a significant exhaust capacity and a portable smoke ejector with flexible duct routed to this area could be used to speed up smoke removal.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA1	R4A	General Floor Area, East on elevation 298'	PFP-RX298-01	East Reactor Building - 298' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation exhaust system will remove smoke from this area. Approximately 4800 CFM of exhaust capacity is available at the exhaust trunk located near column line P-12. An additional 2000 CFM of exhaust capacity is available in the ECIV Room; however, smoke in this room will actuate the Halon 1301 fire suppression system and close the damper to the exhaust duct.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.
FA1	R4C	Emergency Condenser Isolation Valve Room on elevation 298'	PFP-RX298-01	East Reactor Bldg. - 298' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these	Pre-Fire Plans specify normal building ventilation exhaust system will remove smoke from this area. Approximately 4800 CFM of exhaust capacity is available at the exhaust trunk located near column line P-12. An additional 2000 CFM of exhaust capacity is available in	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	the ECIV Room; however, smoke in this room will actuate the Halon 1301 fire suppression system and close the damper to the exhaust duct.		
			PFP-RX298-02	Emergency Condenser Iso. Vlv. Room - 298' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation exhaust can remove smoke from this area. Approximately 2000 CFM of exhaust capacity is available in the room and approximately 4800 CFM of exhaust capacity is available outside the room at the exhaust trunk near column line P-12. Dampers for room exhaust actuate closed when the Halon system actuates. These dampers must be reopened to establish smoke removal by operating the "Damper Open" button located on the west wall outside the ECIV Room.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA1	R5A	General Floor Area, East on elevation 318'	PFP-RX318-01	East Reactor Building - 318' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 7000 CFM of exhaust capacity is available at the exhaust system trunk near column line P-12.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.
FA1	R6A	General Floor Area, East on elevation 340'	PFP-RX340-01	Refuel Floor - 340' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 26,400 CFM of exhaust capacity is available near the roof truss area above 340' elevation. Flow paths from the Reactor Building in this area include the	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Reactor Building in this area include the Turbine Building stairwell near column line J-12 at 333' elevation. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	Turbine Building stairwell near column line J-12 at 333' elevation. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.		
FA2	R1B	Containment Spray Pump Room (111, 121), and General Floor Area West on elevations 198'/237'	PFP-RX198-01	Northwest Reactor Building - 198'/218' Elevation	Y	The floor drain sump, having a capacity of 58 gallons, automatically transfers potentially contaminated water runoff from fire suppression activities inside the Reactor Building to either the Utility Collector Tank (Normal Lineup), the	Pre-Fire Plans specify the utilization of portable smoke ejectors to route smoke to Northeast section of Reactor Building at 237' elevation for removal by the normal building exhaust in CU Sludge and Drain Tank Rooms and Equipment and	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Personnel Lock Room. Approximately 5200 CFM of normal building exhaust available in this area.	release paths.	
			PFP-RX198-03	Southwest Reactor Building - 198'/218' Elevation	N	N/A	N/A	N/A	N/A
			PFP-RX237-02	West Reactor Building - 237' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building exhaust in the Shutdown Cooling Pumps room and the Railroad Lock at 261' elevation will remove smoke from this area via opening in the ceiling. Approximately 3200 CFM of exhaust capacity is available for removing smoke in this area.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA2	R2B	General Floor Area, West on elevation 261'	PFP-RX261-02	West Reactor Building - 261' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Reactor Building in this area include the air lock at column line K-11 and air lock at the Track Bay. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	Pre-Fire Plans specify normal building ventilation can be used to exhaust smoke from this area. Approximately 3200 CFM of exhaust capability is available with exhaust registers in the track bay and Shutdown Cooling Pumps and Heat Exchanger Room. Portable smoke ejectors and flexible duct may also be used to route smoke directly to the outside via the Track Bay airlock. This is a breach of secondary containment. Rad. Protection and Operations approval are required prior to releases outside the building. An additional flow path from the Reactor Building in this area includes the air lock at column line K-11. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA2	R2C	Shutdown Cooling Room on elevation 261'	PFP-RX261-02	West Reactor Building - 261' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Reactor Building in this area include the air lock at column line K-11 and air lock at the Track Bay. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	Pre-Fire Plans specify normal building ventilation can be used to exhaust smoke from this area. Approximately 3200 CFM of exhaust capability is available with exhaust registers in the track bay and Shutdown Cooling Pumps and Heat Exchanger Room. Portable smoke ejectors and flexible duct may also be used to route smoke directly to the outside via the Track Bay airlock. This is a breach of secondary containment which requires approval by Rad. Protection and Operations prior to making releases outside the building. An additional flow path from the Reactor Building in this area includes the air lock at column line K-11. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
							contaminated.		
FA2	R2D	Reactor Building Track Bay on elevation 261'	PFP-RX261-02	West Reactor Building - 261' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Reactor Building in this area include the air lock at column line K-11 and air lock at the Track Bay. Responding fire	Pre-Fire Plans specify normal building ventilation can be used to exhaust smoke from this area. Approximately 3200 CFM of exhaust capability is available with exhaust registers in the track bay and Shutdown Cooling Pumps and Heat Exchanger Room. Portable smoke ejectors and flexible duct may also be used to route smoke directly to the outside via the Track Bay airlock. This is a breach of secondary containment which requires approval by Rad. Protection and Operations prior to making releases outside the building. An additional flow path from the Reactor Building in this area includes the air lock at column line K-11. Responding fire brigade	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.		
FA2	R3B	General Floor Area, West on elevation 281'	PFP-RX281-02	West Reactor Building - 281' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation exhaust on the east side of the Reactor Building can remove smoke from this area. Smoke will also rise through the stairway and hatch openings to 298' elevation where the normal building ventilation exhaust (on the east side) will assist in smoke removal. Smoke removal will be very slow and cannot be improved by the use of smoke ejectors.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA2	R4B	General Floor Area, West on elevation 298'	PFP-RX298-03	West Reactor Building - 298' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 2000 CFM of exhaust capacity is available in the ECIV room; however, smoke in this room will actuate the Halon 1301 fire suppression system and close the damper to the exhaust duct. Approximately 4800 CFM of exhaust capacity is available at the exhaust trunk near column line P-12.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.
FA2	R5B	General Floor Area, West on elevation 318'	PFP-RX318-02	West Reactor Building - 318' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 7000 CFM of exhaust capacity is available at the exhaust trunk located near column line P-12. Portable smoke ejectors and flexible duct may be	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	used to route smoke to the vicinity of this exhaust register. Smoke which rises up through the open stairwell will be removed by the exhaust ventilation near the roof truss above the refuel floor. Approximately 26,400 CFM of exhaust capacity is available in this area.		
FA2	R6A	General Floor Area on elevation 340'	PFP-RX340-01	Refuel Floor - 340' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 26,400 CFM of exhaust capacity is available near the roof truss area above 340' elevation. Flow paths from the Reactor Building in this area include the Turbine Building stairwell near column line J-12 at 333' elevation. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						Flow paths from the Reactor Building in this area include the Turbine Building stairwell near column line J-12 at 333' elevation. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.			
FA2	R6B	General Floor Area, West on elevation 340'	PFP-RX340-01	Refuel Floor - 340' Elevation	Y	Floor drain sumps, each having a capacity of 58 gallons, are located in the corners of the Reactor Building (elev. 198'). Potentially contaminated water runoff from fire suppression activities inside the Reactor Building would drain through floor drains or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 26,400 CFM of exhaust capacity is available near the roof truss area above 340' elevation. Flow paths from the Reactor Building in this area include the Turbine Building stairwell near column line J-12 at 333' elevation. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as	Training reinforces Pre-Fire Plan use and use of N1-OP-10 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						prior to processing by the liquid waste system. Flow paths from the Reactor Building in this area include the Turbine Building stairwell near column line J-12 at 333' elevation. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	they may be contaminated.		
FA4	AB1F	Foam Room	N/A	N/A	N	N/A	N/A	N/A	N/A
FA5	OG1	General Floor Area on elevation 232'	PFP-OG232-01	Offgas Building - 232' Elevation	N	N/A	N/A	N/A	N/A
FA5	OG2	General Floor Area on elevation 247'	PFP-OG247-01	Offgas Building - 247' Elevation	N	N/A	N/A	N/A	N/A
FA5	OG3	General Floor Area on elevation 261'	PFP-OG261-01	Offgas Building - 261' Elevation	Y	A floor drain sump, having a capacity of 1,600 gallons is provided in the Off-Gas Building. Potentially contaminated water runoff from fire suppression activities in the Off-Gas Building would drain through the floor drains or down stairwells to the floor drain sump. The associated sump pump automatically transfers	Pre-Fire Plans specify normal building ventilation exhaust system may be used to remove smoke from this area. Approximately 1800 CFM of exhaust capacity is available in this area. At this rate, a complete air change will take 30-60 minutes. Exhaust registers are located at the ceiling level along the East wall near column lines	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Off-Gas Building in this area include doorways into adjacent plant buildings. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	G and H. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.		
FA5	T1	Turbine Condenser/Heater Bay Area	PFP-TB250-11	Condenser Area - 243'/250' Elevations	N	N/A	N/A	N/A	N/A
			PFP-TB261-05	Condenser Area - 261' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire	Pre-Fire Plans specify normal building ventilation can remove a significant amount of smoke from this area. Approximately 33,000 CFM of exhaust capacity is available within the condenser bay and an additional 62,000 CFM of exhaust capacity is available at	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	the Turbine floor area to remove smoke that rises through openings to this area. Turbine Building roof smoke and heat vents can be opened manually or will open automatically on high temperature to provide additional ventilation capacity. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.		
			PFP-TB277-03	North Moisture Separator Area - 277' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The	Pre-Fire Plans specify normal building ventilation exhaust system will remove smoke from this area. Approximately 7000 CFM of exhaust capacity is available for the moisture separator areas and 20,000 CFM is available for the North Condenser area.	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.			
			PFP-TB277-04	South Moisture Separator Area - 277' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the	Pre-Fire Plans specify normal building ventilation exhaust system will remove smoke from this area. Approximately 7000 CFM of exhaust capacity is available for the moisture separator areas and 20,000 CFM is available for the South Condenser area.	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

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Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.			
			PFP-TB277-05	Condenser Area - 277' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation can remove a significant amount of smoke from this area. Approximately 33,000 CFM of exhaust capacity is available within the condenser bay and an additional 62,000 CFM of exhaust capacity is available at the Turbine floor area to remove smoke that rises through openings to this area. Turbine Building roof smoke and heat vents-can be opened manually or will open automatically on high temperature to provide additional ventilation capacity. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
			PFP-TB300-03	North Reheater Area - 300' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 7,000 CFM of exhaust capacity is available above the Reheater area.	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.
			PFP-TB300-04	South Reheater Area - 300' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev.	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 62,000 CFM of exhaust capacity is available at	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	the roof truss area above the Turbine Generator. Turbine Building roof smoke and heat vents can be manually opened or will open automatically on high temperature to provide additional ventilation capacity. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	monitoring potential radioactive release paths.	
			PFP-TB300-05	Turbine Area - 300' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor	Pre-Fire Plans specify smoke from this area will be removed by the normal building ventilation exhaust system in the roof truss area. Approximately 62,000 CFM or exhaust capacity is available in this area. Turbine Building roof smoke and heat vents can be opened manually or will open automatically on high temperature to provide additional	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	ventilation capacity. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.		
			PFP-TB300-08	Generator Area - 300' Elevation	N	N/A	N/A	N/A	N/A
FA5	T1A	MSIV Room and Steam Tunnel on elevation 240'	PFP-MS240-01	Main Steam Line Room - 240' Elevation	N	N/A	N/A	N/A	N/A
FA5	T3A	General Floor Area East of MSIV Room and Fire Zone T1 on elevation 261'	PFP-TB261-01	Turbine Auxiliary Extension - 261' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB261-02	Northeast Turbine Building - 261' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB261-03	East Turbine Building - 261' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB261-04	Turbine Oil Reservoir Room - 261' Elevation	N	N/A	N/A	N/A	N/A

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA5	T3B	General Floor Area West of MSIV Room; also South and West of Fire Zone T1 on elevations 237' and 261'	PFP-TB261-06	Southeast Turbine Building - 261' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB261-09	Center North Turbine Building - 261' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify smoke from this area will rise (and be drawn) through ceiling openings to upper elevations where the normal building ventilation exhaust system will remove it. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.
			PFP-TB261-10	Northwest Turbine Building - 261' Elevation	N	N/A	N/A	N/A	N/A

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
			PFP-TB261-11	Center West Turbine Building - 261' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB261-12	Southwest Turbine Building - 261 ' Elevation	N	N/A	N/A	N/A	N/A
FA5	T4A	General Floor Area East of Fire Zone T1 on elevation 277'	PFP-TB277-01	Northeast Turbine Building - 277' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify smoke from a fire in this area will rise (and be drawn) to upper elevations where the normal building ventilation exhaust will remove the smoke Approximately 62,000 CFM of exhaust capacity is available from the Turbine floor area. Turbine Building roof smoke and heat vents can also be opened manually or will open automatically on high temperature to provide additional ventilation capacity. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

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Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls: Liquid	Engineering Controls: Gaseous	Training Review Results	Conclusions
			PFP-TB277-02	East Turbine Building - 277' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB289-01	Turbine Aux. Extension Building - 289' Elevation	N	N/A	N/A	N/A	N/A
FA5	T4B	General Floor Area West of Fire Zone T1 on elevation 277'	PFP-TB277-06	South Turbine Building - 277' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB277-10	Southwest Turbine Building - 277' Elevation	N	N/A	N/A	N/A	N/A
FA5	T4C	Hydrogen Seal Oil Unit Room on elevation 277'	PFP-TB277-09	Hydrogen Seal Oil Area - 277' Elevation	N	N/A	N/A	N/A	N/A
FA5	T4D	Battery Room 14 on elevation 277'	PFP-TB277-01	Northeast Turbine Building - 277' Elevation	N	N/A	N/A	N/A	N/A
FA5	T5A	General Floor Area, North on elevation 291'	PFP-TB291-01	North Turbine Building - 291' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down	Pre-Fire Plans specify smoke from this area will rise (and be drawn) to the Turbine Floor where the normal building ventilation exhaust system will remove it. Approximately 62,000 CFM of exhaust capacity is available for removing smoke from the Turbine Floor area. Turbine Building roof smoke and heat vents can be opened manually or	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	automatically on high temperature to provide additional ventilation capacity. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.		
			PFP-TB291-02	Northwest Turbine Building - 291' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal	Pre-Fire Plans specify smoke from this area will rise (and be drawn) to the Turbine Floor where the normal building ventilation exhaust system will remove it. Approximately 62,000 CFM of exhaust capacity is available for removing smoke from the Turbine Floor area. Turbine Building roof smoke and heat vents can be opened manually or automatically on high temperature to provide additional ventilation capacity. Responding fire brigade and radiation protection personnel shall take measures to contain	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

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Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	and monitor smoke and gaseous fire suppression agents as they may be contaminated.		
FA5	T6A	General Floor Area, North on elevation 305'	PFP-TB305-01	Center North Turbine Building - 305' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB305-02	Northwest Turbine Building - 305' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB305-03	Lube Oil Area - 305' Elevation	N	N/A	N/A	N/A	N/A
FA5	T6B	Turbine Laydown Area, East on elevation 300'	PFP-TB300-01	Northeast Turbine Building - 300' Elevation	N	N/A	N/A	N/A	N/A
FA5	T6C	General Floor Area, South on elevation 300'	PFP-TB300-02	Southeast Turbine Building - 300' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. Approximately 62,000 CFM of exhaust capacity is available at the roof truss area above the Turbine Generator. Turbine Building roof smoke and heat vents can be manually opened or will open automatically on high temperature to provide additional ventilation capacity.	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.		
			PFP-TB300-06	Storage Area - South Turbine - 300' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste	Pre-Fire Plans specify normal building ventilation exhaust at the roof truss area above the Turbine will remove smoke from this area. Approximately 62,000 CFM of exhaust capacity is available for smoke removal in this area. Turbine Building roof smoke and heat vents can be manually opened or will open automatically on high temperature to provide additional ventilation capacity. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

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Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	contaminated.		
FA5	T6D	Mechanical Storage Area on elevation 320'	PFP-TB300-07	Mechanical Storage Area - 300' Elevation	Y	Two floor drain sumps with a 1,170 gallon capacity and 6 floor drain sumps with a 980 gallon capacity are located in the lower elevations of the Turbine Building (elev. 243'). Potentially contaminated water runoff from manual/automatic fire suppression activities in the Turbine Building would drain through floor drains, floor grating, or down stairwells to these floor drain sumps. The associated sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify the use of portable smoke ejectors and flexible duct to route smoke from this room into the Turbine Generator area where the normal building ventilation exhaust system will remove it. Approximately 62,000 CFM of exhaust capacity is available in this area. Turbine Building roof smoke and heat vents can be manually opened or will open automatically on high temperature to provide additional ventilation capacity. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of N1-OP-26 and Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

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Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA5	T7A	General Floor Area, South on elevation 320'	PFP-TB320-01	Southwest Turbine Building - 320' Elevation	N	N/A	N/A	N/A	N/A
FA5	T8A	General Floor Area, North; General Floor Area, North; General Floor Area, East on elevation 333', 351', and 369'	PFP-TB369-01	Storage Tanks Area - 369' Elevation	N	N/A	N/A	N/A	N/A
FA5	T8B	General Floor Area, West on elevation 369'	PFP-TB369-01	Storage Tanks Area - 369' Elevation	N	N/A	N/A	N/A	N/A
FA6	T2A	General Floor Area, North on elevation 250'	PFP-TB250-03	Turbine Auxiliary Extension - 250' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB250-04	Northeast Turbine Building - 250' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB250-07	Old Pipe Storage Area - 250' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB250-08	East Feedwater Heater Bay - All Elevations	N	N/A	N/A	N/A	N/A
			PFP-TB250-09	Center Feedwater Heater Bay - All Elevations	N	N/A	N/A	N/A	N/A
			PFP-TB250-10	West Feedwater Heater Bay - All Elevations	N	N/A	N/A	N/A	N/A

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Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA7	T2B	General Floor Area, West on elevation 250'	PFP-TB250-05	Turbine Auxiliary Equipment Area - 250' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB250-06	Northwest Turbine Building - 250' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB250-12	Southeast Turbine Building - 250' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB250-13	Center South Turbine Building - 250' Elevation	N	N/A	N/A	N/A	N/A
			PFP-TB250-14	Southwest Turbine Building - 250' Elevation	N	N/A	N/A	N/A	N/A
FA7	T2E	UPS Battery Room on elevation 250'	PFP-TB250-14	Southwest Turbine Building - 250' Elevation	N	N/A	N/A	N/A	N/A
FA9	T2C	Offgas Tunnel on elevation 250'	PFP-TB250-02	Offgas Pipe Tunnel - 250' Elevation	N	N/A	N/A	N/A	N/A
FA9	T2D	General Floor Area, East on elevation 250'	PFP-TB250-01	Turbine Auxiliary Room East - 250' Elevation	N	N/A	N/A	N/A	N/A
FA10	C1	Cable Spreading Room	PFP-CC250-01	Control Complex Cable Room - 250' Elevation	N	N/A	N/A	N/A	N/A

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Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA11	C2	Auxiliary Control Room, Computer Room on elevation 261'	PFP-CC261-01	Auxiliary Control Room - 261' Elevation	N	N/A	N/A	N/A	N/A
FA11	C3	Control Room on elevation 277'	PFP-CC277-01	Control Room - 277' Elevation	N	N/A	N/A	N/A	N/A
FA12	AB1A	Record Storage Vault	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB1B	SAS Equip. Room.	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB1C	CPU Equip. Area.	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB1D	Administration Building - Elevation 250'	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB1E	Administration Building - Elevation 261'	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB2A	Access Passageway	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB2B	Tech Support Center/Library.	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB2C	Rad Protection	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB2D	Warehouse	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB3A	Administration Building Addition Warehouse	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB3B	Administration Building Addition Warehouse Store Room	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB3C	Administration Building Addition Warehouse Room	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB3D	Maintenance Shop	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB3E	Telephone Room	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB4A	Administration Building Gen Office	N/A	N/A	N	N/A	N/A	N/A	N/A

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Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
		Area							
FA12	AB4B	File Room and Offices	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB4C	Offices	N/A	N/A	N	N/A	N/A	N/A	N/A
FA12	AB4D	Administration Building Gen Office Area	N/A	N/A	N	N/A	N/A	N/A	N/A
FA13	S1	Circ. Water Pump Level; Cable Tunnel; General Floor Area; Building Entrance Level, elevations 233', 250' and 256'	N/A	Screen and Pump House - 233' Elevation	N	N/A	N/A	N/A	N/A
			PFP-SH261-01	Screen and Pump House - 261' Elevation	N	N/A	N/A	N/A	N/A
FA14	S2	Diesel Fire Pump Room, elevation 256'	PFP-SH261-01	Screen and Pump House - 261' Elevation	N	N/A	N/A	N/A	N/A
FA15	RS1A	Drum Waste Storage Vaults	PFP-RS244-01	RSSB - 244' Elevation	Y	The Radwaste Solidification & Storage Building (RSSB) is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains or down stairwells to these floor drain sumps.	Pre-Fire Plans specify smoke/Halon may be removed from the electrical equipment room utilizing the RSSB Fresh Air Supply System after opening dampers closed by the fire or the operation of the Halon system. The Recirculating Atmosphere Cleanup System can be used for smoke removal in other areas until the filters become clogged.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.			
FA15	RS1B	Electrical Equipment Room	PFP-RS244-01	RSSB - 244' Elevation	Y	The RSSB is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains or down stairwells to these floor drain sumps. The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste	Pre-Fire Plans specify smoke/Halon may be removed from the electrical equipment room utilizing the RSSB Fresh Air Supply System after opening dampers closed by the fire or the operation of the Halon system. The Recirculating Atmosphere Cleanup System can be used for smoke removal in other areas until the filters become clogged.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						Neutralizer Tank, or the *Floor Drain Collector* Tank where it is stored prior to processing by the liquid waste system.			
FA15	RS1C	Feed Equipment Area	PFP-RS244-01	RSSB - 244' Elevation	Y	The RSSB is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains or down stairwells to these floor drain sumps. The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify smoke/Halon may be removed from the electrical equipment room utilizing the RSSB Fresh Air Supply System after opening dampers closed by the fire or the operation of the Halon system. The Recirculating Atmosphere Cleanup System can be used for smoke removal in other areas until the filters become clogged .	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA15	RS2A	Truck Loading Area	PFP-RS261-01	RSSB - 261' Elevation	Y	The RSSB is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains or down stairwells to these floor drain sumps. The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the RSSB in this area include the Truck Bay and personnel doors. Responding fire brigade and radiation protection personnel shall take measures to contain	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke or Halon gas from these areas. Approximately 2000 CFM of exhaust capacity is available from the RSSB Control Room (provided dampers have not isolated). Approximately 4000 CFM of exhaust capacity is available from the corridor area and approximately 5000 CFM of exhaust capacity is available from each truck bay area. Flow paths from the RSSB in this area include the Truck Bay and personnel doors. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						and monitor fire suppression products, such as water runoff as it may be contaminated.			
FA15	RS2B	Truck Loading Area	PFP-RS261-01	RSSB - 261' Elevation	Y	The RSSB is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains or down stairwells to these floor drain sumps. The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the RSSB in this area	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke or Halon gas from these areas. Approximately 2000 CFM of exhaust capacity is available from the RSSB Control Room (provided dampers have not isolated). Approximately 4000 CFM of exhaust capacity is available from the corridor area and approximately 5000 CFM of exhaust capacity is available from each truck bay area. Flow paths from the RSSB in this area include the Truck Bay and personnel doors. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and gaseous fire suppression agents as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						include the Truck Bay and personnel doors. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.			
FA15	RS2C	North of the RSSB Control Room Including the Storage Area	PFP-RS261-01	RSSB - 261' Elevation	Y	The RSSB is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains or down stairwells to these floor drain sumps. The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke or Halon gas from these areas. Approximately 2000 CFM of exhaust capacity is available from the RSSB Control Room (provided dampers have not isolated). Approximately 4000 CFM of exhaust capacity is available from the corridor area and approximately 5000 CFM of exhaust capacity is available from each truck bay area. Flow paths from the RSSB in this area include the Truck Bay and personnel doors. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor smoke and	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						prior to processing by the liquid waste system. Flow paths from the RSSB in this area include the Truck Bay and personnel doors. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	gaseous fire suppression agents as they may be contaminated.		
FA15	RS2D	RSSB Control Room	PFP-RS261-01	RSSB - 261' Elevation	Y	The RSSB is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains or down stairwells to these floor drain sumps. The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke or Halon gas from this area. Approximately 2000 CFM of exhaust capacity is available from the RSSB Control Room (provided dampers have not isolated). Approximately 4000 CFM of exhaust capacity is available from the corridor area and approximately 5000 CFM of exhaust capacity is available from each truck bay area.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	*Gaseous*		
						Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.			
FA15	RS2E	Area Above Feed Equipment Room	PFP-RS261-01	RSSB - 261' Elevation	Y	The RSSB is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains or down stairwells to these floor drain sumps. The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke or Halon gas from these areas. Approximately 2000 CFM of exhaust capacity is available from the RSSB Control Room (provided dampers have not isolated). Approximately 4000 CFM of exhaust capacity is available from the corridor area and approximately 5000 CFM of exhaust capacity is available from each truck bay area.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA15	RS3A	HVAC Fan Room, Chiller Room, and the HVAC Equipment Room	PFP-RS281-01	HVAC Area RSSB - 281' Elevation	Y	The RSSB is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains or down stairwells to these floor drain sumps. The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building Recirculating Atmosphere Cleanup System can be used to remove smoke from this area until the filters become clogged. Approximately 2820 CFM of recirculated air can be moved through this system.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.
FA15	RS4A	Vent Exhaust System Room	PFP-RS281-01	HVAC Area RSSB - 281' Elevation	Y	The RSSB is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor	Pre-Fire Plans specify normal building Recirculating Atmosphere Cleanup System can be used to remove smoke from this area until the filters become clogged.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains or down stairwells to these floor drain sumps. The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Approximately 2820 CFM of recirculated air can be moved through this system.	potential radioactive release paths.	
FA15	RS5B	Recirculating Atmosphere Cleanup System Room	PFP-RS281-01	HVAC Area RSSB - 281' Elevation	Y	The RSSB is equipped with floor drains throughout the building. There are 2 floor drain sumps, each having a capacity of 700 gallons that are designed to contain normal floor drain leakage. Potentially contaminated water runoff from manual/automatic fire suppression activities in the RSSB would drain through the floor drains	Pre-Fire Plans specify normal building Recirculating Atmosphere Cleanup System can be used to remove smoke from this area until the filters become clogged. Approximately 2820 CFM of recirculated air can be moved through this system.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						or down stairwells to these floor drain sumps. The floor drain sump pumps will automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.			
FA15	WD1	Waste Disposal Building	PFP-WD229-01	Waste Disposal Building - 229' Elevation	Y	Two floor drain sumps have a capacity of 1,000 gallons and a third has a capacity of 2,000 gallons. Potentially contaminated water runoff from fire suppression activities inside the Waste Disposal Building would drain through floor drains or down stairwells to these floor drain sumps. The sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored	Pre-Fire Plans specify the following: For West part (West of column line 19) - Direct smoke to 236' and 247' elevations where normal building exhaust will remove it. Exhaust registers are located in the drum storage aisles on 225' elevation. For East part (East of column line 19) - Normal building exhaust will remove smoke from this elevation and from 247' and 261' elevations above if the smoke rises through openings. Exhaust registers are located near column lines K-19.3, Kg-19.3, and L-19.3 on 229'	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						prior to processing by the liquid waste system.	*elevation.*		
FA15	WD2	Waste Disposal Building	PFP-WD247-01	Waste Disposal Building - 236'/247' Elevation	Y	Two floor drain sumps have a capacity of 1,000 gallons and a third has a capacity of 2,000 gallons. Potentially contaminated water runoff from fire suppression activities inside the Waste Disposal Building would drain through floor drains or down stairwells to these floor drain sumps. The sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building exhaust at 247' elevation will remove smoke from this area. Smoke rising through openings in the ceiling will be removed by normal building exhaust at 261' elevation. Exhaust registers are located near column lines N_B,-17A, K-19.3 and in all of the tank rooms.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.
FA15	WD3A	Waste Sample Tank Room	PFP-WD261-01	Waste Disposal Building - 261' Elevation	Y	Two floor drain sumps have a capacity of 1,000 gallons and a third has a capacity of 2,000 gallons. Potentially contaminated water	Pre-Fire Plans specify normal building ventilation exhaust system may be used to remove smoke from this area. Approximately 4000 CFM of exhaust	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls: Liquid	Engineering Controls: Gaseous	Training Review Results	Conclusions
						runoff from fire suppression activities inside the Waste Disposal Building would drain through floor drains or down stairwells to these floor drain sumps. The sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Radwaste Building in this area include the Truck Load Platform door at column line Pc-19. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	capacity is available from this area. Exhaust registers are located near column lines K-20, Nc-16, and P-17A. Flow paths from the Waste Disposal Building in this area include the Truck Load Platform door at column line Pc-19. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor gaseous fire suppression agents and smoke as they may be contaminated.	monitoring potential radioactive release paths.	
FA15	WD3B	Waste Disposal Control Room	PFP-WD261-01	Waste Disposal Building - 261' Elevation	Y	Two floor drain sumps have a capacity of 1,000 gallons and a third has a capacity of 2,000 gallons. Potentially contaminated water	Pre-Fire Plans specify normal building ventilation exhaust system may be used to remove smoke from this area. Approximately 4000 CFM of exhaust	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						runoff from fire suppression activities inside the Waste Disposal Building would drain through floor drains or down stairwells to these floor drain sumps. The sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Radwaste Building in this area include the Truck Load Platform door at column line Pc-19. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	capacity is available from this area. Exhaust registers are located near column lines K-20, Nc-16, and P-17A. Flow paths from the Waste Disposal Building in this area include the Truck Load Platform door at column line Pc-19. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor gaseous fire suppression agents and smoke as they may be contaminated.	monitoring potential radioactive release paths.	
FA15	WD3C	Bailer Room	PFP-WD261-01	Waste Disposal Building - 261' Elevation	Y	Two floor drain sumps have a capacity of 1,000 gallons and a third has a capacity of 2,000 gallons. Potentially contaminated water	Pre-Fire Plans specify normal building ventilation exhaust system may be used to remove smoke from this area. Approximately 4000 CFM of exhaust	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						runoff from fire suppression activities inside the Waste Disposal Building would drain through floor drains or down stairwells to these floor drain sumps. The sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Waste Disposal Building in this area include the Truck Load Platform door at column line Pc-19. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	capacity is available from this area. Exhaust registers are located near column lines K-20, Nc-16, and P-17A. Flow paths from the Waste Disposal Building in this area include the Truck Load Platform door at column line Pc-19. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor gaseous fire suppression agents and smoke as they may be contaminated.	monitoring potential radioactive release paths.	
FA15	WD3D	DOW Area	PFP-WD261-01	Waste Disposal Building - 261' Elevation	Y	Two floor drain sumps have a capacity of 1,000 gallons and a third has a capacity of 2,000 gallons. Potentially	Pre-Fire Plans specify normal building ventilation exhaust system may be used to remove smoke from this area. Approximately	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						contaminated water runoff from fire suppression activities inside the Waste Disposal Building would drain through floor drains or down stairwells to these floor drain sumps. The sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Waste Disposal Building in this area include the Truck Load Platform door at column line Pc-19. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as it may be contaminated.	4000 CFM of exhaust capacity is available from this area. Exhaust registers are located near column lines K-20, Nc-16, and P-17A. Flow paths from the Waste Disposal Building in this area include the Truck Load Platform door at column line Pc-19. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor gaseous fire suppression agents and smoke as they may be contaminated.	FIRMEMC07 for monitoring potential radioactive release paths.	

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA15	WD3E	Truck Load Platform	PFP-WD261-01	Waste Disposal Building - 261' Elevation	Y	Two floor drain sumps have a capacity of 1,000 gallons and a third has a capacity of 2,000 gallons. Potentially contaminated water runoff from fire suppression activities inside the Waste Disposal Building would drain through floor drains or down stairwells to these floor drain sumps. The sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system. Flow paths from the Waste Disposal Building in this area include the Truck Load Platform door at column line Pc-19. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor fire suppression products, such as water runoff as	Pre-Fire Plans specify normal building ventilation exhaust system may be used to remove smoke from this area. Approximately 4000 CFM of exhaust capacity is available from this area. Exhaust registers are located near column lines K-20, Nc-16, and P-17A. Flow paths from the Waste Disposal Building in this area include the Truck Load Platform door at column line Pc-19. Responding fire brigade and radiation protection personnel shall take measures to contain and monitor gaseous fire suppression agents and smoke as they may be contaminated.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
						it may be contaminated.			
FA15	WD4	Fan Room	PFP-WD277-01	HVAC Area Waste Disposal 277' Elevation	Y	Two floor drain sumps have a capacity of 1,000 gallons and a third has a capacity of 2,000 gallons. Potentially contaminated water runoff from fire suppression activities inside the Waste Disposal Building would drain through floor drains or down stairwells to these floor drain sumps. The sump pumps automatically transfer the potentially contaminated water to either the Utility Collector Tank (Normal Lineup), the Waste Neutralizer Tank, or the Floor Drain Collector Tank where it is stored prior to processing by the liquid waste system.	Pre-Fire Plans specify normal building ventilation exhaust system can be used to remove smoke from this area. The mezzanine is open to the Waste Disposal Building which has an exhaust register near column line N_C-16 (corridor) with 800 CFM capacity. The fan room has an exhaust register along the west wall near the ceiling.	Training reinforces Pre-Fire Plan use and use of Ventilation Training Document, S107-FIRMEMC07 for monitoring potential radioactive release paths.	This satisfies performance requirements of NFPA 805 for Radioactive Release.
FA16 A	B1A	Battery Board Room 12	PFP-TB261-07	Battery Board Room 12 - 261' Elevation	N	N/A	N/A	N/A	N/A
FA16 B	B1B	Battery Board Room 11	PFP-TB261-08	Battery Board Room 11 - 261' Elevation	N	N/A	N/A	N/A	N/A

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
FA17 A	B2A	Battery Room 11	PFP-TB277-07	East Battery Room - 277' Elevation	N	N/A	N/A	N/A	N/A
FA17 B	B2B	Battery Room 12	PFP-TB277-08	West Battery Room - 277' Elevation	N	N/A	N/A	N/A	N/A
FA18	D3	DG 102 Control Cable Enclosure	N/A	N/A	N	N/A	N/A	N/A	N/A
FA19	D1A	DG 103 Foundation Room	PFP-DG250-02	Diesel Generator 102 and 103 Foundation - 250' Elevation	N	N/A	N/A	N/A	N/A
FA19	D2A	DG103 Room	PFP-DG261-04	Diesel Generator 103 -261' Elevation	N	N/A	N/A	N/A	N/A
FA20	D1C	DG 103 Routing Area	PFP-DG250-02	Diesel Generator 102 and 103 Foundation - 250' Elevation	N	N/A	N/A	N/A	N/A
FA21	D1D	DG 102 and 103 Power Board Foundation Area	PFP-DG250-01	Diesel Generator Power Boards 102 and 103 Foundation - 250' Elevation	N	N/A	N/A	N/A	N/A
FA22	D1B	DG 102 Foundation Room	PFP-DG250-02	Diesel Generator 102 and 103 Foundation - 250' Elevation	N	N/A	N/A	N/A	N/A
FA22	D2B	DG102 Room	PFP-DG261-03	Diesel Generator 102 - 261' Elevation	N	N/A	N/A	N/A	N/A
FA23	D2C	DG 102 Power Board Room	PFP-DG261-02	D. G. Power Board 102 Room - 261'	N	N/A	N/A	N/A	N/A

NEI 04-02 Radioactive Release Transition

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan No.	Pre-Fire Plan Title	Screened In	Engineering Controls		Training Review Results	Conclusions
						Liquid	Gaseous		
				Elevation					
FA24	D2D	DG 103 Power Board Room	PFP-DG261-01	D. G. Power Board 103 Room - 261' Elevation	N	N/A	N/A	N/A	N/A

A. 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 1, 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 1, 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 NMP1 MSOs of concern:

- Piping and Instrumentation Diagrams (P & IDs) for all safe shutdown systems and Fire PRA risk model systems
- Design Basis Documents for all safe shutdown systems and Fire PRA risk model systems
- System descriptions and training modules for all safe shutdown systems and Fire PRA risk model systems
- NMP1 UFSAR, Section X and Appendix 10B
- NMP1 safe shutdown logic diagrams
- NMP1 RIS 2004-03 independent assessment
- Electric schematics for various components
- PRA insights (NEI 00-01 Chapter 4; Appendix F)
- BWROG generic list of MSOs

The MSO expert panel for NMP1 was originally convened in April, 2008 using the guidance in NEI 00-01, Revision 1. The expert panel was reconvened in March 2010 to review, and revise as necessary, the list of the results of the previous expert panel based on the revised generic MSO scenarios documented in the NEI 00-01, Revision 2. In addition, new MSO scenarios resulting from Hope Creek and Susquehanna MSO reviews were also included.

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:

Initial Expert Panel 2008

The initial expert panel was convened originally in April 22-25, 2008, at the NMP1 site using the guidance in NEI 00-01, Revision 1. Representatives from NMP1 fire protection and Electrical/Mechanical Safe Shutdown (Appendix R) Engineering, NMP1 Operations, NMP1 PRA Engineering, NMP1 Transient Analysis Engineering, and supporting staff were in attendance. 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:

- BWR Owners Group Draft Working MSO list
- NFPA 805 FAQ 07-0038, Lessons Learned on Multiple Spurious Operations, Revision 1

Training for the initial 2008 expert panel was conducted in the form of an introductory overview. Topics discussed included:

- Purpose and scope
- PRA overview
- Overview training on the MSO issue, including:
    - Background on Fire-Induced MSOs
    - Types of circuit failures that can occur and result in spurious operations
    - Results of the fire testing (EPRI/NEI testing)
    - Role of the MSO resolution in NFPA 805 Transition.

Key points of the training 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 should not be on individual fire area locations, but rather on a system/component approach, in order to allow the analysis following the expert panel (e.g., PRA model and scenario development) to determine the vulnerability of the proposed interactions to credible fires.

A pre-job brief was conducted by a conference call in August 2008 with most of the expert panel participants to address the topics described above. The topics were reviewed again at the beginning of the onsite expert panel meeting.

The first day of the expert panel focused on a review of the BWROG 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

These reviews considered system flowpaths and addressed items such as deadheading of pumps, pump runout, and flow diversion.

By using the BWROG generic MSO list as guidance, a step-by-step discussion was held, typically by reviewing P&IDs, 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 NMP1 SSA
- Whether the scenario of concern was currently modeled in the NMP1 Internal Events PRA
- Whether procedures addressed the potential scenarios of concern
- Additional analyses or justification that may be necessary to document exclusion of a particular scenario

Consensus was achieved in the expert panel process by discussing individual scenarios, reaching a conclusion, and asking for any dissenting opinions. The findings of the expert panel were documented in a report issued in 2008.

Follow-on Expert Panel 2010:

A follow-on expert panel meeting in March 2010 utilized the same information as described above, but with an updated BWROG generic MSO list from NEI 00-01 Revision 2 and current lessons learned from the MSO process in FAQ 07-0038 and the NFPA 805 pilot plants. A brief overview training was conducted since all of the

participants were familiar with the process and had previously participated in the earlier MSO expert panel meetings. The follow-on meeting in 2010 consisted of a review of outstanding action items and a review of items that had been added or changed from the BWROG generic MSO list from NEI 00-01, Revision 2.

Consensus was achieved in the expert panel process by discussing individual scenarios, reaching a conclusion, and asking for any dissenting opinions.

Meeting of Expert Panel 2012:

The expert panel was reconvened in May 2012 to review proposed changes to the *expert panel report. The same process described above was used to perform the* review.

Consensus was achieved in the expert panel process by discussing individual scenarios, reaching a conclusion, and asking for any dissenting opinions.

Step 3 – Update the Fire PRA model and NSCA to include the MSOs of concern.

This includes:

- 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)
- Identifying 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 results of the expert panel were included in Task 7.3.1 (NUREG/CR-6850 Task 2) and Task 7.4 (NUREG/CR-6850 Task 3) within the scope of the NMP1 Fire PRA, and in Task 4.2.2, Table B-3 and Fire Area Analysis within the scope of the NMP1 NSCA. Task 7.3.1 addressed spurious operations, including multiple spurious operations identified in the post-fire safe shutdown analysis, and those that resulted from the expert panel review.

The results of the Fire PRA model update are included in NMP1 Fire PRA Notebook, "Equipment Selection," which includes the following MSO related information:

- Identification and disposition of equipment from the review of MSOs (Table D-1 of the "Equipment Selection" notebook); and
- Fire PRA equipment list, which includes MSO identified components and their associated basic events (Table G-1 of the "Equipment Selection" notebook).

The MSO combination components were also evaluated for inclusion into the NMP1 NSCA. As necessary, components were added to the NSCA Equipment List and Logics, and the appropriate circuit analysis and cable routing were performed.

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 evaluated as part of the NMP1 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 NMP1 Results are documented in:

- "Resolution of Issues Related to Fire-Induced Circuit Failures, Technical Report on Identification & Classification of the NMP1 MSO Scenarios using an Expert Panel"
- NMP1 Fire PRA Notebook, N1-ES-F001, "Equipment Selection (ES)"
- NMP1 Fire PRA Notebook, N1-CS-F001, "Cable Selection, Detailed Circuit Analysis and Route Location (CS)"
- NMP1 Fire PRA Notebook, N1-PRM-F001, "Plant Response Model"
- NMP1 Fire Area Transition – See Attachment C (NEI 04-02 Table B-3) of the Transition Report
- EIR 51-9133191, NMP1 Nuclear Safety Capability Assessment (NSCA) Report

G. Recovery Actions Transition

35 Pages Attached

In accordance with the guidance provided in NEI 04-02, FAQ 07-0030, Revision 5, and RG 1.205, the following methodology was used to determine recovery actions required for compliance (i.e., determining the population of post-transition recovery actions). The methodology consisted of the following steps:

- Step 1: 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 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.

An overview of these steps and the results of their implementation are provided below.

Step 1 - Clearly define the primary control station(s) and determine which pre-transition OMAs are taken at primary control station(s)

The first task in the process of determining the post-transition population of recovery actions was to apply the NFPA 805 definition of recovery action and the RG 1.205 definition of primary control station to determine those activities that are taken at primary control station(s).

Results of Step 1:

Based on the definition provided in RG 1.205, and the additional guidance provided in FAQ 07-0030, the following locations are considered as taking place at the primary control station(s):

1. Remote Shutdown Panel 11 is located in Fire Area 7, Fire Zone T2B, Turbine Building Elevation 250'-0".
2. Remote Shutdown Panel 12 is located in Fire Area 5, Fire Zone T4A, Turbine Building Elevation 277'-0".

The remote shutdown panels were approved by the NRC in SER entitled "Subject Modifications and Alternate Safe Shutdown Capabilities to Comply with the Requirements of Appendix R", dated March 3, 1983.

Table G-1 - "Recovery Actions and Activities Occurring at the Primary Control Station(s)" identifies the activities that occur at the primary control station(s). Activities necessary to enable the primary control station(s) are also identified in Table G-1 as primary control station(s) activities. These activities do not require the treatment of additional risk and are compliant with NFPA 805, Section 4.2.3.1.

Note that the Remote Shutdown Panels (RSPs) are primary control station(s) only for a fire in Fire Area 11 for which a MCR evacuation is credited.

Step 2 – Determine the population of recovery actions that are required to resolve VFDRs (to meet the risk or defense-in-depth criteria)

On a fire area basis, all VFDRs were identified in the NEI 04-02 Table B-3 (see Attachment C). Each VFDR not brought into compliance with the deterministic approach was evaluated using the performance-based approach of NFPA 805 Section 4.2.4. The performance-based evaluations resulted in the need for recovery actions to meet the risk acceptance criteria or maintain a sufficient level of defense-in-depth.

Results of Step 2:

The FRE report provides the determination of recovery actions required to resolve VFDRs. These recovery actions are listed in Table G-1, "Recovery Actions and Activities Occurring at the Primary Control Station(s)."

The actions contained in Table G-1 are identified on a fire area basis. Many of the same actions are repeated in different fire areas. To assist in understanding the various types of recovery actions contained in Table G-1, Table 4-1 has been created. Table 4-1 is a list of unique recovery actions only and does not include primary control station actions. It is important to note that not every component listed in the Components column of Table 4-1 is associated with every fire area listed in the Fire Areas of Concern column. The Fire Areas of Concern reflects the aggregate list of fire areas where the type of unique recovery action is credited to support shutdown. It is also important to note that item 15 in Table 4-1 is a proposed modification identified in Attachment S, specifically Table S-1. The final set of recovery actions is provided in Table G-1 - Recovery Actions and Activities Occurring at the Primary Control Station(s).

Table 4-1: Unique Operator Recovery Actions

No.	Action Description	Components	Fire Areas of Concern
1	Operator determines vital parameters at RSP	LI-36-26, LI-36-28, LI-60-22C, LI-60-23C, PI-201.2-94, PI-201.2-5, PI-36-25, PI-36-27, TI-201-50B, TI-201-51B TI-201.2-521B, TI-201.2-522B, TI-32-02B, TI-32-03B, TI-32-04B, TI-32-05B	1, 2, 5, 10
2	Operate locally to provide cooling water to Shutdown Cooling Pumps	BV-70-53	4, 5, 6, 7, 9, 10, 11, 12, 13, 14, 15,16A, 16B, 17A, 17B, 19, 20, 21
3	Operate locally to maintain cool down rate	BV-38-04, FCV-38-09, FCV-38-10, FCV-38-11, IV-38-01, IV-38-02, IV-38-13, PMP-38-152	4, 5, 6, 7, 9, 10, 11, 12, 13, 14, 15,16A, 16B, 17A, 17B, 19, 20, 21
4	Wiring repaired and component operated at Power Board	IV-38-01, IV-38-13, PMP-38-149, PMP-38-152	5, 7, 10, 11
5	Valve locally throttled to control make-up to the Emergency Condensers	VLV-60-11, VLV-60-12	4, 5, 6, 7, 9, 10, 11, 12, 13, 14, 15,16A, 16B, 17A, 17B, 18, 19, 20, 21, 22, 23, 24
6	Connect portable charger to charge Batteries	BAT-B11, BAT-B12	5, 6, 7, 9, 10, 11
7	Manually isolate to prevent inventory loss	FCV-39-15, FCV-39-16, VLV-05-31, VLV-05-32	5, 6, 9, 10, 11, 18, 22, 23, 24
8	Vent air to close valve to prevent inventory loss	IV-01-03, IV-01-04	5, 6, 7, 9, 10, 11, 18, 22, 23, 24
9	Vent air to open to establish Emergency Condensers on failure to open	IV-39-05, IV-39-06	5, 10
10	Verify tripped for load shedding	PB-BB11, PB-BB12, UPS-UPS162A , UPS-UPS162B, , UPS-UPS172A , UPS-UPS172B	5, 6, 7, 9, 10, 11
11	Shut down locally for load shedding	PMP-79.1-01, PMP-79.1-07, PMP-79.1-20, PMP-79.1-26	5, 6, 7, 9, 10, 11
12	Locally operate to establish decay heat removal through Emergency Condensers	IV-39-07R, IV-39-08R, IV-39-09R, IV-39-10R	10
13	Emergency Condenser Level Control Transfer switch to local	LCV-60-17, LCV-60-18	10
14	Operate manually to isolate Containment spray on spurious start	VLV-93-13, VLV-93-16	5, 11, 24
15	Open NEW Disconnect to load Emergency Diesel Generator to Dead bus (Currently a Damage Repair Procedure action)	Recover Emergency Diesel Generator	5, 7, 9, 10, 11
16	Scram control rods by venting the scram air header	BV-113-3091, VLV-113-230	5, 7, 10, 11

Step 3: Evaluate the Additional Risk of the Use of Recovery Actions

NFPA 805 Section 4.2.3.1 does not allow recovery actions when using the deterministic approach to meet the nuclear safety performance criteria. However, the use of recovery actions is allowed by NFPA 805 using a risk informed, performance-based, approach, provided that the additional risk presented by the recovery actions is evaluated in accordance with NFPA 805 Section 4.2.4.

Results of Step 3:

The set of recovery actions that are necessary to demonstrate the availability of a success path for the nuclear safety performance criteria (see Table G-1) were evaluated for additional risk using the process described in NEI 04-02, FAQ 07-0030, Revision 5, and RG 1.205 and compared against the guidelines of RG 1.174 and RG 1.205. None of the recovery actions were found to have an adverse impact on the Fire PRA. The additional risk of recovery actions is provided in Attachment W.

Step 4: Evaluate the Feasibility of Recovery Actions

Recovery actions were evaluated against the feasibility criteria provided in NEI 04-02, FAQ 07-0030, Revision 5, and RG 1.205. Note that since actions taken at the primary control station are not recovery actions their feasibility is evaluated in accordance with procedures for validation of off normal procedures.

Results of Step 4:

The HRA evaluated the feasibility of recovery actions modeled in the Fire PRA and used to resolve VFDRs identified in the B-3 Table. This includes recovery actions related to AC power, Emergency Diesel Generators, and long-term decay heat removal among others. Feasibility of these recovery actions were evaluated in the HRA against the criteria outlined in NEI 04-02, FAQ 07-0030 Revision 5, and RG 1.205, making extensive use of HEP quantifications.

Recovery actions that are required by the FRE but not addressed in the HRA were evaluated for feasibility using the NEI 04-02, FAQ 07-0030 Revision 5, and RG 1.205 criteria and documented in EIR 51-9156521 entitled, "Recovery Action Review for Nine Mile Point Nuclear Power Station Unit 1 Transition to NFPA 805."

Since actions taken at primary control stations are not recovery actions, no independent feasibility evaluation is required.

Results of the feasibility assessments in the HRA and in the EIR demonstrate that all credited NFPA 805 recovery actions are feasible.

Implementation items resulting from the feasibility evaluation include:

Modify, as needed, the following procedures for recovery actions being evaluated:

- N1-SOP-21.1
- N1-SOP-21.2

Operators will be trained and qualified on the revised procedures.

These items are included as implementation items in Attachment S.

Step 5: Evaluate the Reliability of Recovery Actions

The evaluation of the reliability of recovery actions depends upon its characterization.

- The reliability of recovery actions that were modeled specifically in the Fire PRA were addressed using Fire PRA methods (i.e., HRA).
- The reliability of recovery actions not modeled specifically in the Fire PRA are bounded by the treatment of additional risk associated with the applicable VFDR. In calculating the additional risk of the VFDR, the compliant case recovers the fire-induced failure(s) as if the variant condition no longer exists. The resulting delta risk between the variant and compliant condition bounds any additional risk for the recovery action even if that recovery action were modeled.

Results of Step 5:

The reliability of recovery actions that were modeled specifically in the Fire PRA were addressed using Fire PRA methods. The HRA addresses the reliability of these recovery actions, with consideration taken for various performance shaping factors, including cues and instrumentation, timing, procedures and training, complexity, workload pressure and stress, human-machine interface, environment, special equipment, specific fitness needs, as well as crew communications, staffing, and dynamics. Accordingly, the HRA also evaluates recovery actions depending on whether they correspond or not to main control room abandonment situations.

Recovery actions that are required by the FRE but not addressed in the HRA are evaluated for reliability and documented in EIR 51-9156521 entitled, "Recovery Action Review for Nine Mile Point Nuclear Power Station Unit 1 Transition to NFPA 805."

Since actions taken at primary control stations are not recovery actions, no independent reliability evaluation is required. It should however be noted that a reliability evaluation documented in the HRA was made for those actions taken at PCSs that are credited and modeled in the Fire PRA.

Results of the reliability assessments in the HRA and in EIR 51-9156521 demonstrate that all credited NFPA 805 recovery actions are reliable.

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
1	LI-36-28 TI-201-50B	REACTOR VESSEL LEVEL DRYWELL TEMPERATURE	Operator determines vital parameters from PNL-RSP 11.	VFDR-01-009 VFDR-01-011	RA
1	LI-36-26 TI-201-51B	REACTOR VESSEL LEVEL DRYWELL TEMPERATURE	Operator determines vital parameters from PNL-RSP12.	VFDR-01-009 VFDR-01-011	RA
2	LI-36-28 TI-201-50B TI-201.2-521B	REACTOR VESSEL LEVEL DRYWELL TEMPERATURE TORUS TEMPERATURE	Operator determines vital parameters from PNL-RSP11.	VFDR-02-007 VFDR-02-008 VFDR-02-009	RA
2	LI-36-26 TI-201-51B TI-201.2-522B	REACTOR VESSEL LEVEL DRYWELL TEMPERATURE TORUS TEMPERATURE	Operator determines vital parameters from PNL-RSP12.	VFDR-02-007 VFDR-02-008 VFDR-02-009	RA
4	BV-70-53	14" AIR OPERATED BLOCKING VALVE - RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-04-005	RA
4	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-04-006	RA
4	FCV-38-10	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 12 FLOW CONTROL VALVE	FCV-38-10 is operated locally to control flow through SDC HX-38-132 to regulate RCS cool down rate.	VFDR-04-007	RA
4	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-04-008	RA
4	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-04-004	RA
4	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-04-003	RA
5	BAT-B11	125 VOLT DC STATION BATTERY 11	Portable charger with a generator connected directly to Battery Board PB-BB11 to charge Battery B11.	VFDR-05-035	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
5	BAT-B12	125 VOLT DC STATION BATTERY NUMBER 12	Portable Charger with Generator connected directly to Battery Board PB-BB12 to charge Battery B12.	VFDR-05-035	RA
5	BKR-(103/1-1) R1032/581	DIESEL GEN 103 OUTPUT BREAKER 103/1-1(R1032/581) to PB-103	Operate disconnect switch locally.	VFDR-05-040 VFDR-05-043	RA
5	BV-113-3091 VLV-113-230	BLOCKING VALVE FOR AIR SUPPLY TO SCRAM AIR SYSTEM BALL VALVE - SCRAM VALVE PILOT HEADER VENT	Operator unlocks and closes BV-113-3091. Operator removes the vent pipe cap, unlocks, and opens VLV-113-230 to vent the SCRAM air header to insert the control rods.	VFDR-05-046	RA
5	BV-38-04	SHUTDOWN COOLING PUMP 12 SUCTION – BLOCKING VALVE	BV-38-04 is operated locally to align SDC pump PMP-38-152 to the RCS to initiate SDC to provide decay heat removal.	VFDR-05-044	RA
5	BV-70-53	14" AIR OPERATED BLOCKING VALVE - RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-05-020	RA
5	EG-EDG103	EMERGENCY GENERATOR – EMERGENCY DIESEL GENERATOR UNIT 103	Operate disconnect switch locally.	VFDR-05-037 VFDR-05-038	RA
5	FCV-38-10	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE – SHUTDOWN COOLING HEAT EXCHANGER 12 FLOW CONTROL VALVE	FCV-38-10 is operated locally to control flow through SDC HX-38-132 to regulate RCS cool down rate.	VFDR-05-021	RA
5	FCV-39-15	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 11 STEAM LINE DRAIN	FCV-39-15 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS return path drain line.	VFDR-05-003	RA
5	FCV-39-16	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 12 STEAM LINE DRAIN	FCV-39-16 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS return path drain line.	VFDR-05-002	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
5	IV-01-03	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM ISOLATION VALVE 3	Air vented manually at MSIV IV-01-03 to close valve to prevent inventory loss.	VFDR-05-009	RA
5	IV-01-04	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM OUT ISOLATION VALVE 4	Air vented manually at MSIV IV-01-04 to close valve to prevent inventory loss.	VFDR-05-009	RA
5	IV-38-01	MOTOR OPERATED SHUTDOWN COOLING OUTLET INSIDE ISOLATION VALVE	At PB-167, BKR-(167/D03)52, the valve wiring is repaired and the valve operated to facilitate establishing a SDC suction flow path from the RCS to *accomplish decay heat removal.*	VFDR-05-025	RA
5	IV-38-02	MOTOR OPERATED SHUTDOWN COOLING OUTLET OUTSIDE ISOLATION VALVE	IV-38-02 is operated locally via the hand wheel to establish a SDC suction flowpath from the RCS to accomplish decay heat removal.	VFDR-05-045	RA
5	IV-38-13	REACTOR SHUTDOWN COOLING RETURN ISOLATION VALVE 1	At PB-167, BKR-(167/G03)52, the valve wiring is repaired and the valve operated to facilitate establishing a SDC discharge flow path to the RCS to accomplish decay heat removal.	VFDR-05-025	RA
5	IV-39-05	EMERGENCY COOLING LOOP 11 CONDENSATE RETURN AIR OPERATED ISOLATION VALVE (GLOBE)	IV-39-05 is manually opened by venting air from the valve to establish a decay heat removal path using EC's 111 & 112.	VFDR-05-012 VFDR-05-013 VFDR-05-014	RA
5	IV-39-06	EMERGENCY COOLING LOOP 12 CONDENSATE RETURN AIR OPERATED ISOLATION VALVE (GLOBE)	IV-39-06 is manually opened by venting air from the valve to establish a decay heat removal path using EC's 121 & 122.	VFDR-05-012 VFDR-05-013 VFDR-05-014	RA
5	PB-BB11	125VDC BATTERY BOARD 11	MG 167 motor BKR-MG167(BB11/E03) verified tripped for battery load shedding to extend battery capability.	VFDR-05-028	RA
5	PB-BB12	125VDC BATTERY BOARD 12	MG 167 *motor* BKR-MG167(BB12/F03) verified tripped for battery load shedding to extend battery capability.	VFDR-05-028	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
5	PMP-79.1-01	EMERGENCY DIESEL GENERATOR 102 TURBO LUBE OIL PUMP	PMP-79.1-01 shutdown locally for battery load shedding to extend battery capability.	VFDR-05-028	RA
5	PMP-79.1-07	EMERGENCY DIESEL GENERATOR 102 CIRCULATING LUBE OIL PUMP (6 GALLONS PER MINUTE)	PMP-79.1-07 shutdown locally for battery load shedding to extend battery capability.	VFDR-05-028	RA
5	PMP-79.1-20	EMERGENCY DIESEL GENERATOR 103 TURBO LUBE OIL PUMP	PMP-79.1-20 shutdown locally for battery load shedding to extend battery capability.	VFDR-05-028	RA
5	PMP-79.1-26	EDG 103 CIRCULATING LUBE OIL PUMP (6 GPM)	PMP-79.1-26 shutdown locally for battery load shedding to extend battery capability.	VFDR-05-028	RA
5	LI-36-26 PI-36-27	REACTOR VESSEL LEVEL REACTOR VESSEL PRESSURE	Operator determines vital parameters from PNL-RSP12.	VFDR-05-016 VFDR-05-017	RA
5	UPS-UPS162A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162A switches HDS-UPS162A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-05-028	RA
5	UPS-UPS162B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162B switches HDS-UPS162B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-05-028	RA
5	UPS-UPS172A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172A switches HDS-UPS172A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-05-028	RA
5	UPS-UPS172B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172B switches HDS-UPS172B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-05-028	RA
5	VLV-05-31	LOOP 11 VENT LINE MANUAL ISOLATION VALVE DOWN STREAM OF 05-01R AND 05-11	VLV-05-31 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS steam supply vent line.	VFDR-05-004	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
5	VLV-05-32	LOOP 12 VENT LINE MANUAL ISOLATION VALVE DOWNSTREAM OF 05-04R AND 05-12	VLV-05-32 is operated locally via the hand wheel to isolate an inventory loss flow path from the EC's 121 & 122 RCS steam supply vent line.	VFDR-05-004	RA
5	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to EC's 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-05-018	RA
5	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-05-018	RA
5	VLV-93-16	12" GATE VALVE - 112 CONTAINMENT SPRAY RAW WATER PUMP DISCHARGE VALVE	VLV-93-16 closed locally via the hand wheel to isolate CTSRW flow to Containment Spray Header #12.	VFDR-05-022	RA
6	BAT-B11	125 VOLT DC STATION BATTERY 11	Portable charger with a generator connected directly to Battery Board PB-BB11 to charge Battery B11.	VFDR-06-017	RA
6	BAT-B12	125 VOLT DC STATION BATTERY 12	Portable Charger with Generator connected directly to Battery Board PB-BB12 to charge Battery B12.	VFDR-06-017	RA
6	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HX's.	VFDR-06-013	RA
6	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE – SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-06-015	RA
6	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE – SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-06-014	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
6	FCV-39-15	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 11 STEAM LINE DRAIN	FCV-39-15 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS return path drain line.	VFDR-06-001	RA
6	FCV-39-16	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 12 STEAM LINE DRAIN	FCV-39-16 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS return path drain line.	VFDR-06-002	RA
6	IV-01-03	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM ISOLATION VALVE 3	Air vented manually at MSIV IV-01-03 to close valve to prevent inventory loss.	VFDR-06-007	RA
6	IV-01-04	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM OUT ISOLATION VALVE 4	Air vented manually at MSIV IV-01-04 to close valve to prevent inventory loss.	VFDR-06-008	RA
6	IV-38-02	MOTOR OPERATED SHUTDOWN COOLING OUTLET OUTSIDE ISOLATION VALVE	IV-38-02 is operated locally via the hand wheel to establish a SDC suction flowpath from the RCS to accomplish decay heat removal.	VFDR-06-018	RA
6	PB-BB11	125VDC BATTERY BOARD 11	MG 167 motor BKR-MG167(BB11/E03) verified tripped for battery load shedding to extend battery capability.	VFDR-06-017	RA
6	PB-BB12	125VDC BATTERY BOARD 12	MG 167 motor BKR-MG167(BB12/F03) verified tripped for battery load shedding to extend battery capability.	VFDR-06-017	RA
6	PMP-79.1-01	EMERGENCY DIESEL GENERATOR 102 TURBO LUBE OIL PUMP	PMP-79.1-01 shutdown locally for battery load shedding to extend battery capability.	VFDR-06-017	RA
6	PMP-79.1-07	EMERGENCY DIESEL GENERATOR 102 CIRCULATING LUBE OIL PUMP (6 GALLONS PER MINUTE)	PMP-79.1-07 shutdown locally for battery load shedding to extend battery capability.	VFDR-06-017	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
6	PMP-79.1-20	EMERGENCY DIESEL GENERATOR 103 TURBO LUBE OIL PUMP	PMP-79.1-20 shutdown locally for battery load shedding to extend battery capability.	VFDR-06-017	RA
6	PMP-79.1-26	EDG 103 CIRCULATING LUBE OIL PUMP (6 GPM)	PMP-79.1-26 shutdown locally for battery load shedding to extend battery capability.	VFDR-06-017	RA
6	UPS-UPS162A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162A switches HDS-UPS162A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-06-017	RA
6	UPS-UPS162B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162B switches HDS-UPS162B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-06-017	RA
6	UPS-UPS172A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172A switches HDS-UPS172A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-06-017	RA
6	UPS-UPS172B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172B switches HDS-UPS172B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-06-017	RA
6	VLV-05-31	LOOP 11 VENT LINE MANUAL ISOLATION VALVE DOWN STREAM OF 05-01R AND 05-11	VLV-05-31 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS steam supply vent line.	VFDR-06-004 VFDR-06-006	RA
6	VLV-05-32	LOOP 12 VENT LINE MANUAL ISOLATION VALVE DOWNSTREAM OF 05-04R AND 05-12	VLV-05-32 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS steam supply vent line.	VFDR-06-003 VFDR-06-005	RA
6	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-06-011	RA
6	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-06-011	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
7	BAT-B11	125 VOLT DC STATION BATTERY 11	Portable charger with a generator connected directly to Battery Board PB-BB11 to charge Battery B11.	VFDR-07-012	RA
7	BAT-B12	125 VOLT DC STATION BATTERY 12	Portable Charger with Generator connected directly to Battery Board PB-BB12 to charge Battery B12.	VFDR-07-012	RA
7	BV-113-3091 VLV-113-230	BLOCKING VALVE FOR AIR SUPPLY TO SCRAM AIR SYSTEM BALL VALVE - SCRAM VALVE PILOT HEADER VENT	Operator unlocks and closes BV-113-3091. Operator removes the vent pipe cap, unlocks, and opens VLV-113-230 to vent the SCRAM air header to insert the control rods.	VFDR-07-013	RA
7	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-07-009	RA
7	FCV-38-10	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 12 FLOW CONTROL VALVE	FCV-38-10 is operated locally to control flow through SDC HX-38-132 to regulate RCS cool down rate.	VFDR-07-010	RA
7	IV-01-03	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM ISOLATION VALVE 3	Air vented manually at MSIV IV-01-03 to close valve to prevent inventory loss.	VFDR-07-001	RA
7	IV-01-04	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM OUT ISOLATION VALVE 4	Air vented manually at MSIV IV-01-04 to close valve to prevent inventory loss.	VFDR-07-002	RA
7	PB-BB11	125VDC BATTERY BOARD 11	MG 167 motor BKR-MG167(BB11/E03) verified tripped for battery load shedding to extend battery capability.	VFDR-07-012	RA
7	PB-BB12	125VDC BATTERY BOARD 12	MG 167 motor BKR-MG167(BB12/F03) verified tripped for battery load shedding to extend battery capability.	VFDR-07-012	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
7	PMP-38-152	SHUTDOWN COOLING PUMP 12 NU02B	SDC pump PMP-38-152 is operated at PB-17B, BKR-(17B/006A)52 to establish a SDC flowpath to accomplish decay heat removal.	VFDR-07-003	RA
7	PMP-79.1-01	EMERGENCY DIESEL GENERATOR 102 TURBO LUBE OIL PUMP	PMP-79.1-01 shutdown locally for battery load shedding to extend battery capability.	VFDR-07-012	RA
7	PMP-79.1-07	EMERGENCY DIESEL GENERATOR 102 CIRCULATING LUBE OIL PUMP (6 GALLONS PER MINUTE)	PMP-79.1-07 shutdown locally for battery load shedding to extend battery capability.	VFDR-07-012	RA
7	PMP-79.1-20	EMERGENCY DIESEL GENERATOR 103 TURBO LUBE OIL PUMP	PMP-79.1-20 shutdown locally for battery load shedding to extend battery capability.	VFDR-07-012	RA
7	PMP-79.1-26	EDG 103 CIRCULATING LUBE OIL PUMP (6 GPM)	PMP-79.1-26 shutdown locally for battery load shedding to extend battery capability.	VFDR-07-012	RA
7	UPS-UPS162A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162A switches HDS-UPS162A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-07-012	RA
7	UPS-UPS162B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162B switches HDS-UPS162B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-07-012	RA
7	UPS-UPS172A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172A switches HDS-UPS172A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-07-012	RA
7	UPS-UPS172B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172B switches HDS-UPS172B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-07-012	RA
7	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-07-007	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
7	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-07-007	RA
9	BAT-B11	125 VOLT DC STATION BATTERY 11	Portable charger with a generator connected directly to Battery Board PB-BB11 to charge Battery B11.	VFDR-09-018	RA
9	BAT-B12	125 VOLT DC STATION BATTERY 12	Portable Charger with Generator connected directly to Battery Board PB-BB12 to charge Battery B12.	VFDR-09-018	RA
9	BKR-(102/2-1) R1022/571	DIESEL GEN 102 OUTPUT BREAKER 2-1(R1022/571) to PB102	Operate disconnect switch locally to allow for recovery of diesel generator.	VFDR-09-019	RA
9	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-09-013	RA
9	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-09-015	RA
9	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-09-014	RA
9	FCV-39-15	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 11 STEAM LINE DRAIN	FCV-39-15 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS return path drain line.	VFDR-09-002	RA
9	FCV-39-16	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 12 STEAM LINE DRAIN	FCV-39-16 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS return path drain line.	VFDR-09-003	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
9	IV-01-04	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM OUT ISOLATION VALVE 4	Air vented manually at MSIV IV-01-04 to close valve to prevent inventory loss.	VFDR-09-008	RA
9	IV-38-02	MOTOR OPERATED SHUTDOWN COOLING OUTLET OUTSIDE ISOLATION VALVE	IV-38-02 is operated locally via the hand wheel to establish a SDC suction flowpath from the RCS to accomplish decay heat removal.	VFDR-09-021	RA
9	PB-BB11	125VDC BATTERY BOARD 11	MG 167 motor BKR-MG167(BB11/E03) verified tripped for battery load shedding to extend battery capability.	VFDR-09-018	RA
9	PB-BB12	125VDC BATTERY BOARD 12	MG 167 motor BKR-MG167(BB12/F03) verified tripped for battery load shedding to extend battery capability.	VFDR-09-018	RA
9	PMP-79.1-01	EMERGENCY DIESEL GENERATOR 102 TURBO LUBE OIL PUMP	PMP-79.1-01 shutdown locally for battery load shedding to extend battery capability.	VFDR-09-018	RA
9	PMP-79.1-07	EMERGENCY DIESEL GENERATOR 102 CIRCULATING LUBE OIL PUMP (6 GALLONS PER MINUTE)	PMP-79.1-07 shutdown locally for battery load shedding to extend battery capability.	VFDR-09-018	RA
9	PMP-79.1-20	EMERGENCY DIESEL GENERATOR 103 TURBO LUBE OIL PUMP	PMP-79.1-20 shutdown locally for battery load shedding to extend battery capability.	VFDR-09-018	RA
9	PMP-79.1-26	EDG 103 CIRCULATING LUBE OIL PUMP (6 GPM)	PMP-79.1-26 shutdown locally for battery load shedding to extend battery capability.	VFDR-09-018	RA
9	UPS-UPS162A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162A switches HDS-UPS162A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-09-018	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
9	UPS-UPS162B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162B switches HDS-UPS162B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-09-018	RA
9	UPS-UPS172A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172A switches HDS-UPS172A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-09-018	RA
9	UPS-UPS172B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172B switches HDS-UPS172B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-09-018	RA
9	VLV-05-31	LOOP 11 VENT LINE MANUAL ISOLATION VALVE DOWN STREAM OF 05-01R AND 05-11	VLV-05-31 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS steam supply vent line.	VFDR-09-005 VFDR-09-007	RA
9	VLV-05-32	LOOP 12 VENT LINE MANUAL ISOLATION VALVE DOWNSTREAM OF 05-04R AND 05-12	VLV-05-32 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS steam supply vent line.	VFDR-09-004 VFDR-09-006	RA
9	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-09-011	RA
9	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-09-011	RA
10	BAT-B11	125 VOLT DC STATION BATTERY 11	Portable charger with a generator connected directly to Battery Board PB-BB11 to charge Battery B11.	VFDR-10-014	RA
10	BAT-B12	125 VOLT DC STATION BATTERY 12	Portable Charger with Generator connected directly to Battery Board PB-BB12 to charge Battery B12.	VFDR-10-014	RA
10	BKR-(102/2-1) R1022/571	DIESEL GEN 102 OUTPUT BREAKER 2-1(R1022/571) TO PB102	Operate disconnect switch locally to allow for recovery of diesel generator.	VFDR-10-021	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
10	BV-113-3091 VLV-113-230	BLOCKING VALVE FOR AIR SUPPLY TO SCRAM AIR SYSTEM BALL VALVE - SCRAM VALVE PILOT HEADER VENT	Operator unlocks and closes BV-113-3091. Operator removes the vent pipe cap, unlocks, and opens VLV-113-230 to vent the SCRAM air header to insert the control rods.	VFDR-10-028	RA
10	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-10-019	RA
10	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-10-020	RA
10	FCV-39-15	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 11 STEAM LINE DRAIN	FCV-39-15 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS return path drain line.	VFDR-10-007	RA
10	FCV-39-16	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 12 STEAM LINE DRAIN	FCV-39-16 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS return path drain line.	VFDR-10-008	RA
10	IV-01-03	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM ISOLATION VALVE 3	Air vented manually at MSIV IV-01-03 to close valve to prevent inventory loss.	VFDR-10-009	RA
10	IV-01-04	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM OUT ISOLATION VALVE 4	Air vented manually at MSIV IV-01-04 to close valve to prevent inventory loss.	VFDR-10-010	RA
10	IV-38-01	MOTOR OPERATED SHUTDOWN COOLING OUTLET INSIDE ISOLATION VALVE	At PB 167, BKR-(167/D03)52, the valve wiring is repaired and the valve operated to facilitate establishing a SDC suction flow path from the RCS to accomplish decay heat removal.	VFDR-10-011	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
10	IV-38-02	MOTOR OPERATED SHUTDOWN COOLING OUTLET OUTSIDE ISOLATION VALVE	IV-38-02 is operated locally via the hand wheel to establish a SDC suction flowpath from the RCS to accomplish decay heat removal.	VFDR-10-027	RA
10	IV-39-05	EMERGENCY COOLING LOOP 11 CONDENSATE RETURN AIR OPERATED ISOLATION VALVE (GLOBE)	IV-39-05 is manually opened by venting air from the valve to establish a decay heat removal path using ECs 111 & 112.	VFDR-10-012	RA
10	IV-39-06	EMERGENCY COOLING LOOP 12 CONDENSATE RETURN AIR OPERATED ISOLATION VALVE (GLOBE)	IV-39-06 is manually opened by venting air from the valve to establish a decay heat removal path using ECs 121 & 122.	VFDR-10-012	RA
10	IV-39-07R	MOTOR OPERATED LOOP 11 STEAM OUTLET OUTSIDE ISOLATION VALVE 112	IV-39-07R is operated locally via the hand wheel to establish a decay heat removal flowpath using ECs 111 & 112.	VFDR-10-012	RA
10	IV-39-08R	MOTOR OPERATED LOOP 12 STEAM OUTLET OUTSIDE ISOLATION VALVE 122	IV-39-08R is operated locally via the hand wheel to establish a decay heat removal flowpath using ECs 121 & 122.	VFDR-10-012	RA
10	IV-39-09R	MOTOR OPERATED LOOP 11 STEAM OUTLET INSIDE ISOLATION VALVE 111	IV-39-09R is operated locally via the hand wheel to establish a decay heat removal flowpath using ECs 111 & 112.	VFDR-10-012	RA
10	IV-39-10R	MOTOR OPERATED LOOP 12 STEAM OUTLET INSIDE ISOLATION VALVE 121	IV-39-10R is operated locally via the hand wheel to establish a decay heat removal flowpath using ECs 121 & 122.	VFDR-10-012	RA
10	LCV-60-17	EMERGENCY CONDENSER 111 - 112 LEVEL CONTROL VALVE (LOOP 11) - AIR ACTUATED FAIL OPEN	Place EC 111/112 Level Control Transfer switch to Local and verify Auto control by observing "A" on status panel at PNL-RSP11 to override false EC high level signal to support decay heat removal via EC's 111 & 112.	VFDR-10-012	RA
10	LCV-60-18	EMERGENCY CONDENSER 121 - 122 LEVEL CONTROL VALVE (LOOP 12) - AIR ACTUATED FAIL OPEN	Place EC 121/122 Level Control Transfer switch to Local and verify Auto control by observing "A" on status panel at PNL-RSP12 to override false EC high level signal to support decay heat removal via EC's 121 & 122.	VFDR-10-012	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
10	LI-36-28 PI-36-25	REACTOR VESSEL LEVEL REACTOR VESSEL PRESSURE	Operator determines vital parameters from PNL-RSP11.	VFDR-10-013	RA
10	PB-BB11	125VDC BATTERY BOARD 11	MG 167 motor BKR-MG167(BB11/E03) verified tripped for battery load shedding to extend battery capability.	VFDR-10-014	RA
10	PB-BB12	125VDC BATTERY BOARD 12	MG 167 motor BKR-MG167(BB12/F03) verified tripped for battery load shedding to extend battery capability.	VFDR-10-014	RA
10	PMP-38-149	SHUTDOWN COOLING PUMP 11 NU02A	SDC PMP-38-149 wiring repaired and operated locally at PB-16B BKR-(16B/009A)52 to establish a SDC flowpath to accomplish decay heat removal.	VFDR-10-026	RA
10	PMP-79.1-01	EMERGENCY DIESEL GENERATOR 102 TURBO LUBE OIL PUMP	PMP-79.1-01 shutdown locally for battery load shedding to extend battery capability.	VFDR-10-014	RA
10	PMP-79.1-07	EMERGENCY DIESEL GENERATOR 102 CIRCULATING LUBE OIL PUMP (6 GALLONS PER MINUTE)	PMP-79.1-07 shutdown locally for battery load shedding to extend battery capability.	VFDR-10-014	RA
10	PMP-79.1-20	EMERGENCY DIESEL GENERATOR 103 TURBO LUBE OIL PUMP	PMP-79.1-20 shutdown locally for battery load shedding to extend battery capability.	VFDR-10-014	RA
10	PMP-79.1-26	EDG 103 CIRCULATING LUBE OIL PUMP (6 GPM)	PMP-79.1-26 shutdown locally for battery load shedding to extend battery capability.	VFDR-10-014	RA
10	UPS-UPS162A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162A switches HDS-UPS162A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-10-014	RA
10	UPS-UPS162B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162B switches HDS-UPS162B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-10-014	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
10	UPS-UPS172A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172A switches HDS-UPS172A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-10-014	RA
10	UPS-UPS172B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172B switches HDS-UPS172B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-10-014	RA
10	VLV-05-31	LOOP 11 VENT LINE MANUAL ISOLATION VALVE DOWN STREAM OF 05-01R AND 05-11	VLV-05-31 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS steam supply vent line.	VFDR-10-005 VFDR-10-006	RA
10	VLV-05-32	LOOP 12 VENT LINE MANUAL ISOLATION VALVE DOWNSTREAM OF 05-04R AND 05-12	VLV-05-32 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS steam supply vent line.	VFDR-10-003 VFDR-10-004	RA
10	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-10-017	RA
10	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-10-017	RA
11	BAT-B11	125 VOLT DC STATION BATTERY NUMBER 11	Portable charger with a generator connected directly to Battery Board PB-BB11 to charge Battery B11.	VFDR-11-020 VFDR-11-030	RA
11	BAT-B12	125 VOLT DC STATION BATTERY NUMBER 12	Portable Charger with Generator connected directly to Battery Board PB-BB12 to charge Battery B12.	VFDR-11-020 VFDR-11-030	RA
11	BV-113-3091 VLV-113-230	BLOCKING VALVE FOR AIR SUPPLY TO SCRAM AIR SYSTEM BALL VALVE - SCRAM VALVE PILOT HEADER VENT	Operator unlocks and closes BV-113-3091. Operator removes the vent pipe cap, unlocks, and opens VLV-113-230 to vent the SCRAM air header to insert the control rods.	VFDR-11-036	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
11	BV-38-04	SHUTDOWN COOLING PUMP 12 SUCTION - BLOCKING VALVE	BV-38-04 is operated locally to align SDC pump PMP-38-152 to the RCS to initiate SDC to provide decay heat removal.	VFDR-11-035	RA
11	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-11-016	RA
11	EMERGENCY COOLING ISOLATION BYPASS SWITCH	EMERGENCY COOLING ISOLATION BYPASS SWITCH	Place Emergency Cooling Isolation Bypass Switch in bypass to enable operation of IV-39-05, IV-39-07, and IV-39-09 from PNL-RSP11.	N/A	PCS
11	EMERGENCY COOLING ISOLATION BYPASS SWITCH	EMERGENCY COOLING ISOLATION BYPASS SWITCH	Place Emergency Cooling Isolation Bypass Switch in bypass to enable operation of IV-39-06, IV-39-08, and IV-39-10 from PNL-RSP12.	N/A	PCS
11	FCV-38-10	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 12 FLOW CONTROL VALVE	FCV-38-10 is operated locally to control flow through SDC HX-38-132 to regulate RCS cool down rate.	VFDR-11-035	RA
11	FCV-39-15	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 11 STEAM LINE DRAIN	FCV-39-15 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS return path drain line.	VFDR-11-009	RA
11	FCV-39-16	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 12 STEAM LINE DRAIN	FCV-39-16 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS return path drain line.	VFDR-11-010	RA
11	IV-01-03	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM ISOLATION VALVE 3	Air vented manually at MSIV IV-01-03 to close valve to prevent inventory loss.	VFDR-11-003	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
11	IV-01-04	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM OUT ISOLATION VALVE 4	Air vented manually at MSIV IV-01-04 to close valve to prevent inventory loss.	VFDR-11-004	RA
11	IV-38-01	MOTOR OPERATED SHUTDOWN COOLING OUTLET INSIDE ISOLATION VALVE	At PB 167, BKR-(167/D03)52, the valve wiring is repaired and the valve operated to facilitate establishing a SDC suction flow path from the RCS to accomplish decay heat removal.	VFDR-11-028	RA
11	IV-38-02	MOTOR OPERATED SHUTDOWN COOLING OUTLET OUTSIDE ISOLATION VALVE	IV-38-02 is operated locally via the hand wheel to establish a SDC suction flowpath from the RCS to accomplish decay heat removal.	VFDR-11-035	RA
11	IV-38-13	REACTOR SHUTDOWN COOLING RETURN ISOLATION VALVE 1	At PB-167, BKR-(167/G03)52, the valve wiring is repaired and the valve operated to facilitate establishing a SDC discharge flow path to the RCS to accomplish decay heat removal.	VFDR-11-029	RA
11	IV-39-05	EMERGENCY COOLING LOOP 11 CONDENSATE RETURN AIR OPERATED ISOLATION VALVE (GLOBE)	IV-39-05 operated from PNL-RSP11 to establish a decay heat removal path through ECs 111 & 112.	N/A	PCS
11	IV-39-06	EMERGENCY COOLING LOOP 12 CONDENSATE RETURN AIR OPERATED ISOLATION VALVE (GLOBE)	IV-39-06 operated from PNL-RSP12 to establish a decay heat removal path through ECs 121 & 122.	N/A	PCS
11	IV-39-07R	MOTOR OPERATED LOOP 11 STEAM OUTLET OUTSIDE ISOLATION VALVE 112	IV-39-07R operated from PNL-RSP11 to establish a decay heat removal path through ECs 111 & 112.	N/A	PCS
11	IV-39-08R	MOTOR OPERATED LOOP 12 STEAM OUTLET OUTSIDE ISOLATION VALVE 122	IV-39-08R operated from PNL-RSP12 to establish a decay heat removal path through ECs 121 & 122.	N/A	PCS
11	IV-39-09R	MOTOR OPERATED LOOP 11 STEAM OUTLET INSIDE ISOLATION VALVE 111	IV-39-09R operated from PNL-RSP11 to establish a decay heat removal path through ECs 111 & 112.	N/A	PCS

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
11	IV-39-10R	MOTOR OPERATED LOOP 12 STEAM OUTLET INSIDE ISOLATION VALVE 121	IV-39-10R operated from PNL-RSP12 to establish a decay heat removal path through ECs 121 & 122.	N/A	PCS
11	LCV-60-17	EMERGENCY CONDENSER 111 - 112 LEVEL CONTROL VALVE (LOOP 11) - AIR ACTUATED FAIL OPEN	Place EC 111/112 Level Control Transfer switch to Local and verify Auto control by observing "A" on status panel at PNL-RSP11.	N/A	PCS
11	LCV-60-18	EMERGENCY CONDENSER 121 - 122 LEVEL CONTROL VALVE (LOOP 12) - AIR ACTUATED FAIL OPEN	Place EC 121/122 Level Control Transfer switch to Local and verify Auto control by observing "A" on status panel at PNL-RSP12.	N/A	PCS
11	LI-36-26 LI-60-23C PI-201.2-94 PI-36-27 TI-201-51B TI-201.2-522B TI-32-04B TI-32-05B	REACTOR VESSEL LEVEL EMERGENCY CONDENSER 121 & 122 DRYWELL PRESSURE REACTOR VESSEL PRESSURE DRYWELL TEMPERATURE TORUS TEMPERATURE REACTOR COOLANT TEMPERATURE REACTOR COOLANT TEMPERATURE	Operator determines vital parameters from PNL-RSP12.	N/A	PCS
11	MG-MG131	MG SET 131	Place MG Set #131 switch in the TRIP position and confirm CONTROL RODS IN white light lit on PNL-RSP11.	N/A	PCS
11	MG-MG141	MG SET 141	Place MG Set #141 switch in the TRIP position and confirm CONTROL RODS IN white light lit on PNL-RSP12.	N/A	PCS
11	PB-BB11	125VDC BATTERY BOARD 11	MG 167 motor BKR-MG167(BB11/E03) verified tripped for battery load shedding to extend battery capability.	VFDR-11-020 VFDR-11-030	RA
11	PB-BB12	125VDC BATTERY BOARD 12	MG 167 motor BKR-MG167(BB12/F03) verified tripped for battery load shedding to extend battery capability.	VFDR-11-020 VFDR-11-030	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
11	PMP-38-152	SHUTDOWN COOLING PUMP 12 NU02B	SDC pump PMP-38-152 is operated at PB-17B, BKR-(17B/006A)52 to establish a SDC flowpath to accomplish decay heat removal.	VFDR-11-012	RA
11	PMP-79.1-01	EMERGENCY DIESEL GENERATOR 102 TURBO LUBE OIL PUMP	PMP-79.1-01 shutdown locally for battery load shedding to extend battery capability.	VFDR-11-020 VFDR-11-030	RA
11	PMP-79.1-07	EMERGENCY DIESEL GENERATOR 102 CIRCULATING LUBE OIL PUMP (6 GALLONS PER MINUTE)	PMP-79.1-07 shutdown locally for battery load shedding to extend battery capability.	VFDR-11-020 VFDR-11-030	RA
11	PMP-79.1-20	EMERGENCY DIESEL GENERATOR 103 TURBO LUBE OIL PUMP	PMP-79.1-20 shutdown locally for battery load shedding to extend battery capability.	VFDR-11-020 VFDR-11-030	RA
11	PMP-79.1-26	EDG 103 CIRCULATING LUBE OIL PUMP (6 GPM)	PMP-79.1-26 shutdown locally for battery load shedding to extend battery capability.	VFDR-11-020 VFDR-11-030	RA
11	UPS-UPS162A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162A switches HDS-UPS162A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-11-020 VFDR-11-030	RA
11	UPS-UPS162B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 162B switches HDS-UPS162B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-11-020 VFDR-11-030	RA
11	UPS-UPS172A	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172A switches HDS-UPS172A-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-11-020 VFDR-11-030	RA
11	UPS-UPS172B	POWER SUPPLY - UNINTERRUPTABLE POWER SUPPLY	UPS 172B switches HDS-UPS172B-OS, BS, DC, IS opened for battery load shedding to extend battery capability.	VFDR-11-020 VFDR-11-030	RA
11	VLV-05-31	LOOP 11 VENT LINE MANUAL ISOLATION VALVE DOWN STREAM OF 05-01R AND 05-11	VLV-05-31 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS steam supply vent line.	VFDR-11-005 VFDR-11-007	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
11	VLV-05-32	LOOP 12 VENT LINE MANUAL ISOLATION VALVE DOWNSTREAM OF 05-04R AND 05-12	VLV-05-32 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS steam supply vent line.	VFDR-11-006 VFDR-11-008	RA
11	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-11-015	RA
11	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-11-015	RA
11	VLV-93-13	12" GATE VALVE - 121 CONTAINMENT SPRAY RAW WATER PUMP DISCHARGE VALVE	VLV-93-13 closed locally via the hand wheel to isolate CTSRW flow to Containment Spray Header #11.	VFDR-11-018	RA
11	VLV-93-16	12" GATE VALVE - 112 CONTAINMENT SPRAY RAW WATER PUMP DISCHARGE VALVE	VLV-93-16 closed locally via the hand wheel to isolate CTSRW flow to Containment Spray Header #12.	VFDR-11-017	RA
12	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-12-005	RA
12	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-12-006	RA
12	FCV-38-10	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 12 FLOW CONTROL VALVE	FCV-38-10 is operated locally to control flow through SDC HX-38-132 to regulate RCS cool down rate.	VFDR-12-007	RA
12	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-12-008	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
12	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-12-004	RA
12	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-12-003	RA
13	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-13-008	RA
13	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-13-005	RA
13	FCV-38-10	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 12 FLOW CONTROL VALVE	FCV-38-10 is operated locally to control flow through SDC HX-38-132 to regulate RCS cool down rate.	VFDR-13-006	RA
13	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-13-007	RA
13	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-13-004	RA
13	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-13-003	RA
14	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-14-005	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
14	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-14-006	RA
14	FCV-38-10	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 12 FLOW CONTROL VALVE	FCV-38-10 is operated locally to control flow through SDC HX-38-132 to regulate RCS cool down rate.	VFDR-14-007	RA
14	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-14-008	RA
14	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-14-004	RA
14	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-14-003	RA
15	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-15-005	RA
15	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-15-006	RA
15	FCV-38-10	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 12 FLOW CONTROL VALVE	FCV-38-10 is operated locally to control flow through SDC HX-38-132 to regulate RCS cool down rate.	VFDR-15-007	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
15	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-15-008	RA
15	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to EC's 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-15-004	RA
15	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-15-003	RA
16A	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-16A-007	RA
16A	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-16A-005	RA
16A	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-16A-006	RA
16A	IV-38-02	MOTOR OPERATED SHUTDOWN COOLING OUTLET OUTSIDE ISOLATION VALVE	IV-38-02 is operated locally via the hand wheel to establish a SDC suction flowpath from the RCS to accomplish decay heat removal.	VFDR-16A-008	RA
16A	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-16A-004	RA
16A	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-16A-003	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
16B	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-16B-006	RA
16B	FCV-38-10	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 12 FLOW CONTROL VALVE	FCV-38-10 is operated locally to control flow through SDC HX-38-132 to regulate RCS cool down rate.	VFDR-16B-005	RA
16B	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-16B-004	RA
16B	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-16B-003	RA
17A	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HX's.	VFDR-17A-007	RA
17A	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-17A-005	RA
17A	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-17A-006	RA
17A	IV-38-02	MOTOR OPERATED SHUTDOWN COOLING OUTLET OUTSIDE ISOLATION VALVE	IV-38-02 is operated locally via the hand wheel to establish a SDC suction flowpath from the RCS to accomplish decay heat removal.	VFDR-17A-008	RA
17A	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-17A-004	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
17A	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-17A-003	RA
17B	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-17B-006	RA
17B	FCV-38-10	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 12 FLOW CONTROL VALVE	FCV-38-10 is operated locally to control flow through SDC HX-38-132 to regulate RCS cool down rate.	VFDR-17B-005	RA
17B	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to EC's 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-17B-004	RA
17B	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to EC's 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-17B-003	RA
18	FCV-39-15	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 11 STEAM LINE DRAIN	FCV-39-15 operated locally via the hand wheel to isolate an inventory loss flow path from the EC's 111 & 112 RCS return path drain line.	VFDR-18-005	RA
18	FCV-39-16	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 12 STEAM LINE DRAIN	FCV-39-16 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS return path drain line.	VFDR-18-004	RA
18	IV-01-04	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM OUT ISOLATION VALVE 4	Air vented manually at MSIV IV-01-04 to close valve to prevent inventory loss.	VFDR-18-002	RA
18	VLV-05-31	LOOP 11 VENT LINE MANUAL ISOLATION VALVE DOWN STREAM OF 05-01R AND 05-11	VLV-05-31 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS steam supply vent line.	VFDR-18-003	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
18	VLV-05-32	LOOP 12 VENT LINE MANUAL ISOLATION VALVE DOWNSTREAM OF 05-04R AND 05-12	VLV-05-32 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS steam supply vent line.	VFDR-18-003	RA
18	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to EC's 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-18-008	RA
18	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to EC's 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-18-008	RA
19	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-19-003	RA
19	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-19-005	RA
19	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-19-004	RA
19	IV-38-02	MOTOR OPERATED SHUTDOWN COOLING OUTLET OUTSIDE ISOLATION VALVE	IV-38-02 is operated locally via the hand wheel to establish a SDC suction flowpath from the RCS to accomplish decay heat removal.	VFDR-19-006	RA
19	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-19-001	RA
19	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-19-001	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
20	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-20-004	RA
20	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-20-006	RA
20	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-20-005	RA
20	IV-38-02	MOTOR OPERATED SHUTDOWN COOLING OUTLET OUTSIDE ISOLATION VALVE	IV-38-02 is operated locally via the hand wheel to establish a SDC suction flowpath from the RCS to accomplish decay heat removal.	VFDR-20-001	RA
20	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-20-002	RA
20	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-20-002	RA
21	BV-70-53	14" AIR OPERATED BLOCKING VALVE – RBCLC INLET VALVE TO SHUTDOWN COOLING SYSTEM	BV-70-53 is operated locally to provide cooling water to the SDC pumps and a heat sink for the SDC HXs.	VFDR-21-004	RA
21	FCV-38-09	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 11 FLOW CONTROL VALVE	FCV-38-09 is operated locally to control flow through SDC HX-38-135 to regulate RCS cool down rate.	VFDR-21-006	RA
21	FCV-38-11	8" DIAPHRAGM OPERATED FLOW CONTROL VALVE - SHUTDOWN COOLING HEAT EXCHANGER 13 FLOW CONTROL VALVE	FCV-38-11 is operated locally to control flow through SDC HX-38-129 to regulate RCS cool down rate.	VFDR-21-005	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
21	IV-38-02	MOTOR OPERATED SHUTDOWN COOLING OUTLET OUTSIDE ISOLATION VALVE	IV-38-02 is operated locally via the hand wheel to establish a SDC suction flowpath from the RCS to accomplish decay heat removal.	VFDR-21-001	RA
21	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to EC's 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-21-002	RA
21	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to EC's 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-21-002	RA
22	FCV-39-16	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 12 STEAM LINE DRAIN	FCV-39-16 operated locally via the hand wheel to isolate an inventory loss flow path from the EC's 121 & 122 RCS return path drain line.	VFDR-22-004	RA
22	IV-01-04	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM OUT ISOLATION VALVE 4	Air vented manually at MSIV IV-01-04 to close valve to prevent inventory loss.	VFDR-22-002	RA
22	VLV-05-32	LOOP 12 VENT LINE MANUAL ISOLATION VALVE DOWNSTREAM OF 05-04R AND 05-12	VLV-05-32 is operated locally via the hand wheel to isolate an inventory loss flow path from the EC's 121 & 122 RCS steam supply vent line.	VFDR-22-003	RA
22	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-22-007	RA
22	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-22-007	RA
23	FCV-39-16	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 12 STEAM LINE DRAIN	FCV-39-16 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS return path drain line.	VFDR-23-003	RA

Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
23	IV-01-04	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM OUT ISOLATION VALVE 4	Air vented manually at MSIV IV-01-04 to close valve to prevent inventory loss.	VFDR-23-002	RA
23	VLV-05-32	LOOP 12 VENT LINE MANUAL ISOLATION VALVE DOWNSTREAM OF 05-04R AND 05-12	VLV-05-32 is operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 121 & 122 RCS steam supply vent line.	VFDR-23-004	RA
23	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-23-006	RA
23	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-23-006	RA
24	FCV-39-15	EMERGENCY CONDENSER STEAM LINE DRAIN PRESSURE CONTROL VALVE - FLOW CONTROL VALVE 11 STEAM LINE DRAIN	FCV-39-15 operated locally via the hand wheel to isolate an inventory loss flow path from the ECs 111 & 112 RCS return path drain line.	VFDR-24-003	RA
24	IV-01-04	24" AIR OPERATED (ANGLE) ISOLATION VALVE WITH SOLENOID VALVES - MAIN STEAM OUT ISOLATION VALVE 4	Air vented manually at MSIV IV-01-04 to close valve to prevent inventory loss.	VFDR-24-002	RA
24	VLV-60-11	MAKEUP VALVE TO LOOP 12 EMERGENCY CONDENSERS	VLV-60-11 is locally throttled to control makeup to ECs 121 & 122 to provide a RCS heat sink for decay heat removal.	VFDR-24-004	RA
24	VLV-60-12	MAKEUP VALVE TO LOOP 11 EMERGENCY CONDENSERS	VLV-60-12 is locally throttled to control makeup to ECs 111 & 112 to provide a RCS heat sink for decay heat removal.	VFDR-24-004	RA
24	VLV-93-13	12" GATE VALVE - 121 CONTAINMENT SPRAY RAW WATER PUMP DISCHARGE VALVE	VLV-93-13 closed locally via the hand wheel to isolate CTSRW flow to Containment Spray Header #11.	VFDR-24-006	RA

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 transition plants. Final closure of the FAQs will occur when RG 1.205, which endorses the new revision of NEI 04-02, is approved by the NRC.

This table includes the approved FAQs used in development of this LAR that have not been incorporated into the current endorsed revision of NEI 04-02:

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	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	4	Non-Power Operations Clarifications	ML082070249	ML082200528
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
08-0046	0	Incipient Fire Detection Systems	ML081200120 ML093220197	ML093220426
08-0047	1	Spurious Operation Probability	ML082770662	ML082950750
08-0048	0	Fire Ignition Frequency	ML081200291 ML092180383	ML092190457
08-0049	0	Cable Fires	ML081200309 ML091470242	ML092100274
08-0050	0	Non Suppression Probability	ML081200318 ML092510044	ML092190555
08-0051	0	Hot Short Duration	ML083400188 ML100820346	ML100900052
08-0052	0	Transient Fire Growth Rate and Control Room Non-Suppression	ML081500500 ML091590505	ML092120501
08-0053	0	Kerite-FR Cable Failure Thresholds	ML082660021	ML120060267

Table H-1 - NEI 04-02 FAQs Utilized in LAR Submittal

No.	Rev.	Title	FAQ Ref.	Closure Memo
07-0054*	1	Demonstrating Compliance with Chapter 4 of NFPA 805	ML103510379	ML110140183
09-0056	2	Radioactive Release Transition	ML102810600	ML102920405
09-0057	3	New Shutdown Strategy	ML100330863	ML100960568
10-0059	5	NFPA 805 Monitoring	ML111180481	ML120750108

* Note: The FAQ submittal number was 08-0054 but the NRC closure memo for the FAQ was listed as 07-0054. 07-0054 was used to be consistent with the Closure Memo.

I. Definition of Power Block

1 Page Attached

The process used to determine which NMP1 structures constitute the power block is consistent with that described in Section 1.6.46 of NFPA 805, and Section K.2 of NEI 04-02. Specifically, only structures housing equipment required for nuclear plant operations are considered as "power block" structures.

For the purpose 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)
Reactor Building	FA1, FA2, FA3
Administration Building	FA4, FA12
Turbine Building	FA5, FA6, FA7, FA9, FA16A, FA16B, FA17A, FA17B
Control Complex	FA10, FA11
Diesel Generator Building	FA18, FA19, FA20, FA21, FA22, FA23, FA24
Screenhouse	FA13, FA14
Radwaste Storage and Solidification Building	FA15
Waste Building	FA15
Offgas Building	FA5
Yard (115kV Switchyard, 345kV Switchyard, West End of Unit 1 (includes HH2, SWG#1 & 2 and duct bank, H2 tank), and Transformer Area (includes transformers XF-101S, XF-101N, XF-TB01, XF-TB02, and XF-T10))	EXT

J. Fire Modeling V&V

4 Pages Attached

1. Fire Models

Fire modeling tools are used in the NMP1 NFPA 805 transition process in the Fire PRA only. 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 and quantified for CDF and LERF. Table J-1 includes the model identification, the technical references for the model, and the validation work available for it. The selected models are listed in the draft Regulatory Guide DG-1218 published in March 2009 as acceptable to the NRC if each model used is shown to have been appropriately applied within the range of its applicability and V&V.

Table J-1 Fire Models used in the Analysis

Fire Model	Reference	Validation (Per NFPA 805 § 2.4.1.2.3)
Heskestad's Plume Temperature Correlation	NUREG-1805 FDT^s, or EPRI-FIVERev1 Library	NUREG-1824, Vol 3 and Vol 4
Point Source Radiation Model	NUREG-1805, FDT^s, or EPRI-FIVERev1 Library	NUREG-1824, Vol 3 and Vol 4
Heskestad's Flame Height Correlation	NUREG-1805, FDT^s, or EPRI-FIVERev1 Library	NUREG-1824, Vol 3 and Vol 4
Detection activation model (Heat and smoke detection)	EPRI-FIVERev1 Library	See Note 1
CFAST/Hot Gas Layer	NIST SP 1026, SP 1041	NUREG-1824, Vol 5, Section 6.1

Note 1: The heat and smoke detection models are not included in the validation and verification study documented in NUREG-1824. However, if used within its stated capabilities, it is the prevailing model for estimating activation times. The technical description of the use of these models in the Fire PRA is available in the FSS Fire PRA notebook N1-FSS-F001.

1.1 Verification and Validation

Section 2.4.1.2.3 in NFPA 805 states that fire models "shall be verified and validated." NUREG-1824, referenced in Table J-1, documents a verification and validation (V&V) study for the fire models listed in the table specifically for commercial nuclear power plant applications. It should be noted that the extent of V&V available is limited to specific model capabilities and scenario configuration and will not cover every scenario requiring fire modeling analysis in the Fire PRA. The available V&V results as applicable to the fire models utilized in the NMP1 Fire PRA are summarized in the FSS Fire PRA notebook N1-FSS-F001.

The V&V methodology recommends that the analyst calculate the dimensionless groups for the scenario under analysis and determine if the validation results are applicable. The fire models listed earlier in this appendix are generally used in

the Fire PRA for screening purposes using conservative input parameters following the guidance of NUREG/CR-6850. The use of screening value input parameters is intended to ensure that the screening decisions are appropriate by minimizing the impact of under-predictions.

When necessary, detailed calculations are performed reducing the level of conservatism in the input parameters. For these calculations, the limitations of the available V&V are addressed by considering sufficient margin between the predicted fire conditions and the damage thresholds applicable to the fire scenarios, as will be discussed throughout this attachment.

1.2 Model Application Range

The V&V study documented in NUREG-1824 specifies a range of applicability for the validation results. This range of applicability is expressed in terms of dimensionless parameters. The range of model input parameters from the validation study are expressed in dimensionless terms so that fire modeling analysts can compare them with plant specific scenarios of different scales.

Engineering Calculations (i.e. Hand Calculations)

For the use of engineering calculations (i.e., hand calculations used for screening and/or determining near-field fire generated conditions), Appendix G of the FSS FPRA notebook, N1-FSS-F001, provides an overview of the V&V considerations in the Fire PRA. Specifically, it evaluates the dimensionless parameters for a range of typically used input values in the Fire PRA. The objective is to provide an evaluation that demonstrates the models are appropriate for the practical range of inputs used for these engineering calculations. The engineering calculations were used for the most part in determining:

- The heat release rate required for generating damage to targets located in the fire plume or exposed to flame radiation. This is the fire intensity required to generate fire conditions exceeding the generic damage criteria specified in Appendix H of NUREG/CR-6850 as applicable to the targets in NMP1,
- The temperature of the fire plume at a given location above the fire,
- The incident heat flux to targets horizontally aligned with the fire source, and
- The time to smoke or heat detection used for determining when a particular automatic fire suppression system can be credited in the fire scenario.

It should be noted that not every hand calculation has been compared with the V&V range documented in NUREG-1824. The V&V considerations described earlier in this appendix provide reasonable assurance that the models are used within the validation range. Although not explicitly compared with the available V&V ranges, the following fire modeling approach/results used in the Fire PRA suggest that potential use of fire modeling outside the validation range should not result in under-prediction of risk contribution from fire scenarios:

1. Due to the numerous applications of these hand calculations in a Fire PRA, the calculations rely on the use of conservative or screening value input parameters to overcome potential model under predictions. As a specific

example, the 98th percentile heat release rate values suggested as screening fire intensities in Appendix E of NUREG/CR-6850 are used for determining the fire generated conditions and consequently, if fire damage will be postulated in the analysis.

2. The fire scenarios are developed using the concept of "transient zones" within a fire zone. A transient zone is considered to be an "expanded" zone of influence intended to capture ALL the targets within the scope of the Fire PRA in the location of the transient zone. As such, transient zones constitute a subdivision of the Fire Zone into specific fire scenarios, capturing the targets and ignition sources associated with that location, which in all cases is larger than the limited zone of influence calculated using fire models (see Section 6.1.2 of the FSS Fire PRA notebook N1-FSS-F001).

Zone Modeling (CFAST)

The zone model CFAST was used for determining hot gas layer temperatures for selected fire zones within the scope of the Fire PRA. These calculations are documented in Appendices I through X of the FSS Fire PRA notebook N1-FSS-F001. If it was determined that hot gas layer temperatures exceeded the damage thresholds, hot gas layer scenarios were postulated and quantified for the corresponding ignition sources. The dimensionless parameters for the CFAST files were not evaluated against the available V&V criteria in NUREG-1824. It should be noted that in some calculations, particularly those associated with the reactor and turbine buildings, there are relatively complex configurations not explicitly covered by the V&V criteria in NUREG-1824. However, these calculations are necessary in order to ensure that the contribution from fire scenarios involving damaging hot gas layer conditions in the fire zones are not excluded from the fire risk model. In order to address in part the scope limitations of the verification and validation study, all hot gas layer calculations were conducted using heat release rate values consisting of the combination of fixed ignition source and intervening combustibles that bound all the potential scenarios in the fire zone. Although not explicitly compared with the available V&V ranges, the following fire modeling approach/results used in the Fire PRA suggest that potential use of fire modeling outside the validation range should not result in under-prediction of risk contribution from fire scenarios:

1. Hot gas layer scenarios (i.e., fire scenarios leading to room wide damage due to fire conditions in the entire fire zone exceeding applicable damage thresholds) have been postulated in the main control room (fire zone C3), auxiliary control room (i.e., relay room, fire zone C2), and the cable spreading room (fire zone C1). The fire modeling results have been conservatively interpreted so that the time to reach room temperatures exceeding the threshold criteria are based on a bounding combination of ignition source and intervening combustibles and the time the gas temperature reaches the damage criteria (no lag time associated with target heating has been considered).

2. Hot gas layer scenarios (i.e., fire scenarios leading to room wide damage due to fire conditions in the entire fire zone exceeding applicable damage thresholds) have been postulated in fire zones with relatively large amounts of oil such as the turbine building, the screen house and the diesel generator rooms.
3. Relatively large sections of the reactor building have been postulated as fire scenarios even if no combination of ignition source and intervening combustible has been found to be capable of affecting a full elevation or multiple elevations of the reactor building.
4. Smaller rooms where damaging hot gas layer conditions have not been postulated (e.g., emergency power board rooms D2C and D2D, battery board and battery rooms B1A, B1B, B2A, and B2B) only include single pieces of equipment (e.g., one power board, one battery bank) which have been failed in the analysis, which for practical purposes simulates room wide damage. In the specific case of the emergency power board rooms, cable trays in the room not associated with the power board have been mapped to the corresponding fire scenarios that could affect them.

The above list of locations covers all of the risk contributing fire zones within the global analysis boundary of the Fire PRA.

K. Existing Licensing Action Transition

5 Pages Attached

Licensing Action Description:

Appendix R Exemption, lack of 3-hour rated boundary walls for Battery Board Room 11/12 (Section III.G.2.a).

Required Post Transition: No

Basis:

The subject exemption was requested in the 10/01/1982 NMPC submittal, as supplemented by information contained in a letter dated 12/03/1982. Information regarding the fire protection features credited for the Battery Board Rooms (11 and 12) in Fire Zone B1A and B1B was provided in NMPC letter dated 10/01/1982. The following justification for the lack of 3-hour rated boundary walls for the Battery Board Rooms was approved by the NRC in a letter dated 03/21/1983.

- Redundant systems, equipment and associated cabling are separated by a 2-hour rated wall
- Absence of transient combustibles
- Provision of smoke detectors
- Provision of manual fire suppression capability
- Minimal fixed combustible loading (combustible loading is significantly lower than the fire resistance rating of the boundary walls)
- Combustible storage and personnel access is controlled
- Provision of automatic suppression systems in adjacent areas

This exemption is no longer required because the subject boundaries have been demonstrated adequate for the hazard in an EEEE.

Fire Area	Fire Zone	Description
FA16A	B1A	Battery Board Room 12, Turbine Building 261 ft. El.
FA16B	B1B	Battery Board Room 11, Turbine Building 261 ft. El.

Reference Document(s):

EIR 51-9077683-001, NFPA 805 Fundamental Fire Protection Program and Design Elements Transition Review
FPEE 1-12-001, NMP1 Engineering Evaluation for lack of 3-hour rated boundary walls for Battery Board Rooms

Licensing Action Description:

Appendix R Exemption, lack of 3-hour rated boundary walls for Battery Room 11/12 (Section III.G.2.a).

Required Post Transition: No

Basis:

The subject exemption was requested in the 10/01/1982 NMPC submittal, as supplemented by information contained in a letter dated 12/03/1982. Information regarding the fire protection features credited for the Battery Rooms (11 and 12) in Fire Zones B2A and B2B was provided in NMPC letter dated 10/01/1982. The following justification for the lack of 3-hour rated boundary walls for the Battery Rooms was approved by the NRC in a letter dated 03/21/1983.

- Redundant systems, equipment and associated cabling are separated by a 2-hour rated wall
- Absence of transient combustibles
- Provision of smoke detectors
- Provision of manual fire suppression capability in adjacent areas
- Light fixed combustible loading (combustible loading is significantly lower than the fire resistance rating of the boundary walls)
- Restricted access to the area

This exemption is no longer required because the subject boundaries have been demonstrated adequate for the hazard in an EEEE.

Fire Area	Fire Zone	Description
FA17A	B2A	Battery Room 12, Turbine Building 277 ft. El.
FA17B	B2B	Battery Room 11, Turbine Building 277 ft. El.

Reference Document(s):

EIR 51-9077683-001, NFPA 805 Fundamental Fire Protection Program and Design Elements Transition Review

FPEE 1-12-001, NMP1 Engineering Evaluation for lack of 3-hour rated boundary walls for Battery Board Rooms

Licensing Action Description:

Appendix R Exemption, lack of 3-hour rated boundary walls for the Control Room ceiling from the Control Room side due to unprotected structural steel members (Section III.G).

Required Post Transition: No

Basis:

The subject exemption was requested in the 10/01/1982 NMPC submittal, as supplemented by information contained in a letter dated 12/03/1982. Information regarding the fire protection features credited for separation of the Control Room from the area above was provided in NMPC letter dated 10/01/1982. The following justification for the lack of 3-hour rated boundary walls for the Control Room ceiling was approved by the NRC in a letter dated 03/21/1983.

- Provision of ionization smoke detectors in the Control Room and in each control console
- Provision of manual fire suppression capability
- The Control Room is continuously manned
- Controlled personnel access
- Prohibition of transient combustible storage
- Capability of removing smoke and byproducts from the Control Room
- Minimal fire loading
- Absence of safe shutdown systems in the fire zones above the Control Room
- The ceiling area is well above the combustibles in the Control Room
- Enclosure of the majority of combustibles in metal panels

This exemption is no longer required because the subject boundaries have been demonstrated adequate for the hazard in an EEEE.

Fire Area	Fire Zone	Description
FA11	C3	Control Room, 277 ft. El.

Reference Document(s):

EIR 51-9077683-001, NFPA 805 Fundamental Fire Protection Program and Design Elements Transition Review

FPEE 1-12-001, NMP1 Engineering Evaluation for lack of 3-hour rated boundary walls for the Control Room ceiling from the Control Room side due to unprotected structural steel members

Licensing Action Description:

Appendix R Exemption, lack of 3-hour rated boundary walls for the Fire Areas 1 and 2 in the Upper Level of the Reactor Building and adjacent area (Fire Area 5 in the Turbine Building), lack of automatic suppression, and lack of an alternate shutdown capability independent of the area (Section III.G.2).

Required Post Transition: No

Basis:

The subject exemption was requested in the 10/01/1982 NMPC submittal, as supplemented by information contained in a letter dated 12/03/1982. Information regarding the fire protection features credited for the lack of 3-hour rated boundary walls for the Upper Level of the Reactor Building and adjacent area (Fire Area 5 in the Turbine Building), lack of automatic suppression, and lack of an alternate shutdown capability independent of the area was provided in NMPC letter dated 10/01/1982. The following justification was approved by the NRC in a letter dated 03/21/1983.

- Redundant systems, equipment and associated cabling are separated by twenty (20) feet of separation without intervening combustibles
- Low and controlled transient combustible loading
- Provisions of smoke detectors
- Provision of manual fire suppression capability
- Extremely light fixed combustible loading
- Less than one (1) minute of maximum fire severity

This exemption is no longer required because the subject boundaries have been demonstrated adequate for the hazard in an EEEE.

Fire Area	Fire Zone	Description
FA1	R6A	East General Floor Area, Reactor Building 369 ft. El.
FA2	R6B	West General Floor Area, Reactor Building 369 ft. El.
FA5	T8A	East General Floor Area, Turbine Building 369 ft. El.
FA5	T8B	West General Floor Area, Turbine Building 369 ft. El.

Reference Document(s):

EIR 51-9077683-001, NFPA 805 Fundamental Fire Protection Program and Design Elements Transition Review
FPEE 1-12-001, NMP1 Engineering Evaluation for lack of 3-hour rated boundary walls for the Fire Areas 1 and 2 in the Upper Level of the Reactor Building and adjacent area (Fire Area 5 in the Turbine Building), lack of automatic suppression, and lack of an alternate shutdown capability independent of the area

Licensing Action Description:

Appendix R Exemption for lack of 3-hour rated boundary walls for the Upper Level of Turbine Building along with common boundary of column line 8 (Section III.G.2).

Required Post Transition: No

Basis:

The subject exemption was requested in the 10/01/1982 NMPC submittal, as supplemented by information contained in a letter dated 12/03/1982. Information regarding the fire protection features credited for separation of Fire Zone T8A and T8B was provided in NMPC letter dated 10/01/1982. The following justification for the lack of 3-hour rated boundary walls for the Upper Level of Turbine Building along with common boundary of column line 8 was approved by the NRC in a letter dated 03/21/1983.

- Redundant systems, equipment and associated cabling are separated by twenty (20) feet of separation without intervening combustibles
- Transient combustible loading is low and controlled
- Provisions of early warning ionization smoke detectors which alarm in the control room
- Provision of manual fire suppression capability within and outside the areas
- Extremely light fixed combustible loading

This exemption is no longer required because the subject boundaries have been demonstrated adequate for the hazard in an EEEE.

Fire Area	Fire Zone	Description
FA5	T8A	East General Floor Area, Turbine Building 369 ft. El.
FA5	T8B	West General Floor Area, Turbine Building 369 ft. El.

Reference Document(s):

EIR 51-9077683-001, NFPA 805 Fundamental Fire Protection Program and Design Elements Transition Review
FPEE 1-12-001, NMP1 Engineering Evaluation for lack of 3-hour rated boundary walls for the Upper Level of Turbine Building along with common boundary of column line 8

L. NFPA 805 Chapter 3 Requirements for Approval (10 CFR 50.48(c)(2)(vii)

2 Pages Attached

Approval Request 1

NFPA 805 Section 3.3.5.1 states:

> *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.*

There are no significant amounts of wiring above suspended or dropped ceilings in the power block, and the wiring and cabling that is installed above the suspended or dropped ceiling is in conduit or cable trays. However, some of the cable trays installed above the suspended ceilings do not comply with the requirements of this code section. Three (3) cable trays installed over the Radwaste Control Room (Fire Area 15) are not enclosed. This is considered to be a deviation from this NFPA 805 element, for which NRC approval is requested.

Basis for Request

The basis for the approval request of this deviation is:

- It is not practical to replace the cable tray installations above the suspended ceilings in the Radwaste Control Room. As discussed in the Acceptance Criteria Evaluation, this deviation presents no adverse impact to the nuclear safety and radiological release performance criteria, or to safety margin and defense-in-depth.

Acceptance Criteria Evaluation

This deviation is technically acceptable based on the following:

- Based on visual inspection and review of drawings (C19484C-001, C18998C, and C19444C-002) there are no other ignition sources in the areas above the suspended ceilings of Fire Area 15 (Fire Zone WD3B).
- Detection is installed within the suspended ceiling above the trays to detect and provide early warning of fire (Detection System D-6043).
- The cables in the three (3) trays are not relied upon for safe shutdown. Additionally, a fire in the Radwaste Control Room will not impact other areas in the plant that contain safe shutdown equipment as the walls that separate it from the Turbine Building are rated for 3 hours.
- For new or modified cabling in suspended or dropped ceiling, CNG-FES-007 includes design requirements for electrical wiring above suspended ceilings and specifically references NFPA 805 as part of the design considerations.

Nuclear Safety and Radiological Release Performance Criteria:

The presence of uncovered cable trays above the Radwaste Control Room suspended ceilings does not affect nuclear safety. Therefore there is no impact on the nuclear safety performance criteria. The location of the cable trays above suspended ceilings has no impact on the radiological release performance criteria. The radiological review was performed based on the potential location of radiological concerns and is not dependent on cable tray design or locations of suspended ceilings.

Safety Margin and Defense-in-Depth:

The impact of the three (3) unenclosed cable trays above the Radwaste Control Room suspended ceilings is minor such that the safety margin inherent in the analysis for the fire event is preserved.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) 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), and
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).

The uncovered cable trays routed above the Radwaste Control Room suspended ceilings do not impact fire protection defense-in-depth. The uncovered cable trays do not compromise administrative fire prevention controls, and do not directly result in challenging automatic fire detection functions, manual fire suppression functions, or post-fire safe shutdown capability.

The area above the suspended ceiling is free of work that would cause any additional fires and any work in that area would be controlled via hot work permits.

There are no cables related to safe shutdown equipment above the ceiling, the Radwaste Control Room is separated from the Turbine Building by rated fire barriers, and detection and suppression above the ceiling are provided; therefore, this deviation provides minimal impact to safe shutdown capability.

Conclusion:

NMPNS determined that the performance based approach utilized to evaluate a variance from the requirements of NFPA 805 Chapter 3:

- 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).

M. License Condition Changes

7 Pages Attached

Replace the current NMP1 fire protection license condition 2.D(7) with the standard license condition from Regulatory Guide 1.205, Regulatory Position 3.1, modified as shown below.

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

No other license conditions need to be revised or superseded.

NMP1 implemented the following process for determining that these are the only license conditions required to be either revised or superseded to implement the new fire protection program which meets the requirements in 10 CFR 50.48(a) and 50.48(c). A review was conducted of the NMP1 Facility Operating License DPR-63. 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 condition.

Supersede License Condition 2.D(7):

> *"2.D(7) Fire Protection*
>
> *Nine Mile Point Nuclear Station, LLC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report (Updated) for the facility and as approved in the Fire Protection Safety Evaluation Report dated July 26, 1979, and in the fire protection Exemption issued March 21, 1983, subject to the following provision:*
>
> *Nine Mile Point Nuclear Station, LLC may makes changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire."*

New License Condition:

Nine Mile Point Nuclear Station, LLC, 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's 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 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 functionality of the component, system, procedure, or physical arrangement, 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 functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- 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).

(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 the NRC safety evaluation 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 the transition to full compliance with 10 CFR 50.48(c) by {date}.

[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.

Proposed Changes to Facility Operating License – Markup

The current version of Facility Operating License page 5 has been marked up to reflect the proposed change.

c. The licensee shall update the collective occupational dose estimate weekly. If the updated estimate exceeds the 1908 person-rem estimate by more than 10%, the licensee shall provide a revised estimate, including the reasons for such changes, to the NRC within 15 days of determination.

d. Progress reports shall be provided at 90-day intervals from June 30, 1982 and due 30 days after close of the interval, with a final report within 60 days after completion of the repair. These reports will conclude:

(1) a summary of this occupational dose received to date by major task, and

(2) a comparison of estimated doses with the doses actually received.

(7) Fire Protection

Insert M-1

~~Nine Mile Point Nuclear Station, LLC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report (Updated) for the facility and as approved in the Fire Protection Safety Evaluation Report dated July 26, 1979, and in the fire protection Exemption issued March 21, 1983, subject to the following provision:~~

~~Nine Mile Point Nuclear Station, LLC may makes changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.~~

(8) Hot Process Pipe Penetrations

Hot Process Pipe Penetrations in the Emergency Condenser Steam Supply (2 each), Main Steam (2 each), Feedwater (2 each), Cleanup Suction (1 each), and Cleanup Return (1 each) piping systems have been identified as not fully in conformance with FSAR design criteria. This anomaly in design condition from the original design is approved for the duration of Cycle 8 or until March 31, 1986, whichever occurs first, subject to the following conditions:

(a) An unidentified leakage limit of a change of 1 gallon per minute in 24 hours to permit operation will be imposed by administrative control (Standing Order) at the facility for the interim period.

Renewed License No. DPR-63
Amendment No.

Insert M-1:

Nine Mile Point Nuclear Station, LLC, 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's 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 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 functionality of the

component, system, procedure, or physical arrangement, 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 functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- 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).

(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 the NRC safety evaluation 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 the transition to full compliance with 10 CFR 50.48(c) by {date}.

[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.

N. Technical Specification Changes

5 Pages Attached

A review of the NMP1 Technical Specifications (TS) and TS Bases was performed to identify any changes to be made to implement the new fire protection program which meets the requirements in 10 CFR 50.48(a) and 10 CFR 50.48(c). The review consisted of reading the TS and TS Bases and performing electronic searches. Outstanding license amendment requests that have been submitted to the NRC were also reviewed.

NMPNS determined that the following are the only NMP1 TS and TS Bases changes that are needed for adoption of the new fire protection licensing basis.

Technical Specification Changes

TS Section 6.0, *Administrative Controls*, Section 6.4, *Procedures*, states, in part, that written procedures and administrative policies shall be established, implemented and maintained covering the following activities:

c. Quality assurance for radioactive effluent and radiological environmental monitoring;

d. Fire Protection Program implementation; and

This LAR proposes to add the word "and" to the end of Item c for clarity, and delete Item d by replacing the existing wording with the word "Deleted." This change is considered adequate for adoption of the new fire protection licensing basis since the requirement for establishing, implementing and maintaining fire protection procedures is contained in 10 CFR 50.48(a) and 10 CFR 50.48(c), as specifically outlined in NFPA 805, Section 3.2.3, *Procedures*.

A markup of the existing TS page to show the proposed changes is provided in the following pages of this attachment.

Technical Specification Bases Changes

The Bases for TS Sections 3.6.13 and 4.6.13, *Remote Shutdown Panels*, states, in part, that *one channel of each Function provides the necessary capabilities consistent with* 10 CFR 50 Appendix R. To reflect adoption of the new fire protection licensing basis, the term "Appendix R" will be replaced with "10 CFR 50.48(c)."

A markup of the existing TS Bases page to show the proposed change is provided in the following pages of this attachment. The TS Bases change will be processed in accordance with the NMP1 TS Bases Control Program described in TS Section 6.5.6.

Proposed Technical Specification Changes – Markup

The current version of TS page 350 has been marked up to reflect the proposed changes to TS Section 6.4.

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for radioactive effluent and radiological environmental monitoring; and

d. ~~Fire Protection Program implementation; and~~ Insert "Deleted"

e. All programs specified in Specification 6.5.

6.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

6.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 6.6.2 and Specification 6.6.3.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

(a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

(b) A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

AMENDMENT NO. ~~142, 181~~, 350

Proposed Changes to Technical Specification Bases – Markup

The current version of TS Bases page 281 has been marked up to reflect the proposed change to the Bases for TS Sections 3.6.13 and 4.6.13.

BASES FOR 3.6.13 AND 4.6.13 REMOTE SHUTDOWN PANELS

The remote shutdown panels provide 1) manual initiation of the emergency condensers 2) manual control of the steam supply valves and 3) parameters monitoring independent of the main/auxiliary control room. Two panels are provided, each located in a separate fire area, for added redundancy. Both panels are also ~~in separate fire areas~~ from the main/auxiliary control room. One channel of each Function provides the necessary capabilities consistent with ~~10CFR50 Appendix R.~~ Therefore, only one channel of either remote shutdown panel monitoring instrument or control is required to be operable. ~~The electrical design~~ of the panels is such that no single fire can cause loss of both emergency condensers.

10 CFR 50.48(c).

Each remote shutdown panel is provided with controls for one emergency condenser loop. The emergency condensers are designed such that automatic initiation is independently assured in the event of a fire 1) in the Reactor Building (principle relay logic located in the auxiliary control room or 2) in the main/auxiliary control room or Turbine Building (redundant relay logic located in the Reactor Building). Each remote shutdown panel also has controls to operate the two motor-operated steam supply valves on its respective emergency condenser loop. A key operated bypass switch is provided to override the automatic isolation signal to these valves. Once the bypass switch is activated, the steam supply valves can be manually controlled from the remote shutdown panels. Since automatic initiation of the emergency condenser is assured, the remote shutdown panels serve as additional manual controlling stations for the emergency condensers. In addition, certain parameters are monitored at each remote shutdown panel.

The remote shutdown panels are normally de-energized, except for the monitoring instrumentation, which is normally energized. To energize the remaining functions on a remote shutdown panel, a power switch located on each panel must be activated. Once the panels are completely energized, the emergency condenser condensate return valve and steam supply valve controls can be utilized.

O. Orders and Exemptions

1 Page Attached

Exemptions

Rescind the following exemptions granted against 10 CFR 50, Appendix R dated March 21, 1983.

- An exemption from the requirements of Section III.G.2 of Appendix R for the battery board rooms (FA 16A and FA 16B), since their boundary walls do not provide the required 3-hour rated barriers.
- An exemption from the requirements of Section III.G.2 of Appendix R for the battery rooms (FA 17A and FA 17B), since their boundary walls do not provide the required 3-hour rated barriers.
- An exemption from the requirements of Section III.G of Appendix R for the control room (FA 11), since the control room ceiling does not have a 3-hour rating from the control room side due to unprotected structural steel members.
- An exemption from the requirements of Section III.G.2 of Appendix R for the wall between the reactor building and the turbine building above elevation 340' (FA 1, FA 2, and FA 5), since the wall is not a 3-hour rated barrier.
- An exemption from the requirements of Section III.G.2 of Appendix R for the fire break zone separating FA 1 and FA 2 in the reactor building upper level (elevation 340'), since the wall is not a 3-hour rated barrier.

Specific details regarding these exemptions are contained in Attachment K.

Orders

No Orders need to be superseded or revised.

NMP1 implemented the following process for making this determination:

- A review was conducted of the NMP1 docketed correspondence. The review was performed by reviewing the correspondence files and performing electronic searches of internal NMP1 records and the NRC's ADAMS document system.

A specific review was performed of the license amendment that incorporated the mitigation strategies required by Section B.5.b of Commission Order EA-02-026 (TAC No. MD4550) to ensure that any changes being made to ensure compliance with 10 CFR 50.48(c) do not invalidate existing obligations applicable to the plant. The review of this order demonstrated that changes to the fire protection program will not affect measures required by B.5.b.

The Fukushima Orders are being independently evaluated. Any plant changes will be evaluated for impact on the fire protection program in accordance with the NMPNS design change process.

P. RI-PB Alternatives to NFPA 805 per 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 NMP1.

Q. No Significant Hazards Evaluation

3 Pages Attached

Nine Mile Point Nuclear Station, LLC (NMPNS) is requesting an amendment to Renewed Facility Operating License DPR-63 for Nine Mile Point Nuclear Station, Unit 1 (NMP1). The proposed amendment would permit NMP1 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.

NMPNS has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below.

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

   Response: No.

   The purpose of the proposed amendment is to permit NMP1 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 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 FR 33536, June 16, 2004).

   Operation of NMP1 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 (UFSAR) documents the analyses of design basis accidents at NMP1. 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 UFSAR. 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.

   NFPA 805, taken as a whole, provides an acceptable alternative for satisfying General Design Criterion 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.

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, and equipment required to mitigate an accident remains capable of performing the assumed function(s). The applicable radiological dose criteria will continue to be met.

Based on the above discussion, it is concluded that the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

Operation of NMP1 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 in such a manner as to introduce new or different accident initiators. 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 requirements of NFPA 805 address only fire protection and the impacts of fire on the plant that have previously been evaluated. Thus, implementation of the proposed amendment would not create the possibility of a new or different kind of accident beyond those already analyzed in the UFSAR. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced, and there will be no adverse effect or challenges imposed on any safety-related system as a result of the proposed amendment.

Based on the above discussion, it is concluded that the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does the proposed amendment involve a significant reduction in the margin of safety?**

   Response: No.

   The purpose of the proposed amendment is to permit NMP1 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).

   The overall approach of NFPA 805 is consistent with the key principles for evaluating license basis changes, as described in Regulatory Guide 1.174, is consistent with the defense-in-depth philosophy, and maintains sufficient safety margins. Engineering analyses, which may include engineering evaluations, probabilistic safety assessments, and fire modeling calculations, have been performed to demonstrate that the performance based methods do not result in a significant reduction in the margin of safety.

   Operation of NMP1 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 UFSAR. 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.

   Based on the above discussion, it is concluded that the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above evaluations, NMPNS 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

The purpose of the proposed amendment is to permit Nine Mile Point Nuclear Station, LLC 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 FR 33536, June 16, 2004).

Nine Mile Point Nuclear Station, LLC 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. Nine Mile Point Nuclear Station, LLC has determined that the proposed amendment meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

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. Transition to NFPA 805 requirements does not impact any type or amount of effluents. 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. Plant Modifications and Items to be Completed During Implementation

13 Pages Attached

Table S-1, Plant Modifications Committed, provided below, includes a description of the modifications along with the following information:

- A problem statement,
- Risk ranking of the modification,
- An indication if the modification is currently included in the FPRA,
- Compensatory measure in place, and
- A risk-informed characterization of the modification and compensatory measure.

The following Risk Ranking Legend should be used when reviewing the table:

- High = Modification would have an appreciable impact on reducing overall fire CDF.
- Med = Modification would have a measurable impact on reducing overall fire CDF.
- Low = Modification would have either an insignificant or no impact on reducing overall fire CDF.

NMPNS will complete implementation of these modifications no later than the end of the first NMP1 refueling outage following issuance of the license amendment.

Table S-1 Plant Modifications Committed

Item	Rank	Unit	Problem Statement	Proposed Modification	*In* FPRA	Comp Measure	Risk *Informed* Characterization

Redacted

Table S-1 Plant Modifications Committed

Item	Rank	Unit	Problem Statement	Proposed Modification	*In* FPRA	*Comp* Measure	*Risk Informed* Characterization

Redacted

Table S-1 Plant Modifications Committed

Item	Rank	Unit	Problem Statement	Proposed Modification	*In* FPRA	*Comp* Measure	*Risk Informed* Characterization

Redacted

Table S-1 Plant Modifications Committed

Item	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization

Redacted

Table S-1 Plant Modifications Committed

Item	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization

Redacted

Table S-1 Plant Modifications Committed

Item	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization

Redacted

Table S-1 Plant Modifications Committed

Item	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization

Redacted

Table S-1 Plant Modifications Committed

Item	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization

Redacted

Table S-2 Implementation Items provided below includes those items (procedure changes, process updates, and training to affected plant personnel) that will be completed prior to the implementation of the new NFPA 805 fire protection program. This will occur 180 days after license amendment issuance unless that falls within a scheduled refueling outage. Then this will occur 60 days following startup from that scheduled refueling outage.

Table S-2 Implementation Items

Item	Unit	Description	LAR Section / Source
1	1	Update procedures such as NIP-OUT-01 to incorporate KSF Pinch Point Analysis.	4.3.2 Attachment D
2	1	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.	4.7.2
3	1	Review plant changes during the NFPA 805 implementation period to ensure that changes are appropriately evaluated for potential impact on the PRA model. This is done per CNG-CM-1.01-3004, "PRA Process for Internal Evaluations," and CNG-CM-1.01-1003, "Design Engineering and Configuration Control." Update, if applicable, NEP-FPP-01, "Appendix R Review" to address the NFPA 805 Change Evaluation Process. Additionally, a comprehensive update of the NFPA 805 analyses is planned as part of the NFPA 805 implementation period to reflect the current plant configurations. The update will include review of plant configuration changes along with changes that may have occurred from RAI responses, updates from industry groups for MSO configurations, new or revised FAQs, and development of modifications. This final review will ensure current plant configurations are appropriately reflected and evaluated in the NFPA 805 documentation prior to full implementation of NFPA 805.	4.7.2
4	1	Incorporate as built risk related modifications and any other additional refinements that may be needed into the Fire PRA and Internal Events Model and verify the risk results are not appreciably changed.	4.7.2 4.8.2

Table S-2 Implementation Items

Item	Unit	Description	LAR Section / Source
5	1	The existing NMP1 Fire Protection QA program will be utilized with the following changes: ▪ In addition to editorial and administrative changes (i.e. replacing references to previous NRC guidelines with those associated with the NFPA 805 transition and ensuring the features required for a performance based program under NFPA 805 are addressed), the components and systems currently considered within the scope of the NMP1 Fire Protection QA Program will be expanded to include those components and systems that are in the power block and are required by Chapter 4 of NFPA 805. This means that certain Fire Protection systems and features in some buildings not currently considered under the Fire Protection QA Program that are required by NFPA 805 Chapter 4 will now fall under the Fire Protection QA program. As such, any future modifications to these systems will be conducted under the design controls required by the Fire Protection QA program. ▪ The audit requirements will be revised to include the periodic review of the Monitoring Program.	4.7.3
6	1	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.	4.7.2
7	1	Revise the fire protection policy document to identify the appropriate AHJ for the various areas of the program	Attachment A, 3.2.2.4
8	1	Performance-based surveillance frequencies will be updated as described in Electric Power Research Institute (EPRI) Technical Report TR-1006756, "*Fire Protection Surveillance Optimization and Maintenance* Guide for Fire Protection Systems and Features".	Attachment A, 3.2.3(1)
9	1	Develop and implement the NFPA 805 monitoring program per NFPA 805 Section 2.6. The monitoring program will include a process that monitors and trends the fire protection program based on specific goals established to measure effectiveness. Program specifics are provided in Section 4.6.2.	Section 4.6.2 Attachment A, 3.2.3(3)

Table S-2 Implementation Items

Item	Unit	Description	LAR Section / Source
10	1	Incorporate the NFPA 805 monitoring program into assessment procedures such that 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.	Section 4.6.2
11	1	Implement administrative controls to ensure plastic sheeting materials used in the power block are qualified in accordance with NFPA 701 or equivalent. A specific edition / year of NFPA 701 will be cited.	Attachment A, 3.3.1.2(2)
12	1	Incorporate station smoking policy into a formal, controlled process by either revising an existing procedure or creating a new procedure.	Attachment A, 3.3.1.3.2
13	1	Implement administrative controls for use of portable electric heaters in the plant and to prohibit use of portable fuel-fired heaters in power block structures.	Attachment A, 3.3.1.3.4
14	1	Implement administrative controls to utilize, to the extent practicable, noncombustible construction as defined by NFPA 220 for walls, floors, and components of new power block buildings and changes to existing power block buildings that are required to maintain structural integrity. A specific edition / year of NFPA 220 will be cited.	Attachment A, 3.3.2
15	1	Implement administrative controls to utilize, to the extent practicable, Class A materials for new interior wall or ceiling finish, and Class I materials for new interior floor finish as defined by NFPA 101. A specific edition / year of NFPA 101 will be cited.	Attachment A, 3.3.3
16	1	Incorporate requirement for new metal roof deck construction to be designed and installed so the roofing system will not sustain a self-propagating fire on the underside when exposed to a fire inside the building. Additionally, roof coverings shall be Class A as determined by tests described in NFPA 256.	Attachment A, 3.3.6
17	1	Incorporate need for plant tours and normal housekeeping activities to inspect for lubricating oil coming in contact with hot pipes and surfaces, including insulated pipes and surfaces into appropriate plant procedures.	Attachment A, 3.3.10
18	1	Revise fire brigade training program to include radioactivity and health physics considerations in quarterly fire brigade meetings.	Attachment A, 3.4.3(a)(2)

Table S-2 Implementation Items

Item	Unit	Description	LAR Section / Source
19	1	Modify, as needed, the following procedures for recovery actions being *evaluated:* N1-SOP-21.1 N1-SOP-21.2 Operators will be trained and qualified on the revised procedures.	Attachment G
20	1	Revise applicable Appendix R documentation to reflect the transition to NFPA 805. This will include UFSAR, procedures, SD, SDBDs, etc.	NFPA 805 Transition

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

15 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."*

The NMP1 internal events PRA was the starting point for the Fire PRA (FPRA). Prior to being used for the FPRA, the PRA underwent a major update to meet the ASME Standard, RA-Sb-2005 and Regulatory Guide 1.200, Revision 1. Subsequently, the updated PRA model underwent an industry peer review in February, 2008. Table U-1 lists and summarizes the findings from this review and their status with regard to resolution and incorporation into the PRA model and documentation. The findings were related primarily to documentation. Findings that affected the PRA model had a negligible impact on the PRA results. Those findings that remain open have no impact on the FPRA model. Assumptions and uncertainties (see IE-D3 and QU-E4) were considered in making this determination and it is noted that the FPRA will consider sensitivity analyses, as appropriate.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
DA-E1	DA-E1-01	DOCUMENT the data analysis in a manner that facilitates PRA applications, upgrades, and peer review.	The documentation of the special basic events (Section 5) should be enhanced to provide a better basis for the values used. Sensitivity studies should also be considered to assess the impact of these events on the overall risk results.	**DOCUMENTATION – COMPLETE** This section and table of special variables were made explicit to ensure their visibility, documentation to facilitate future peer, applications, etc. Also, finding is not very specific with regard to which special variables are of concern if any. Also, Section 6 of the Data Analysis notebook explicitly refers to these as containing potentially important assumptions that can be assessed from the Quantification notebook. This is an opportunity for potential Improvement In the future as are numerous assumptions throughout the PRA (this is not a finding). Some of these variables are set to 1.0 and are place holders for future updates (e.g., Bennett's Bridge, Portable Charger).

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
SY-A11	SY-A11-01	INCORPORATE the effect of variable success criteria (i.e., success criteria that change as a function of plant status) into the system modeling. Example causes of variable system success criteria are: (a) different accident scenarios. Different success criteria are required for some systems to mitigate different accident scenarios (e.g., the number of pumps required to operate in some systems is dependent upon the modeled initiating event); (b) dependence on other components. Success criteria for some systems are also dependent on the success of another component in the system (e.g., operation of additional pumps in some cooling water systems is required if *noncritical loads are not isolated*); (c) time dependence. Success criteria for some systems are time-dependent (e.g., two pumps are required to provide the needed flow early following an accident initiator, but only one is required for mitigation later following the accident); (d) sharing of a system between units. Success criteria may be affected when both units are challenged by the same initiating event (e.g., LOOP).	The system notebooks in Section 2.6 provide the success criteria but do not evaluate EDG mission time for different LOSP type initiators.	**NO IMPACT – COMPLETE** NMPNS disagrees this is a finding. Added explanation to SY.01c that adding convolution for each LOSP cause and recovery would result in different EDG recovery times for each LOSP cause group, but once weighted correctly it would give the same result as already done with average weighted LOSP and recovery convolution case. If future application is required, then breaking out LOSP causes and modeling of this level of detail is required.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
SY-B10	SY-B10-01	When modeling a system, INCLUDE appropriate interfaces with the support systems required for successful operation of the system for a required mission time. (See also AS-A6.) Examples include: (a) actuation logic (b) support systems required for control of components (c) component motive power (d) cooling of components (e) any other identified support function (e.g., heat tracing) necessary to meet the success criteria and associated systems	Diesel generator modeling needs to include actuation logic and air compressors.	**DOCUMENTATION – COMPLETE** EDG failure history shows that the fast start system (actuation logic) is very reliable and has not resulted in any failures to start the EDG. The reliability of the fast start sequence is subsumed in the failure data used to determine the EDG failure rates. Modeling the sequencer along with detailed modeling of its components will not provide additional risk insights and will add unnecessary complexity to the model. This precludes the need to explicitly model the fast start sequencer as a subsystem outside the EDG boundary and this explanation has been documented in SY.01c Section 3.1. All the components of the air start system are tested during the monthly EDG operability test; procedure N1-ST-M4 A (B). Air start system failures are captured in the data used to evaluate EDG failures. This precludes the need to explicitly model the air start system as a subsystem outside the EDG boundary. Regarding the air receivers staying at proper pressure for 24 hours, this is dependent on the availability of AC power (EDG), the reliability of the two air compressors, and the condition of the piping. EDG recovery is limited to 8 hours due to uncertainty about air availability beyond 8 hours. Condition reports were reviewed and no evidence was found that leakage and air compressor reliability were problematic to the extent that additional modeling detail was required. SY.01c Section 3.1 has been enhanced to document this modeling.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
IE-D3	IE-D3-01	DOCUMENT the key assumptions and key sources uncertainty with the initiating event analysis.	The IE notebook (as well as the other PRA notebooks) contains a discussion of the assumptions used. The notebooks also provide a discussion of plant-specific sources of uncertainty. However, this SR requires a systematic evaluation of all sources of uncertainty, including industry-wide issues with data and modeling approaches. Recent EPRI reports are available that document generic industry uncertainty sources. These items should be reviewed for applicability to NMP1 and added to the PRA documentation.	**DOCUMENTATION – OPEN (see QU-E4)** **NO IMPACT ON FPRA MODEL** From initial reviews, it is agreed this will mostly involve more documentation and sensitivity studies. During a subsequent Unit 2 PRA update, the QU notebook was enhanced to address this subject including NUREG-1855 and EPRI Guidance (1016737); industry generic resolution to this requirement is to address sensitivities and uncertainties further in applications. Until the NMP1 QU notebook is updated, the Unit 2 review will be used as guidance for applications.
SY-B13	SY-B13-01	DO NOT USE proceduralized recovery actions as the sole basis for eliminating a support system from the model; however, INCLUDE these recovery actions in the model quantification. For example, it is not acceptable to not model a system as HVAC or CCW on the basis that there are procedures for dealing with losses of these systems.	Section 3.1.1 assumption in system notebook SY-01c (page SY-01-21) violates the Capability Category II requirements.	**NO IMPACT – COMPLETE** This refers to assumption 11 where it was noted that failure of the lockout relays 86-16 and 86-17 to reset was modeled within the basic event for operator action. These relays were added to the model and assumptions 10 and 11 were deleted. This had an insignificant impact on results, but adds to model completeness.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
IF-D5a	IF-D5a-01	GATHER plant-specific information on plant design, operating practices, and conditions that may impact flood likelihood (i.e., material condition of fluid systems, experience with water hammer, and maintenance-induced floods). In determining the flood-initiating event frequencies for flood scenario groups, USE a combination of (a) generic and plant-specific operating experience; (b) pipe, component, and tank rupture failure rates from generic data sources and plant-specific experience; and (c) engineering judgment for consideration of the plant-specific information collected.	Provide an in depth discussion about the applicability of generic flood data to NMP1 in the notebook or update the flood frequencies over the years with NMP1 data which would be typically zero flood events.	**DOCUMENTATION – COMPLETE** Revised Section 5.2 to indicate that plant specific CR (Condition Report) search did not turn up any significant events that required Bayesian update to generic data. Note that a review of potential maintenance induced events is addressed in the notebook.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
DA-C8	DA-C8-01	When required, USE plant-specific operational records to determine the time that components were configured in their standby status.	Alignment fractions are estimated based only on the number of trains available (e.g., 1 of 2 = 50% alignment fraction). This is adequate for Category I. For Category II, need to base fractions on actual operating experience.	**NO IMPACT – COMPLETE** The current basis for alignment fractions is not exhaustive and additional data analysis could be performed. However, this effort will result in a very minimal impact on the model. The alignment basic events were reviewed for importance in the model. None had a significant RAW or Fussel-Vessely value. Highest RAW was 1.03 for CRD pump 11 in standby. Fussel-Vessely values were also small. The only value greater than 5E-3 was 2.6E-2 for CRD pump 11 in maintenance. From experience, one CRD pump is always in standby and equipment rotation is important for these pumps. Additionally, rotation of this equipment has been discussed in system engineer interviews. The 50% alignment fraction is deemed most appropriate and more detailed data analysis will not significantly alter the values. These explanations were added to the DA notebook Section 6.
MU-C1	MU-C1-01	The PRA configuration control process shall consider the cumulative impact of pending changes in the performance of risk applications.	CNG-CM-1.01-3003, PRA Configuration Control, Section 5.13, PRA Applications documents a current living applications list exists but was not located.	**PRA CONFIGURATION CONTROL PROCEDURE – OPEN NO IMPACT ON FPRA MODEL**

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
DA-D4	DA-D4-01	When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: (a) confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered (e) confirmation of the reasonableness of the posterior distribution mean value.	Section 2.7 (3rd paragraph) makes a statement that Bayesian results are reasonable. However, there is no discussion of the criteria that is used. In several data variables, the plant evidence seems to be quite different from the prior. These should be investigated with regard to the Bayesian update process to assure the plant point estimate is not in the extremes of the prior distribution. Example: failure rate GAZR1 has a prior of 2.90e-3 and posterior of 4.69e-3, while the plant information is 2 failures in 240 hrs (0.008 per hr). Similarly, failure rate VMZD1 has a prior of 1.07e-3 and posterior of 3.03e-3, while the plant information is 4 failures in 550 hrs (0.007 per demand).	**DOCUMENTATION – COMPLETE** DA-D4 (e) was done, but not documented well. An additional explanation is provided at the end of first paragraph in Section 2.7. The cited examples were reviewed and determined to be reasonable. Table 2-7 was also added to the DA notebook. This table compares the prior and posterior and discusses the reasonableness of the posterior of each data variable that received a Bayesian update.
SY-B11	SY-B11-01	MODEL those systems that are required for initiation and actuation of a system. In the model quantification, INCLUDE the presence of the conditions needed for automatic actuation (e.g., low vessel water level). INCLUDE permissive and lockout signals that are required to complete actuation logic.	The diesel generator initiation system is not completely modeled.	**DOCUMENTATION – COMPLETE** See SY-B10

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
IF-C8	IF-C8-01	USE potential human mitigating actions as additional criteria for screening out flood sources if all the following can be shown: (a) flood indication is available in the control room; (b) the flood source can be isolated; and (c) the mitigating action can be performed with high reliability for the worst flood from that source. High reliability is established by demonstrating, for example, that the actions are procedurally directed, that adequate time is available for response, that the area is accessible, and that there is sufficient manpower available to perform the actions.	This finding has a relationship to the suggestion in F&O IF-C6-01. Table 4-5 uses the term YES with very little descriptive matter other than the criteria prior to the table in the IF notebook. In order to fully review this as per the standard more detail about alarms or operator intervention needs to be provided.	**DOCUMENTATION - COMPLETE** Improved Section 4.6.2 by adding reference to applicable screening criteria. Added note to Table 4-5 to reference Section 4.6.2. Also, note that ASME Quality Table references Section 4.6, which explains the screening (appears that reviewer did not see this, otherwise probably would not be a finding).
IF-C3	IF-C3-01	For the SSCs identified in IF-C2c, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. EITHER: a) ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions; OR b) NOTE that these mechanisms are not included in the scope of the evaluation.	Submergence, pipe whip, and environmental effects are not addressed as per the ASME standard Category II. Reg. Guide 1.200 has more information on this and there are NRC overrides identified in the standard that need to be addressed.	**DOCUMENTATION - COMPLETE** Submergence is addressed, spray/impingement are considered and documented. Sections 4.5, 2.3, 3.4 and 4.7 revised to make it clearer that these types of impacts were considered. Note also that Table 2-1 identifies areas where HELB analysis is considered in Appendix B and IE notebook.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
IF-E5a	IF-E5a-01	For all human failure events in the internal flood scenarios, INCLUDE the following scenario-specific impacts on PSFs for control room and ex-control room actions as appropriate to the HRA methodology being used: (a) additional workload and stress (above that for similar sequences not caused by internal floods) (b) cue availability (c) effect of flood on mitigation, required response, timing, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) (d) flooding-specific job aids and training (e.g., procedures, training exercises).	The requirements for this category are not met. Flooding effects on other HEPs in the Transient event accident sequence logic does not appear to be accounted for. The Internal Flooding HRA did not include an evaluation of how HFEs in the base (i.e., non-flooding) model that are credited in the flooding scenarios would be impacted by the occurrence of the flood event. The HR analysis should be updated to reflect either walkthroughs or talk-throughs of the flooding event sequences with plant operations.	**DOCUMENTATION - COMPLETE** This has been addressed with HR-E3-01 and is documented in Appendix C of the HR (Human Reliability) notebook. This included talk-through with Operations and consideration of impacts with and without successful isolation as well as other operator actions in the model.
HR-E3	HR-E3-01	TALK THROUGH (i.e., review in detail) with plant operations and training personnel the procedures and sequence of events to confirm that interpretation of the procedures is consistent with plant observations and training procedures.	During development of the flooding HEPs, neither plant operations nor training personnel were contacted for the review of procedures and anticipated sequence of events. Also, the flooding HEP response models were not confirmed using simulator observations/talk-throughs.	**DOCUMENTATION - COMPLETE** This has been addressed with HR-E3-01 and is documented in Appendix C of the HR notebook. This included talk-through with Operations and consideration of impacts with and without successful isolation as well as other operator actions in the model.
LE-F1a	LE-F1a-01	PERFORM a quantitative evaluation of the relative contribution to LERF from plant damage states and significant LERF contributors from Table 4.5.9-3.	The LERF accident sequences are quantified as a function of the Level 1 accident sequence classes. This was determined to not satisfy the ASME requirement that the relative contribution to LERF from plant damage states and significant LERF contributors are quantified in terms of the LERF contributors identified in Table 4.5.9-3 (e.g., pressure suppression bypass, isolation condenser tube rupture, etc.).	**DOCUMENTATION – COMPLETE** This is a useful comparison and has been included in new Section 4.3.8 of the QU notebook. No new plant damage states are required nor should they be developed to provide this breakdown.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
QU-C1	QU-C1-01	IDENTIFY cutsets with multiple HFEs that potentially impact significant accident sequences/cutsets by re-quantifying the PRA model with HEP values set to values that are sufficiently high that the cutsets are not truncated. The final quantification of these post-initiator HFEs may be done at the cutset level or saved sequence level.	In doing sensitivity studies to meet this SR, use HEP values higher than nominal; e.g., 0.1 to ensure that all dependencies are captured.	**DOCUMENTATION - COMPLETE** The "0.1 HEP" sensitivity calculation has been completed and is documented in Section 3.6 of the HRA notebook.
QU-D1a	QU-D1a-01	REVIEW a sample of the significant accident sequences/cutsets sufficient to determine that the logic of the cutset or sequence is correct.	Appendices to the QU notebook present the top 200 CDF and LERF cutsets. The top 20 CDF cutsets are specifically discussed in section 4.2.3 in the context of plant response, significant assumptions made, etc. While the top 200 cutsets are included, the analysis of these cutsets should be expanded to include a greater number of cutsets, as the top 20 only constitutes about 60% of the CDF and is dominated by cutsets with only an initiator and one failure (i.e., does not demonstrate a comprehensive review).	**DOCUMENTATION – OPEN (SEE QU-D4)** **NO IMPACT ON FPRA MODEL** The SR says to review a sample, not 99%. A significant number of accident sequences were originally reviewed during model development and as part of final results review. This also included a review of non-significant sequences in the model (QU notebook Section 4.3.5). This is only considered open based on the assumption of the need to document and provide evidence of this review. This is a nice-to-do that will be considered in future updates.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
DA-C14	DA-C14-01	For each SSC for which repair is to be modeled (see SY-A22), IDENTIFY instances of plant-specific or applicable industry experience and for each repair, COLLECT the associated repair time with the *repair time being the period from* identification of the component failure until the component is returned to service.	The Recovery/Repair of equipment is generally neglected in the model except for offsite power recovery, diesel recovery, instrument air initiating event recovery and *screenhouse recovery (these are* addressed in Section 5). Industry data is used for DG recovery, and no discussion is provided concerning plant-specific repair. Also, the industry data used was not reviewed for applicability. Instrument air and screenhouse recoveries use a screening value based on long time to recover.	**NO IMPACT – COMPLETE** No screening of LOSP or EDG recovery is used - generic data from NUREG is acceptable. Documentation was added to IE *notebook Section 4.2 describing* the review of plant-specific events and justifying why a Bayesian analysis is not required.
AS-A8	AS-A8-02	DEFINE the end state of the accident progression as occurring when either a core damage state or a steady-state condition has been reached.	In SBO trees, late recovery of power is taken to OK state without checking for system availability. This treatment can be improved by checking for availability of the mitigating systems.	**NO IMPACT – COMPLETE** Resolved by model update. Injection (top event INJ) and heat removal (top event CHR) were added to the SBO model and are required for success when AC is recovered. This had a negligible quantitative impact on results, but adds completeness to model.
MU-B4	MU-B4-01	PRA Upgrades shall receive a peer review (in accordance with the requirements specified in Section 6 of the ASME PRA Standard) for those aspects of the PRA that have been upgraded. Refer to Section 2 of the ASME PRA Standard for the distinction of a PRA Upgrade versus PRA maintenance and update.	CNG-CM-1.01-3003, PRA Configuration Control, Section 5.12, PRA Revisions needs to implement requirement that "External peer review is required for PRA upgrades."	**DOCUMENTATION – PRA PROCEDURE – COMPLETE** Section 5.12.A.1.g of procedure contains requirement that "external peer reviews required for PRA upgrades."

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
QU-D5b	QU-D5b-01	REVIEW the importance of components and basic events to determine that they make logical sense.	Some insights from the importance listings (for equipment and operator actions) are discussed in Section 4.2.4 of QU NB. However, further discussion should be provided to specifically address the requirements of this SR. Provide a more detailed discussion how this SR is met. QU NB does not have adequate discussion.	**DOCUMENTATION – OPEN** **NO IMPACT ON FPRA MODEL** This was done, but needs to be documented. Add more discussion of symmetry review (Section 4.2.5) and plant understanding, etc. (probably to Section 4.3.2, which was not referenced in this finding, but is referenced in the ASME Quality table for this SR). Make sure Section 4.2.4 reasonableness check and comparison is referenced in Section 4.3.2 and vice versa. Reasonableness check for HRA has been enhanced via comparison with Oyster Creek (Section 3.5 of the HRA Notebook). Improve documentation in future updates as appropriate.
HR-G6	HR-G6-01	CHECK the consistency of the post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to check their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	An overall “global operator dependency failure” event is applied to account for a complete breakdown in crew functionality.	**NO IMPACT – COMPLETE** Long Term Loss of Heat Removal Dependency group (ZQDHR) added to show that the dominant dependent groups were associated with this action. The global action (ZQQQQ) is now a small contributor and not masking other contributors.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
HR-G7	HR-G7-01	For multiple human actions in the same accident sequence or cut set, identified in accordance with supporting requirement QU-C1, ASSESS the degree of dependence, and calculate a joint human error probability that reflects the dependence. ACCOUNT for the influence of success or failure in preceding human actions and system performance on the human event under consideration including (a) the time required to complete all actions in relation to the time available to perform the actions (b) factors that could lead to dependence (e.g., common instrumentation, common procedures, increased stress, etc.) (c) availability of resources (e.g., personnel) [Note (1)].	The ZQQQQ_DEPOPERATO event is included in the model to consider the potential for a cross-cutting operator failure during an accident. The basis for the numerical value assigned to this event, while not unreasonable, is not well-established. The event participates in the dominant cutsets, and may be masking the risk contribution from other failures.	**NO IMPACT – COMPLETE** Long Term Loss of Heat Removal Dependency group (ZQDHR) added to show that the dominant dependent groups were associated with this action. The global action (ZQQQQ) is now a small contributor and not masking other contributors.
QU-D4	QU-D4-01	REVIEW a sampling of nonsignificant accident cutsets or sequences to determine they are reasonable and have physical meaning.	Section 4.3.5 of the QU notebook briefly notes that a review was performed. However, there is no evidence presented in the notebook. The QU notebook should include a sampling of several non-significant cutsets and demonstrate that these cutsets correctly represent plant features, operator actions, and expected plant behavior.	**DOCUMENTATION – OPEN (See QU-D1a)** **NO IMPACT ON FPRA MODEL** SR does not require explicitly that this be documented; thus, the interpretation is that some evidence with a sampling be documented along with a better explanation of the review process. This will be considered in future updates (QU notebook Section 4.3.5).
QU-E4	QU-E4-01	EVALUATE the sensitivity of the results to key model uncertainties and key assumptions using sensitivity analyses [Note (1)].	Section 5.2 in the QU notebook addresses this issue qualitatively. More sensitivity runs need to be done to evaluate model uncertainties; e.g., set all HEPs, CCFs etc. at 5th and 95th percentile during quantification.	**DOCUMENTATION – OPEN (see IE-D3)** **NO IMPACT ON FPRA MODEL** From initial reviews, it is agreed that this will involve more documentation and sensitivity studies to support applications.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	F&O ID	CATEGORY II REQUIREMENT	Finding/Observation	Disposition
QU-F6	QU-F6-01	DOCUMENT the quantitative definition used for significant basic event, significant cutset, and significant accident sequence. If other than the definition used in Section 2, JUSTIFY the alternative.	This is not documented in the QU notebook. This SR is therefore not met. Need to provide a discussion in the QU notebook as to how this topic is met.	**DOCUMENTATION – COMPLETE** Adopted ASME Section 2 definition as footnote to QU notebook Section 4.2.2.

V. Fire PRA Quality

79 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."*

The NMP1 Fire PRA Peer Review was performed October 24 to 28, 2011 using the NEI 07-12 Fire PRA peer review process, the ASME PRA Standard (ASME/ANS RA-Sa-2009) and Regulatory Guide 1.200, Rev. 2. The purpose of this review was to establish the technical adequacy of the Fire PRA for the spectrum of potential risk-informed plant licensing applications for which the Fire PRA may be used. The 2011 NMP1 FPRA Peer Review was a full-scope review of all of the technical elements of the NMP1 at-power Fire PRA (2011 Model of Record) against all technical elements in Section 4 of the ASME/ANS Combined PRA Standard, including the referenced internal events Supporting Requirements (SRs). The Peer Review noted a number of facts and observations (F&Os). The F&Os and the disposition of the F&Os are provided in Table V-1. All F&Os are being provided and have been dispositioned. The Fire PRA is adequate to support the NFPA 805 Licensing Basis.

The Fire PRA meets Capability Category II in most but not all cases. A limited number of ASME/ANS areas were identified by the peer review team as meeting Category I only requirements. The capability categories are defined in ASME/ANS RA-Sa-2009. An evaluation of the impact of those areas where only the Capability Category I requirement was met is provided in Table V-2.

Table V-1 Fire PRA Peer Review – Facts and Observations

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
CF-A1	CF - Circuit Failure	Closed	Based on a review of the results, and performance of several ones-runs, there are a number of spurious operation probability events that are presently set to 1.0 that are potentially significant. Dominant to the results is the spurious opening (or failure to close due to hot short) of the MSIVs. However, other components show up in the results (e.g., BKR_R1042__CAZM1). Based on discussions with NMP1 PRA, the determination for the existing CF analysis was based on a previous model, and additional review for needed CF probability analysis has not been performed. (F&O ID 1-22)	Consideration of spurious opening of the air operated MSIVs (IV-01-03 and IV-01-04) was added to the CF notebook. In addition, to identify additional candidates for CF likelihood calculations, FRANX "Ones" runs were performed for fire scenarios with CCDP < 0.1 and CDF > 5E-7. The CCDP constraint is based on the fact that Ones runs have not been successful when CCDP > 0.1. The CDF constraint is meant to limit the number of scenarios to a manageable number while capturing most of the remaining aggregate risk. LERF Ones runs were also performed for the same set of scenarios. The cut set files for the Ones runs were combined into a combined CDF and LERF cut set files. From these cut set files, importance measures were calculated. For top risk scenarios with CCDP > 0.1, Browser-based cut set reviews were performed using the regular FRANX Trues cut sets. By examining the candidates identified by the importance measures and the Browser reviews, four additional components were identified for CF likelihood calculations and the CF notebook was revised (Section 2.3.2.2) to include probability estimates for the additional components. The four components added are spurious closure of AC circuit breakers BKR-(16/008B)R1042/602 (POWER CIRCUIT BREAKER - 16A / 16B BUS TIE) and BKR-(17/007B)R1052/612 (POWER CIRCUIT BREAKER - A/B BUS TIE); and spurious closure of feedwater pump discharge block valves VLV-29-08 (11 FEEDWATER PUMP DISCHARGE BLOCK VALVE) and VLV-29-09 (12 FEEDWATER PUMP DISCHARGE BLOCK VALVE).

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
FQ-A2	FQ – Fire Quantification	Closed	A review of the fire sequence T3B~PB-12-EC-8-2 (and similar ones) shows that it becomes an SBO sequence with the following observation: 1. Direct failure of PB 12 due to fire 2. PB 11 due to the failure of reserve transformer 101N due to interlock: One question on this – Could breaker 8106 open after the 101S upstream breaker opens as a result of faults at PB12? 3. PB 103 failure due to: a. Loss of reserve transformer 101S b. PB12 failures causes the breaker failure to load EDG 103. 4. PB 102 failure due to the following failures: a. Loss of reserve transformer 101N b. EDG 102 failure due to room cooling failure, which is in turn failed by the power supplies. The failures modeled under gate AC-PB16B-S1MCC show that it was introduced by the failure of BE BKR_R1-43__CAZND. Tracing this BE in FRANX shows that cable 16-3 failed, which apparently passed above PB12. No spurious operation failure probability was evaluated for this cable failure, which may reduce its risk contribution by 2/3. Operator action to open door for EDG room cooling was failed in this scenario, which is not expected since the room heatup may not be that limiting. Please provide room heatup calc if available. NMP response concludes that the heatup calcuation for this room does not show that opening the door provides sufficient cooling if the fan is not running. It is assumed that even if the operator opens the door when the fan is failed, the cooling for the room will not be adequately restored. Therefore, this operator action is treated as failed. Since the cable 16-3 is assumed to be a control cable (routing from C3 all the way to EDG room), it should be assumed that operator can manually close the spuriously open breaker after the fire-induced hot short is cleared or manually bypass the faulted control. Such operator action should be postulated. NMP response confirms that this cable is a control cable for BKR-(16B/013B)R1043/603. Manual operation is possible for this breaker because it is a 600V breaker with manual spring charging capability. (F&O ID 2-32)	With further FPRA model development, including detailed circuit analysis and introduction of proposed modifications and new operator actions, SBO is no longer the primary accident sequence for the fire scenarios T3B~PB-12-EC-8-2, T3B~PB-12-EC-8-1, T3B~PB-12-EC-8-2, and T3B~PB-12-HVHEAF-7-1. Although SBO still occurs in some cut sets of these scenarios, a new operator action to close the spuriously open breaker has not been added due to the reduced risk associated with these scenarios. (The results presented during the peer review showed a CCDP of 0.341 for T3B~PB-12-HVHEAF-7-1, and a CDF contribution of 2.71E-5. Results as of 02/26/12 give a CCDP of 0.0104 and a CDF contribution of 7.29E-7.) The cable 16-3 can cause spurious opening basic event BKR_R1043__CAZND as has been noted. To evaluate the importance of this event, a sensitivity run of FRANX was performed (02/26/12) with cable 16-3 excluded from damage. This resulted in a modest reduction of the CDF contribution of the scenario T3B~PB-12-HVHEAF-7-1 from 7.29E-7 to 5.92E-7. Therefore, this basic event is not considered further for a circuit failure probability calculation. In addition, during the cut set reviews during the week of 12/12/2011, other SBO scenarios were refined by further model development including proposed plant modifications and new operator actions. As noted in the finding, opening the EDG room door is not credited to provide sufficient cooling if the fan is not running. That is, even if the operator opens the door, if the fan is failed, the cooling for the room will not be adequately restored. Therefore, this operator action is treated as failed for the fire PRA. The event name "BKR_R1-43__CAZND" (noted the finding) is not a valid name and contains a typographical error. The event intended in the finding is BKR_R1043__CAZND.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
CS-A5	CS – Cable Selection	Closed	The provide example of detailed circuit failure show that FW Isolation Valve 31-07's power cable would fail as-is. However, the power cable 161-55 associated with this component still fails basic events IV_31_07___VMZMF, IV-31-07 MOV Transfers Closed. IV_31_07___VMZMF is also altered in FRANX to a probability of 0.32. Please verify whether this valve is normally powered or not. If normally powered, the failure of power cable should have no affect since it is fail AS-IS. If it is not normally powered, the power cable failure should have a failure probability less than 0.32 if three-phase Proper Polarity Hot Shorts are assumed. In addition, the failure of this valve should not be simply the availability of power cable. The failure should be AND'ed with fire-induced spurious control failures. (F&O ID 2-7)	The F&O refers to the power cable 161-55 being inappropriately selected. However, as a result of detailed circuit analysis, the power cables are no longer mapped to these BEs (spurious closure basic events IV_31_07___VMZMF and IV_31_08___VMZMF). The CF notebook has been revised to reflect the cables currently mapped to these BEs (Table B-1). Power is not removed from the feedwater isolation valves IV-31-07 and IV-31-08 (CF notebook, Section 2.3.2.2). Therefore, the three-phase proper polarity hot short probability cannot be applied, and the CF probability calculation in the CF notebook stands. These are motor operated valves that fail as is on loss of power. Only control cables are mapped to the spurious closure basic events IV_31_07___VMZMF and IV_31_08___VMZMF. Because the power cables are not considered for spurious closure events, the fault tree logic is correct as is.
CF-B1	CF – Circuit Failure	Closed	The uncertainty bounds used for the probability of exceeding a hot short duration of 15 minutes for AC circuits is likely a higher uncertainty than assigned. The analysis performed by GEH was based on a limited set of data and testing. Recent DC testing indicates that there may be conditions where the hot short may not clear. One condition, for example, would be the damage of a thermoplastic cable at low temperature. (F&O ID 1-20)	The upper bound for the probability distribution for AC hot short duration is set to its highest value that would accommodate the uniform distributional form and preserve the nominal value. It is not clear that the value from the draft DC hot short probability curve should be preferred. Therefore, the suggestion is not taken and the originally assigned value is kept.
CF-A1	CF – Circuit Failure	Closed	Events for high rad monitor alarm are not included in the CF report. Events are set to true in the analysis, but could be caused by a hot short where a CF probability can be assigned.This includes events RAMRN06A11_IIZL1, RAMRN06B11_IIZL1, RAMRN06A12_IIZL1, and RAMRN06B12_IIZL1. In addition, there are two annunciator basic events identified in the ES notebook which are not included in the CF report. (ANN_F1_2_7_IAZL1, ANN_F4_2_2_IAZL1). (F&O ID 1-21)	The spurious operation list in the CF notebook was revised to include the additional basic events mentioned in the finding. These events were considered for application of CF probabilities.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
ES-A1	ES – Equipment Selection	Closed	Although the NMP1 Initiating Event review identifies the possible causes for each event, a list of plant equipment causing each initiating event was not developed. As a result, the cables that may be damaged to cause each Initiating Event are not identified against the possible IE. Based on questions to NMP1, the identification of areas causing a loss of %B2X (Loss of Powerboard 12 Initiating Event) was performed by walkdown. In this example, the IE was only assigned for fires in Panel 5A. No IE was assumed for failure of cables affecting %B2X. The assignment of the Initiating Event for each fire scenario in FRANX is not based on specific equipment damage or combinations of equipment damaged in order to determine the fire-induced initiating event modeled in the FPRA. (F&O ID 1-10)	The selection of either %RPS or %TT as the initiating event for all fire scenarios is justified by the structure of the fault tree logic. All sequences that represent equivalent sequence progressions are combined in the fault tree. In the top logic for the sequences, all potential initiating events are input into the logical representation of the sequence. Whenever any initiating event other than %RPS or %TT appear in the initiating event logic portion of the fault tree, one or both of %RPS and %TT also appear. Therefore, the selection of these two initiating events covers all potential accident sequences associated with potential fire scenarios. The inputs into the required mitigating system failures (including support systems) include both the random component failures and the initiating events that could result in the failure of the mitigating system. Again, as with the initiator portion of the fault logic, whenever an initiating event other than %RPS or %TT appears in the mitigating system failure logic one or both of these two initiating events also appears. Therefore, in evaluation of individual fire scenarios, the fire induced failure of components that are both part of an initiator and the mitigating/support systems are properly modeled in the fault tree logic. Since only the two initiating events are used in the fire model and all components susceptible to fire damage are explicitly modeled, a table correlating components to initiating events was deemed to be unnecessary. To verify that the results are insensitive to the selection of initiating events other than %RPS and %TT several sensitivity analyses were run. In these analyses, the top ten scenarios were evaluated to assess which support systems would fail given the modeled fire and which resulting initiating event other than %TT and %RPS would result. The scenarios were rerun with the identified support system failure as the initiating event. The initiators selected for the sensitivity runs included loss of offsite power, loss of feedwater, loss of instrument air, and loss of service water. The results from this analysis were incorporated into the Uncertainty Analysis notebook. An expanded discussion of the selection of initiating events was added to Section 3.1.1 of the Plant Response Model (PRM) Notebook.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
CF-A1	CF – Circuit Failure	Closed	There are numerous components listed as CableCode = Blank and nothing under comment and blank under MEL. For example; RLY_K8616__RAZD1. Response to a question on this indicates that recent detailed cable mapping has been done and not all the cable codes appear to be populated. (F&O ID 1-6)	A review of the basic event table in the nmp1.rr database was conducted. For records having a blank in the cablecode column, a cablecode was added and a cablecode comment was provided as appropriate. The MEL field was verified to be populated where the basic event was tied to a specific component.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
ES-A5	ES – Equipment Selection	Closed	It is not apparent from the documentation that a systematic review of MSO combinations that would cause an initiating event, simultaneously affect the operability of equipment required to mitigate the event, result in an initiating event where mitigating systems are not addressed in the current Appendix R analysis or result in loss of RCS integrity was performed. E.g. Loss of service water pumps 11 and 12 will lead to a loss of normal service water initiating event and cause loss of normal service water. This was not considered as a fire induced initiating event that would also cause loss of the function. (F&O ID 6-7)	The selection of components that could be impacted in individual fire scenarios was based on the components modeled in the internal events PRA. As identified in the response to Finding 1-10 (addressed in the Equipment Selection (ES) Notebook), system failures that can result in both an initiating event and the failure of mitigating/support systems are explicitly identified during the evaluation of the individual fire scenarios. Because the selection of components to be addressed in the fire analysis was based on the internal events PRA, they are not limited to those identified as part of the Appendix R analysis. Specifically addressing the Service Water example. Any fire scenario resulting in the loss of operability of the two service water pumps would be explicitly modeled in the fire impact tables of FRANX. Damage to the cables results in failure of the pumps. For example, in the fault tree gate S3-1PUMP-12-R has as inputs the failure of pump NSW-12 from random failures and as an initiating event. Any fire that would fail this pump would result in the propagation of the failure through the system logic. As stated in the response to F1-10, the initiating event portion of the sequence logic for which loss of service water is a potential initiator would be addressed through the %RPS or %TT initiating events. Thus failure of service water as both an initiator and as a mitigating system failure is addressed. To verify that the results are insensitive to the selection of initiating events other than %RPS and %TT several sensitivity analyses were run. (This analysis was performed in response to issue 1-10.) In these analyses, the top ten scenarios were evaluated to assess which support systems would fail given the modeled fire and which resulting initiating event other than %TT and %RPS would result. The scenarios were rerun with the identified support system failure as the initiating event. The initiators selected for the sensitivity runs included loss of offsite power, loss of feedwater, loss of instrument air, and loss of service water. The results from this analysis were incorporated into the Uncertainty Analysis notebook. The Loss of Service Water was included in this analysis.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
ES-A1	ES – Equipment Selection	Closed	Feedwater Overfeed was not included as an IE in the Fire PRA, as described in Scenario 2ak of the MSO report. The event is included in the PRA logic for a possible subsequent overfeed, which is similar to the internal events logic (see 2.2.2.6 of the IE notebook). However, since the Fire may impact the FW controls, Level 8 Signal, and the Operator Error failure (including success criteria and timing); this should be considered as a specific Fire IE. Also, a FW overfeed may result in water relief through the SRVs/ERVs, and a consequential LOCA. This impact is not included in either the FPRA or internal events PRA. (F&O ID 1-11)	The response analysis for actions in a post fire event are deemed to be sufficiently conservative to allow for any minor timing differences that may result from differences in the sequential failures (i.e. a reactor trip followed by a loss of feedwater versus a loss of feedwater induced trip) in any accident sequence. Feedwater overfill to the extent that it would lead to a liquid flow through the ERVs was not modeled in either the internal events PRA or the Fire PRA. The sequence of events modeled in the PRA for an overfill scenario does not explicitly model overfill as causing a LOCA (ERVs open and then fail to reclose), but does model the following that essentially covers the same impact given overfill and is conservative: (1) EC success requires operator action to drain RPV etc (during this time they would also be trying to prevent continued overfilling), (2) EC failure in the model requires emergency depressurization (another operator action) to allow low pressure injection success (i.e., it is not assumed that overfill continued to the point of opening ERVs and providing successful depressurization), and (3) Feedwater is assumed to fail when the EC failure and ADS failure occur, which conservatively provides overfill but then fails. The continued operation of the feedwater system to provide injection cooling is not credited.
ES-B4	ES – Equipment Selection	Closed	Table G-1 includes a summary of the FPRA components included in the model. This includes a list of primary and sub-components (see sub-component column). However, if the FPRA did not include a basic event for the subcomponent, the subcomponents were never listed in the FPRA equipment list. Functionally, the AREVA cable selection process traced all of the subcomponent cables and circuits, based on the NEI 00-01 process and the AREVA circuit selection process. So, although a list of subcomponents, including power supplied, interlocks and instrumentation is not available, the impacts are included in the model, including spurious actuation. Overall, the inclusion of subcomponents in the FPRA model is inconsistent, with some modeling of handswitches, relays, interlocks and power supplies, but not all. F&O ID 1-12)	A list of subcomponents from the Fire PRA was generated and was correlated to a primary component. This list was reviewed to verify that the cable selection for the primary components included the cables for the subcomponent failure modes. A list of subcomponents correlated to primary components was added to the ES notebook. Additionally, a description of the review of the subcomponent to primary component relationship and modeling was added to Section 2.1 of the ES notebook.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
ES-B2	ES – Equipment Selection	Closed	NEI 00-01, revision 3 was recently issued with a new version of the BWROG MSO list. This revision of the MSO list should be reviewed for possible impact to the FPRA. (F&O ID 1-19)	The NMP1 NFPA 805 analysis and submittal is based on Revision 2 of NEI 00-01. The ES notebook section 2.4 documents that NEI 00-01 Revision 2 is the basis for the NMP1 multiple spurious operation review. Future PRA updates will consider Revision 3 of NEI 00-01 as appropriate. In addition, a review of NEI 00-01, Revision 3 was conducted against the guidance from NEI 00-01, Revision 2. There were no gaps relative to MSOs identified.
ES-B2	ES – Equipment Selection	Closed	Table D-1 of the ES Notebook includes discussion that an RWCU LOCA can be caused by 33-02, 33-04 or 33-41 or 37-07 in combination with spurious opening of 33-39. However, the logic under the CAFTA gate RWCU includes 33-41, 33-04, 33-02, and 33-39. Note that the same error is in the Expert Panel Review. Per response to a peer review question, the component 37-07 is a typo. (F&O ID 1-3)	Reference to BV-37-07 in the comment field for IV-33-02, IV-33-04, BV-33-41, and PCV-33-39 was removed. The comment field for BV-37-07 was revised to delete the MSO description. The "MSO" and "Hi consequence" check boxes were removed for BV-37-07 and the "Exclude from Fire PRA equipment list" checkbox was marked. These changes affected the ES notebook Tables D-1 and G-1.
ES-B1	ES – Equipment Selection	Closed	[TJA] N1-ES-F-001 rev 0 states in section 2.2 'the fire safe-shutdown equipment list (SSEL) [Ref. 3] was reviewed to determine whether additions to the FPRA equipment list were appropriate. Table C-1 lists the equipment from the SSEL that was not found to be needed in the FPRA along with the rationale for rejection. The remaining equipment on the SSEL is included on the FPRA equipment list'. Reference 3 is document 51-9121684-000 which is the SSEL. There are a number of components on the SSEL that are not listed in Table C-1 as being excluded from the FPRA equipment list and are not included on the FPRA equipment list. Table B-1 lists primary PRA components from the internal events PRA that are included on the FPRA list. Section 2.1 of N1-ES-F001 documents the review of the event trees and sequences. Fire PRA components are listed in the NMP1.rr CAFTA database. Those components requiring cable selection are listed with a Y in the CABLECODE column, N signifies no cable selection required, S indicates a subcomponent to a primary component. (F&O ID 6-1)	The check box that causes BAT-B11 to be included on Table C-1 was unchecked so that it no longer indicates that it is not included on the Fire PRA equipment list. The SSEL was reviewed and any components that were missing from Table C-1 were added (including BV-210-33). The codes for excluding components and the listing of subcomponents are being handled under finding 1-12. Additionally, a discussion was added to Section 2.1 of the ES notebook that describes how components from the internal events PRA were coded in the CAFTA NMP1.rr database and documented in the tables in the ES notebook.

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SR	Topic	Status	Finding	Disposition
DA-D3	UNC – Uncertainty & Sensitivity	Closed	This SR is evaluated for CF-A2. The uncertainty values for the cable failure data were generated based on NUREG/CR-6850 guidance. However the uncertainty values have not been used in the final quantification. (F&O ID 2-16)	The model has been quantified using estimated uncertainty distributions for the fire-induced circuit failure probabilities, human error probabilities, other random failure probabilities, and the fire scenario frequencies. This is documented in Section 4.2 of the UNC notebook.
QU-B9	FQ – Fire Quantification	Closed	The reported CDF and LERF values in the Fire PRA Notebook Fire Risk Quantification report disagrees with the combined cutset file provided to the peer review team. Several issues were identified. The most significant involves the use of basic events set to 1.0 (versus true) in the FRANX cutsets. Several events were set to true in the combined cutset run, followed by cutset subsuming, resulting in almost a factor of 2 reduction in the risk values. Additionally, there is an error in the FRANX code which causes false cutsets to be developed in some cases, when the human error event replacement occurs in FRANX. (F&O ID 2-34)	FRANX is configured to execute a set of QRecover commands as part of the processing of each scenario's cut set file. The QRecover input file sets all probabilities that are equal to 1 in each cut set to TRUE, and then subsumes the nonminimal cut sets and compresses the subsumed cut sets. This procedure ensures that the FRANX results are free of nonminimal cut sets as documented in Section 6 of the FQ notebook. Bogus cut sets may be generated when FRANX moves all of the basic events out from under an OR gate to the OR gate directly above. CAFTA interprets the empty OR gate as a basic event and assigns a value if one is available in the CAFTA database. The FRANX software error that creates bogus cut sets under certain circumstances is overcome by using QRecover to identify the associated bogus basic events and set them to FALSE. This procedure eliminates the bogus cut sets as documented in Section 6 of the FQ notebook.
FQ-D1	FQ – Fire Quantification	Closed	LERF (2.5E-4/yr) is high relative to CDF (3.8E-4/yr). Based on a review of the top scenarios, it appears that failure of emergency condenser isolation is a significant factor for this outcome. Refinements to the FPRA to improve realism should include considerations for LERF. An example of the F&Os that were identified for CDF should also be considered with respect to LERF model refinements: 4-1, 2-5, 1-13, 2-7, 1-9, 4-17, and 2-32. (F&O ID 4-24)	Refinements to the FPRA to improve realism with respect to LERF have been implemented. During the cut set review that was performed December 12 through 16, 2011, LERF results were examined and model refinements were made to reduce LERF as documented in Section 6.12 of the FQ notebook. For example, the operator action to close MSIVs by locally venting air was reduced by refinement of the detailed human reliability analysis. Further improvements were subsequently implemented. The ratio of CDF to LERF increased from 1.5 at the time of the peer review to approximately 7 currently as a result of the improvements.

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SR	Topic	Status	Finding	Disposition
CS-B1	CS – Cable Selection	Closed	The Nine Mile Point Unit 1 NFPA 805 Coordination Study found breaker uncoordination for several 600 VAC, 208-120 VAC, and 240 VAC power supplies. The FPRA modeling, however, assumes proper coordination for all power supplies and therefore does not represent the as-built, as-operated plant. A sensitivity evaluation was performed and is documented in the Uncertainty Notebook. NMP1 plans to provide proper coordination through plant modifications for uncoordinated power supplies that are found to be risk-significant. (F&O ID 4-3)	A strategy was developed to determine breakers that may need to be coordinated based on the risk impact. The list of uncoordinated breakers identified in the breaker coordination study was used as a starting point. A number of iterations were performed using the FPRA model to determine the breakers that were risk significant. The iterations used a partial population of breakers that were considered coordinated in the model. Sensitivity studies were run to determine which breakers were risk significant and would be required to be coordinated. Based on the sensitivity runs performed in the FPRA model it was identified that two (2) breakers were deemed to be risk significant. The two (2) breakers are the tie-breakers that connect Powerboard 16A section with 16B section and 17A section with 17B section. The issue with these tie breakers is that they are not coordinated with the breakers that supply Powerboard 167 and 1671. The supply breakers for Powerboards 167 and 1671 are coordinated with the respective supply breaker to Powerboards 16 and 17 but are not coordinated with the tie breakers between the A and B sections of Powerboards 16 and 17. The cable impacts of uncoordinated breakers identified by AREVA took a conservative approach and did not take into account actual breaker position. A conservative approach was taken and all breaker impacts were used in the FPRA model to run the initial sensitivity case. When modeling the impacts of the tie breakers as coordinated the remaining uncoordinated breakers are low risk. To achieve insignificant risk values for breaker coordination the model will assume that breakers R1042 and R1052 are coordinated with the supply breakers for Powerboards 167 and 1671. This was done by removing the cable impacts in the model for the breakers associated with Powerboards 167 and 1671. By doing this the model assumed coordination with the tie breakers.

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SR	Topic	Status	Finding	Disposition
				Based on one-line drawings and the system description documents the tie breakers R1042/R1052 are normally open with the breakers R1041/R1051 feeding the 'A' sections of PB16/17 and breakers R1043/R1053 feeding the 'B' section of PB16/17. Being normally open, the closing of Powerboard 16 and 17 intertie breakers is administratively controlled by NMP Operating procedure N1-OP-30 '4.16 KV 600V and 480V House Service.' There are no automatic transfer functions for breakers R1042 and R1052. Therefore, since the tie breakers are normally open and are controlled administratively there is no reason to replace the breakers to improve coordination. Based on the FPRA model impacts and this evaluation no modification is required for breaker coordination.
MU-B1	FQ – Fire Quantification	Closed	Future Modifications are credited in the FPRA model in support of the NFPA-805 application. This does not reflect the standard requirement for the FPRA to reflect the current as-built as operated plant. When modifications are complete and final installation and operation details become available the FPRA should be updated and changes incorporated. (F&O ID 5-13)	As documented in Section 5 of the FQ notebook, plant modifications that CENG has committed to remain in the base model. The sensitivities of the results to the modifications are documented in Section 4.3 of the UNC notebook.
QU-D2	FQ – Fire Quantification	Closed	Notes from the cutset reviews were retained for the purpose of documentation that cutset reviews had been performed for the Peer Review but are not documented as part of the FPRA analysis. (F&O ID 5-18)	Final cut set reviews are documented in the FQ notebook: A summary of model changes that were recommended during the cut set reviews is provided by reference in Section 6.12; Appendix A reproduces summary notes from the cut set review meetings. The summary of final cutset reviews supports the discussion of risk contributions of fire scenarios in Section 5 of the FQ notebook.

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SR	Topic	Status	Finding	Disposition
QU-D3	FQ – Fire Quantification	Closed	The FPRA analysis and section 6.10 do not explicitly describe that issues with flag settings, mutually exclusive events and recoveries were looked for in the cutset reviews. However, cutset reviews performed by NMP appeared to have been performed in a manner that would identify issues with flag settings, mutually exclusive events and recoveries. No instances were found by NMP, and Peer Review of selected sequences did not identify any issues. Additionally, utilization of the 'Ones' method of quantification to achieve cutsets results which can be reviewed was discussed with NMP as having been performed. This is suggested to be included in the 6.10 discussion also. (F&O ID 5-20)	Section 6 of the FQ notebook has been revised to include discussions of flag setting, mutually exclusive events, and recoveries. Limited use of "Ones" quantifications has been made as documented in Section 6.12 of the FQ notebook. The use of "Ones" runs as a tool for cut set review is limited by the fact that the Ones runs were generally not successful for high CCDP cut sets.
QU-D5	FQ – Fire Quantification	Closed	FQ Section 6.10 discusses cutset reviews, but does not address review of non-significant cutsets. The cutset review notes provided by NMP did not discuss review of non-significant cutsets, and were focused on highest risk contributors. (F&O ID 5-21)	As required by QU-D5, a sampling of non-significant accident cutsets or sequences were reviewed to determine that they are reasonable and have physical meaning. This is documented in Section 6.12 of the FQ notebook.
QU-D6	FQ – Fire Quantification	Closed	SSCs and Operator actions that contribute to initiating event frequencies are not identified. Mapping of SSCs and operator actions contributing to Initiating events was not performed in ES-A1 (see F&O 1-10). (F&O ID 5-22)	Consistent with the resolution of F&O 1-10, in which the rationale for the selection of two initiating events (%TT for scenarios expected to result in a fire induced turbine trip and %RPS for scenarios expected to result in a plant trip) was delineated, no explicit mapping of SSCs and operator actions to initiating events has been performed. Instead, it has been shown in the response to F&O 1-10 that the fire-induced failures of components and associated human actions that are both part of an initiator and the mitigating/support systems are properly modeled in the fault tree logic.

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SR	Topic	Status	Finding	Disposition
LE-F1	FQ – Fire Quantification	Closed	FQ Notebook Quantitative evaluation of LERF is performed and documented in Section 5.0. LERF contribution to Scenarios/Compartments is presented by SSC contribution, including LERF importance results, however information is not documented on the relative contribution to LERF by *plant damage state, or containment failure mode.* (F&O ID 5-23)	A summary assessment of the contributions from significant cut sets in the top 20 LERF fire scenarios to LERF by Level 2 end state and by Level 1 accident sequence corresponding to Large Early Release has been included in Section 5 of the FQ notebook.
QU-F1	FQ – Fire Quantification	Closed	The FQ Notebook describes the major inputs to quantification in a general manner, however, the processes followed to link the inputs prior to quantification are not clearly described, including software settings used to perform the quantification and achieve the stated quantification results. (F&O ID 5-24)	The process used to link quantification inputs and software, including all of the software configuration settings required to achieve the stated quantification results is documented in Section 6.11 of the FQ notebook.
QU-F3	FQ – Fire Quantification	Closed	Significant contributors to CDF LERF are described in *terms of Basic Events and operator actions for individual* top scenarios or compartments. Major accident sequences are described based on the contribution to the individual scenarios or compartments. A detailed description of significant accident sequences are not provided in relation to the overall FPRA results. (F&O ID 5-25)	The FQ reporting tables have been revised. *Specifically, the table of CDF scenario contributions* has been revised to include discussion of the top 95% of fire scenarios. Tables of end states and accident sequences from a one-top model that approximates the relative distributions of accident sequences from significant accident sequences have been added. The Fussel-Vesely importance measure is used for the determination of significant basic events (FV>0.005) in accordance with the definition in the Standard and tables of all significant basic events for CDF and LERF have been included. Tables of CDF and LERF for all fire scenarios have been included.

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SR	Topic	Status	Finding	Disposition
QU-F6	FQ – Fire Quantification	Closed	FQ Roadmap describes use of the definition from 1-2.2 of the standard. Reporting in tables in the FQ are not completely consistent with these definitions, and have no justification for alternative bounds. Significant basic event = FV greater than 0.005 or RAW greater than 2. Tables 6 and 7 do not appear to go to this detail. Significant cutset = 95% of CDF or 1% of the hazard group. Significant accident sequence = 95% of CDF or 1% of the hazard group. Top cutsets and accident sequences are not reported individually. Compartment contributions to a total of 95% of CDF/LERF are presented. Scenarios are reported only to 70%. Only top contributors to each compartment/scenario are discussed. (F&O ID 5-26)	The FQ reporting tables have been revised. Specifically, the table of CDF scenario contributions has been revised to include discussion of the top 95% of fire scenarios. Tables of end states and accident sequences from a one-top model that approximates the relative distributions of accident sequences from significant accident sequences have been added. The Fussel-Vesely importance measure is used for the determination of significant basic events (FV>0.005) in accordance with the definition in the Standard and tables of all significant basic events for CDF and LERF have been included. Tables of CDF and LERF for all fire scenarios have been included.
CF-B1	CF – Circuit Failures	Closed	The application of CF failure probabilities and the CF duration probabilities is applied in two FRANX tables including the FIREINITIATORHRA table and the FIREALTEREDBE table. The implementation is confusing and difficult to trace, since there is no single location or approach used to include the CF probabilities (and duration probabilities) in the FPRA results. (F&O ID 1-23)	A description of the approach for implementing the CF probabilities using FRANX has been added to the FQ notebook.
QU-B1	FQ – Fire Quantification	Closed	Quantification of the FPRA uses the EPRI FRANX code, which has been demonstrated to general appropriate results. Limitations of the code are discussed in the quantification notebook, Section 6.9. However, the limitations of FTREX and FRANX are not discussed in the report. (F&O ID 1-25)	Discussions of the limitations of FRANX, FTREX, CAFTA, and QRecover have been added to Section 6.11 of the FQ notebook.

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SR	Topic	Status	Finding	Disposition
QU-B4	FQ – Fire Quantification	Closed	The NMP1 PRA model used the upper bound approximation in quantification, using CAFTA and FTREX. However, there are numerous CCDP values at 0.1 or above. As a result, a recommendation is provided to use ACUBE or similar in estimating exact solutions for these cutset solutions. (F&O ID 1-26)	This suggestion has value in evaluating the extent of conservatism in the computation of cut-set probabilities. The extent of conservatism in the CDF and LERF results has been evaluated using ACUBE and is presented and discussed in Section 6.13 of the FQ notebook.
QU-F6	FQ – Fire Quantification	Closed	FQ Roadmap describes use of the definition from 1-2.2 of the standard. However, the formal definition should be included in the FQ report. (F&O ID 2-35)	The formal definitions of significant cut sets, basic events, and accident sequences have been included in the FQ report in Section 5.
QU-D1	FQ – Fire Quantification	Closed	The provided documentation of cutsets reviewed implies that scenarios with CCDP=1.0 were reviewed when they contributed significantly to fire CDF. Review of all CCDP 1.0 cutsets is suggested to ensure correct logic, modeling consistency, and operational consistency. Low Probability, high consequence scenarios can greatly affect the plant when the underlying assumptions causing the low CDF are later changed or proven incorrect. General example: Full room burnout CCDP of 1.0 with low frequency of occurrence from only one or two fire scenarios. The final CDF may be small, however large impacts could be observed due to changes in the state of fire modeling knowledge, failures of the plant to maintain electrical cabinet doors in a closed position, or future additions of scenarios or cables to the room. (F&O ID 5-19)	Cut set reviews are documented in Section 6.12 of the FQ notebook. Scenarios with CCDP=1 have been subjected to cut-set review.

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SR	Topic	Status	Finding	Disposition
FSS-D7	FSS – Fire Scenario Selection Detailed Fire Modeling	Closed	The unavailability of fire protection system credited in PRA was not based on the review of NMP1 plant historical maintenance / availability data. The NUREG/CR-6850 (page P-6) generic data does not include maintenance unavailability. (F&O ID 2-30)	Condition reports for a period of 10 years were evaluated for determining any outlier behavior in the availability of the credited fire protection features, which include smoke detectors, Halon system, and automatic sprinkler systems. The CR search for Halon system only received 14 hits (CRs). Of the 4 that may indicate operability concerns, only two indicated inoperability in the CR. The CR did not offer specific information on the duration of the inoperable condition; however, communications with Fire Protection engineering suggested that the system has not been down for extended periods of time, and therefore the generic value should bound the actual availability of the system. For the unavailability of fire protection water systems, the results suggest 145.05 hours over 10 years. All unavailability for reasons other than one fire pump unavailable (electric or diesel) were included. If both pumps were unavailable then the hours were included as well. The resulting value is 145/(87600 hrs in 10 years) = 1.6E-3. Therefore, the generic value of 0.02 is justified. In summary, the review of the CRs suggests no outlier behavior; therefore the generic failure probabilities in NUREG/CR-6850 are appropriate. The details are presented in Section 6.1.3.3 of the Detailed Fire Modeling report.

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SR	Topic	Status	Finding	Disposition
FSS-G5	FSS – Fire Scenario Selection Multi-Compartment	Closed	During review of the Fire Modeling Database, Multi-Compartment Analysis it was noted that several duplicate fire wall barrier element records exist which may result in overestimation of the fire barrier failure probability. Some examples of duplicate records are listed as follows: B1A to B2A Two records for Fire Wall recorded in the FMDB C1 to C2 Two records for Fire Wall recorded in the FMDB *D1A to D1B Two records for Fire Wall recorded in the* FMDB D1A to D1C Two records for Fire Wall recorded in the FMDB D1B to D1A Two records for Fire Wall recorded in the FMDB D1B to D1C Two records for Fire Wall recorded in the FMDB D1C to D1A Two records for Fire Wall recorded in the FMDB D1C to D1B Two records for Fire Wall recorded in the FMDB D2A to D2B Two records for Fire Wall recorded in the FMDB D2A to D2D Two records for Fire Wall recorded in the FMDB D2D to D2C Two records for Fire Wall recorded in the FMDB EXT to T3B Two records for Fire Wall recorded in the FMDB D2D to OG3 Two records for Fire Wall recorded in the FMDB OG3 to D2D Two records for Fire Wall recorded in the FMDB R2B to R1B Two records for Fire Wall recorded in the FMDB (F&O ID 3-10)	All of the multicompartment pairs in FMDB were reviewed and duplicate records were deleted. The multi-compartment report, Table 3, was updated to reflect corrections. No changes were needed in the Detailed Fire Modeling notebook.

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SR	Topic	Status	Finding	Disposition
FSS-G4	FSS – Fire Scenario Selection Multi-Compartment	Closed	Calculation of the fire barrier failure probability includes failure probabilities for the wall, fire doors, and fire dampers. However no accounting is made for the failure probability of fire penetration seals. Characterization of the barrier failure probability should include a failure probability for seals where installed. As is the practice for doors and dampers the failure probability should be established for each barrier evaluated. (F&O ID 3-8)	The multi-compartment analysis was revised to assume one penetration seal per wall as there is no clear guidance in NUREG/CR-6850 for the use of the penetration seal failure probabilities. Consequently, for each wall, the Fire PRA conservatively postulates the failure of the wall and a penetration seal, with a combined failure probability of 1.2E-03 + 1.2E-03 = 2.4E-03. A screening process was conducted for determining scenarios where the penetration seal failure probability can be an important factor in the quantification. For those walls, a sensitivity analysis has been conducted where the penetration seals have been counted and the failure probability has been increased accordingly.
FSS-C7	FSS – Fire Scenario Selection	Closed	The fire PRA assumes that the fire suppression water supply can supply both core injection and suppression at the same time. The basis for this assumption is not documented. It also noted that currently the NFPA 805 project is evaluating plant modifications and procedure changes associated with the potential simultaneous demand of fire water for suppression and core injection. Consequently, the FPRA assumes that the system will be capable of handling both demands. This F&O is generated to confirm the fire PRA assumption with the evaluation results once plant modification is done. A sensitivity study case has been constructed to evaluate the significance of this fire PRA assumption. The delta CDF is >200%. Therefore this is a risk-significant assumption in NMP1 fire PRA. (F&O ID 6-6)	Proposed modification #4 in the S-1 Table of Attachment S will open the Unit 2 firewater crosstie when needed. Opening this crosstie will ensure adequate water supply is available to supply both core injection and fire suppression demands simultaneously.

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SR	Topic	Status	Finding	Disposition
PRM-A1	PRM – Plant Response Modeling	Closed	Queries have been built in FRANX database to identify potential inconsistencies: 159 records are associated with the 'non-existing' raceways (e.g., 1C0, 1S2, 1S-2618-2, etc.). Since there is no record in 'Raceway' table and no information exists for raceways in the 'ZONETAG' table, the reviewer cannot make any conclusion from these records. Site response stated two reasons for this observation: 1) There are some 'old raceway id's' in the fire modeling database that have not been updated or deleted. They generate the no-mapping problem identified but has no impact by not having any mapping to basic events, which will be deleted. 2) In C2 fire zone, the panel ID has been listed as a raceway (so that they act as targets). This works when the panel ID is part of the route string of the cable-most likely a cable end point. When researching this question it was found that some of the end points do not match the route id in the zone tag table because our walkdown notes indicate that the panel has been removed from the zone and therefore, targets are not found in the main target list. Need to remove these non-existing cabinets from the database. (F&O ID 2-14)	The non-existing raceways and cabinets were deleted from the database.
FSS-A6	FSS – Fire Scenario Selection Control Room Abandonment	Closed	The calculation of MCB fire scenario frequencies was documented in MCR FSS notebook section 1.5.1. Figure 2 shows the Evaluation Logic for Main Control Room Panel Fires. An example is shown in Figure 3. However, the sequence probabilities for sequence #1 are not consistent with the values used in Table 4. Response from NMP confirmed that the values in Figure 3 are not accurate. Thus this is a documentation issue. On the other hand, the sequence probabilities associated with the MFP scenarios appear to be too conservative for sequences 5 & 6. In Table 4, sequence #1's sequence probability is assigned to be 0.63, which appears to be correct. Sequences 5 & 6 currently add up to 100% (0.87 + 0.13), which should add up to 0.37. However, the total contribution of MPFs is negligible to the total fire PRA CDF/LERF. Thus this conservatism has no significant impact to the final results. (F&O ID 2-23)	Figure 3 in the Main Control Room notebook was corrected to reflect the actual sequence probability of Scenario 1 (1.3E-03). Scenarios 5 and 6 are conservative since the fire does not propagate beyond the panel and therefore, there should be no control room abandonment. The Sequences 5 & 6 (Control Room Abandonment) currently add up to 100% (0.87 + 0.13), which is correct (should not add up to 0.33).

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SR	Topic	Status	Finding	Disposition
FSS-C6	FSS – Fire Scenario Selection Detailed Fire Modeling	Closed	NMP1 Fire PRA Notebook: Detailed Fire Modeling, Appendix G Assumption 7 implies that fire modeling calculations assume damage when exposure environment exceeds damage thresholds. Section 6.1.3.1 states that each damage state is defined by a time and a set of targets damaged within that time. Fire modeling is used to determine the targets affected in each damage state and the associated time at which this occurs. It appears that the thermal response time is not evaluated. The assumption on target damage with respect to exposure environment is not documented explicitly. (F&O ID 2-24)	Section 6.1.3.1 of the Detailed Fire Modeling notebook was revised to state "Target damage occurs when the exposure environment exceeds the damage threshold (i.e., the thermal response time of the targets is not calculated)." This course of action is conservative.
FSS-F1	FSS – Fire Scenario Selection – Structural Steel	Closed	NMP1 FIRE PRA Notebook Fire Scenario Selection and Analysis (FSS) Structural Steel, N1-FSS-F002, Tables 7 and 8 list the structural steel-related fire scenarios, which are then included in the FRANX models. However, the calculated fire scenario frequencies in Table 8, which have been confirmed to be correctly calculated, are not consistent with the results in Table 7. While FRANX fire scenarios use Table 7 results, inconsistencies exist between the FRANX quantification and N1-FSS-F002. It is noted that almost all the 'changed' frequencies in Table 8 are lower than the ones used in FRANX, the current results are conservative. Because the total risk contribution from structural steel scenarios are low, the risk impact of this finding is not significant. (F&O ID 2-27)	Table 7 and FRANX structural steel-related fire scenarios were updated to be consistent with Table 8 of NMP1 notebook N1-FSS-F002.

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SR	Topic	Status	Finding	Disposition
FSS-C3	FSS – Fire Scenario Selection Detailed Fire Modeling	Closed	As discussed in FSS-C2, fire modeling of risk significant ignition sources utilizes time dependent HRR profiles for each of the fire types evaluated. This evaluation is performed using a fire growth and peak HRR stages. Incipient and decay stages are not considered in the NMP1 FPRA, as documented in Appendix G of the *Detailed Fire Modeling Notebook*. *Fire burnout is* considered based on guidance provided in NUREG/CR-6850 and its supplement. Cable tray burnout is not considered when modeling fire growth; accordingly cable tray fires continue to grow in intensity. Consideration of *cable tray burnout would more realistically describe the* fire propagation and HRR generated. (F&O ID 2-29)	The HRR generated by considering cable tray burnout has been performed using the FLASH-CAT model described in NUREG/CR-7010. This model is based on a series of cable tray fire tests and predicts heat release rate as a function of time from a stack of cable trays considering propagation, flame spread and fuel *consumption. See Appendix A in the Detailed Fire* Modeling Notebook (Section A.8) for a detailed technical description of the model application. Therefore, the current approach meets the intent of the F&O.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
FSS-B1	FSS – Fire Scenario Selection Control Room Abandonment	Closed	Section 2.0 of the Main Control Room Abandonment Study identifies the abandonment criteria as being taken from NUREG/CR-6850. The criteria identified are as follows: 1. The hot gas is 6 ft or less above the floor and reaches a temperature of at least 200 °F (93 °C). 2. The heat flux to the control room floor is above 1 kW/m2. 3. The hot gas layer is 6 ft or less above the floor and has an optical density of equal to or greater than 3.0 per meter. Such optical density is assumed to prevent operators from seeing through the smoke. These criteria do not match that provided in NUREG/CR-6850. The referenced criteria indicate that abandonment should be assumed if: 1. (Paraphrased) The heat flux at 6' above the floor exceeds 1kW/m^2 (relative short exposure). Approximating radiation from the smoke layer, a smoke layer of around 95 deg. C (200 deg. F) could generate such heat flux. Items 1 and 2 above reference the same value. As presented these could be interpreted to be separate criteria. The parameter measured from abandonment should be heat flux exceeding 1 kW/m^2 at the 6' elevation. The reference to a 200 deg. F HGL is merely a comparative point of reference which should not be substituted for the acceptance criteria. NOTE: Discussion of the application of criterion no. 1 in Section 2.3.6 of the Abandonment Study indicates that the CFAST models were built to measure for 1 kW/m^2 heat flux at a height of 6 ft as required. 2. The smoke layer descends below 6' from the floor, *and the optical density of the smoke is less than 3.0 m^-1.* The units of the criteria referenced are incorrect. The density is measured in 1/m instead of m. NOTE: As presented in Section 2.5.1 of the Abandonment Study CFAST measured Optical Density is in terms of 1/m, accordingly the parameter measured is correct for the 6850 referenced parameter. (F&O ID 3-4)	The referenced criteria 1 and 2 in Section 2.0 of the Main Control Room Abandonment notebook were changed to (1) The hot gas layer temperature or room temperature reaches a temperature of at least 200° F (93° C) and (2) The heat flux at 6 ft above the control room floor is equal to or greater than 1 kW/m^2.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
FSS-B1	FSS – Fire Scenario Selection Main Control Room	Closed	During performance of the plant walkdowns the locations of Main Fire Panel 1 (PNL-MFP1) and Main Fire Panel 2 (PNL-MFP2) inside the Main Control Room was being verified and it was noted by Control Room Operators that PNL-MFP1 has been removed. The Fire Modeling Database lists both of these panels as being electrical cabinet initiators inside the Main Control Room. The scenarios for PNL-MFP1 need to be removed from the database and from the Main Control Room Risk Assessment. (F&O ID 3-5)	The scenarios for PNL-MFP1 were removed from the database and from the Main Control Room Risk Assessment. Removing the panel was also reflected in the electrical cabinet counting.
FSS-B2	FSS – Fire Scenario Selection Main Control Room	Closed	According to Section 1.6 of the MCR Fire Risk Assessment a geometric weighting factor of 0.004 is being applied to describe the fraction of Control Room floor space from which a transient fire may be expected to damage an electrical panel/MCB. This factor assumes placement of the transient near the evaluated panel which is a fraction of the total Control Room floor space, i.e., 4,020 sqft, where it could be placed. This value does not account for obstructed space, i.e., cabinet locations where transients cannot be located which should be eliminated for the total available space. (F&O ID 3-6)	The geometric weighting factor for the control room was recalculated to exclude the floor area containing cabinets. The MCR Risk Assessment notebook was revised and the FMDB was revised with the new value of 0.005.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
FSS-D6	FSS – Fire Scenario Selection Detailed Fire Modeling	Closed	Empirical models were not developed specifically for this FPRA. The only empirical models used are based in the guidance found in NURGE/CR-6850 and its supplement. Accordingly the required technical reference is provided by these documents and they are appropriately referenced in the NMP1 Detailed Fire Modeling FPRA Notebook. In one case an empirical model is used that may be inappropriate given the cable type at NMP1. The FPRA assumes all cable at NMP1 is unqualified; however the fire spread model identified in NUREG/CR-6850 Appendix R is used to describe fire propagation through cable tray stacks. According to NUREG/CR-6850, Supplement 1 and the Errata sheet for NUREG/CR-6850 this empirical model is not appropriate for use with thermoplastic cable and should only be applied to cable trays that are separated in accordance with Reg. Guide 1.75. Justification of the appropriateness of using this model at NMP1 needs to be established. (F&O ID 3-7)	The flash-cat model as described in F&O 2-29 is used for cable tray propagation using the thermoplastic HRR value of 265 kW/m^2. Therefore, the current approach meets the intent of the F&O.
FSS-C4	FSS – Fire Scenario Selection Detailed Fire Modeling	Closed	The area factors calculated for transients do not consider obstructions that may be located in the fire zone of interest. This is a conservative assumption. (F&O ID 6-8)	This is a conservative assumption because the area factors for transients assume a transient fire could occur anywhere, even though some areas are filled with equipment. Provided the risk is not a significant contributor, this conservative assumption is the best approach for NMP1 because a transient fire is postulated everywhere in the plant and it is easier to maintain in the program.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HRA-B3	HRA – Human Reliability Analysis	Closed	Several of the HFE HRA Calculator worksheets are missing detail or have inaccuracies: ZECA3_ECOPERATOR worksheet Table 3 indicates the action is the main control room. However, the action is local at the ECA valves. Note that the HEP listed from NUREG-1278 table 20-13 is for local actions, so this appears to be only a documentation issue. ZOR14_OROPERATOR is listed as a 12 minute execution time to remove flanges, install a spool and open several valves, under SBO conditions. The timing appears to be based on the permanent installation of the spool piece. However, the HFE description still includes the installation of the spool. This should be corrected. Note that the operator who walked the action down with us estimated the action (still installing the spool piece) would take up to 45 minutes. (F&O ID 1-18)	ZECA3_ECOPERATOR action location in the Execution Unrecovered screen of the HRA Calculator file (Table 3) has been changed to note that it is a local action rather than in the MCR. ZOR14_OROPERATOR includes the spool piece installation steps in the Execution Unrecovered screen of the HRA Calculator file (Table 3) to capture the history of the evaluation of this event, but has set the quantification of them to 0.0.
CF-A1	CF – Circuit Failure	Closed	Review of Local Manual Actions credited in the FPRA indicate a number of valves have a 92-18 potential problem (no 92-18 evaluation). As such, credit for the manual actions, given circuit failures for the valves, should not be included in the FPRA without a 92-18 evaluation. Examples of valves include: IV-31-07, 31-08 (ZIN01_INOPERATOR), and MOV 201-17 (ZCV02_CVOPERATOR). This is a random sampling, and more valves are likely not evaluated. (F&O ID 1-29)	The valves that were credited for local manual action following circuit failures were evaluated for the potential for 92-18 type unrecoverable failures. For valves IV-31-07, IV-31-08, IV-39-09R, and IV-39-10R the PRA model has been revised to allow recovery only for damage to cables that do not cause unrecoverable failures. Proposed modifications will allow operator action to recover the valves by redesign of circuits to eliminate cables that cause unrecoverable damage. For valves IV-38-13, FCV-80-118, FCV-93-71, FCV-93-74, BV-93-25, and BV-93-27 it was determined that no credit for recovery was currently taken in the model. For valves IV-38-01 and IV-38-02 it was determined that they were not susceptible to the 92-18 type failure. For valves IV-39-07R and IV-39-08R the model was revised to preclude recovery for damage to cable pairs that could cause unrecoverable damage. For valves IV-83.1-09, IV-83.1-11, IV-201-07, IV-201-09, IV-201-17, and IV-201-31 it was determined that only selected cables could cause the 92-18 type failure; therefore, the model was revised to preclude recovery for those cables only. This information was incorporated in the Plant Response Model (PRM) notebook Appendix C.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HRA-B4	HRA – Human Reliability Analysis	Closed	The undesired action identified by HRA-B3, 'Spurious High radiation Signal Causes Operators to Close MSIVs,' is modeled at gate CNMSIV-RAD in the NMP1 FPRA. The logic is appropriate. Failure of both annunciators 2-2 and 2-7 or their inputs are modeled to cause the undesired action, which is the manual closure of the MSIVs. It was difficult to locate the modeling for this undesired action in the FPRA model, and no description was provided of the fault tree logic. (F&O ID 4-11)	A discussion of this subtree and the undesired operator action in response to spurious instrumentation that it represents is now included in the Fire HRA Notebook Section 3.2 Evaluation under "Events Caused by Incorrect Human Responses Associated with Identified Actions".
HRA-D1	HRA – Human Reliability Analysis	Closed	Section 5.0 of the HRA Notebook discusses recovery actions identified for the FPRA. A few documentation considerations were noted: The recovery actions are as follows: - ZOD1EF33ODOPRATR is not modeled in the FPRA, although it is listed in Section 5.0 as a recovery action. - No discussion is provided on how the FPRA recoveries, ZSD03F31SDOPRATR and ZSD04F31SDOPRATR, were identified. Based on the review of importance results, these two recoveries are not risk significant, so there is not apparent risk basis for selecting them. - Recovery actions modeled in the internal events PRA that are carried over to the fire PRA are not documented in Section 5. (F&O ID 4-12)	Table 5-1 in Section 5.0 on Recovery Actions in the Fire HRA Notebook has been revised to enhance the discussion of recovery events, including the rationale for their inclusion. The recovery events in this table include events added to both the internal events and fire PRA models for recovery.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HRA-D1	HRA – Human Reliability Analysis	Closed	The peer review examined the FPRA results to analyze the risk importance of recovery type human failure events set to failed. Four of these HFEs were found to be risk significant, and they appear potentially feasible for FPRA scenarios: - ZDG02F11DGOPRATR, Screening HEP: Operators Fail to Cross-Tie DG Fuel Oil, FV = 6.08E-03 - ZDG03F31DGOPRATR, Detailed HEP (no procedure): Operators Fail to Open Doors for EDG Room Cooling, FV = 1.14E-01; note there is no procedure for this action, however, high risk significance may prompt consideration for procedure development - ZDPC2F23DCOPRATR, Scoping HEP for MCRA: Operator Fails to Align Portable Charger (OLS=F): FV = 2.27E-02 - ZZDG8F11REDG28HR, Screening HEP: Recovery of 1 of 2 EDGs in 8 Hours (mod for fire), FV = 2.07E-01 (F&O ID 4-13)	The ZDG02 diesel fuel transfer pump cross-tie action was reviewed with operators but was not credited, since this has not been a part of the plant's design. For now this event is a placeholder for any future modifications. Regarding ZDG03, the HEP was set to 1.0 not only due to the uncertainty of the operator action but also because there are no calculations that demonstrate that opening the diesel room doors would be adequate to keep the temperature down. ZDPC2 is for aligning the portable charger for the case when the operator previously fails to load-shed DC. The HEP is set to 1.0 because the time gets tight (8 hours if they shed, but only 4 if they don't) and the action itself takes 3 hours. Also, the action is in a damage repair procedure that is less well structured and trained upon. For the 8 hour case, this might not be significant, but for the 4 hour case, such uncertainties are potentially important. ZZDG8 is a historical data-based value for diesel recovery, not a human error probability. Risk significant HFEs including those set to fail were all reviewed during 12/12 - 16/2011 cutset review and refinements were made as appropriate. Section 4.2.3 of the Fire HRA Notebook and particularly Table 4-6 has been updated to reflect these changes.
HR-H3	HRA – Human Reliability Analysis	Closed	The NMP1 FPRA HEP dependency analysis utilizes the dependent HFE combinations identified for internal events. The recovery actions identified for fire, ZSD03F31SDOPRATR and ZSD04F31SDOPRATR therefore are not included in the dependency evaluation. (F&O ID 4-14)	Both ZSD03 and ZSD04 are associated in the Operator.caf model with the dependent group event ZQDHR_DEPOPERATO Common Failure to Align Containment Heat Removal, which correlates them to 18 other events by virtue of how the model has been set up. Therefore, the dependencies for these recovery actions have been addressed in the model consistent with the dependency analysis approach used for the internal events and fire HFEs.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HRA-E1	HRA – Human Reliability Analysis	Closed	It is difficult to correlate operator actions documented in the HRA calculator and those documented in the HRA Notebook and FPRA model. The HRA calculator utilizes nomenclature from the internal events PRA, whereas the FPRA has unique naming: '31' events are in the FPRA base calculator, '32' events are in the in MCR HRA calculator. Additionally, scoping HFEs (the '21' and '22' and '23' events) are documente4d in the HRA Notebook. (F&O ID 4-16)	Section 4.3.1 of the Fire HRA Notebook, HEP Values for FPRA Model, has been revised to provide a better explanation of the coding scheme used for HFE application to different quantification cases. In addition, Section 4.2.2 on scoping analysis provides the scoping values and Section 4.2.3 on detailed analysis provides the detailed analyses for the base case in section 4.2.3.2 and the In-MCR case in Section 4.2.3.3. MCR Abandonment values are provided in section 4.2.4. Summary table 4-15, HEP input to Quantification, shows which values were calculated for which cases. Separate appendices and HRA Calculator files are included for the detailed analysis for the three cases: base case, In-MCR and MCR abandonment.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HR-G1	HRA – Human Reliability Analysis	Closed	The following risk-significant HFEs set to failed. These actions have received scoping analysis. Since they are risk-significant, it is worthwhile to consider detailed analyses, if the actions are feasible: - ZECA3F23ECOPRATR, Fire HEP MCRA: Operators Fail to Locally Open IV-39-07, 09; FV = 7.08E-03, time margin 30% - ZECI1F23ECOPRATR, Fire HEP MCRA: Operators Fail to Close Manual Valves 05-31, 32, FV = 7.03E-03, time margin 32% - ZECI2F23ECOPRATR, Fire MCRA: Operators Fail to Manually Close FCV-39-15, 16 with Local HWO, FV = 7.03E-03 - ZHRA4F23HRAOPRAT, Fire HEP MCRA: Op Fails to Use East/ West Instrument Rooms (SBO), FV = 2.01E-02 - ZOD10F23ODOPRATR, Fire HEP MCRA: Op Fail to Open ERVs from RB Given Fire Induced ERV Circuit Failure, FV = 4.32E-02 - ZOLS1F23OLSOPRAT, Fire HEP MCRA: Op Fails to Implement DC Load Shedding FV = 2.20E-02 - ZOR12F23OROPRATR, Fire HEP MCRA: Op Fails to Align Fire Water FV = 3.71E-02 - ZOR13F23OROPRATR, Fire HEP MCRA: Op Fails to Align Fire Water (SBO 1 hour) FV = 1.99E-02 - ZDPC2F23DCOPRATR, Fire HEP MCRA: Op Fails to Use East/ West Instrument Rooms (SBO), FV = 1.34E-2 - ZIS02F23MSIVLOC1, Fire HEP MCRA: Operators Fail to Locally Vent Air from MSIVs, FV (LERF) = 1.9E-2 - ZRX03F21INJCTRCV, Scoping fire HEP: operator fails to align alternate inj sources in level 2 (conditional), FV (LERF) = 1.8E-2 (F&O ID 4-17)	Detailed analysis has now been performed for all the listed events except ZDPC2 and ZRX03. ZDPC2 is for aligning the portable charger for the case when the operator previously fails to load-shed DC. The HEP is set to 1.0 because the time gets tight (8 hours if they shed, but only 4 if they don't) and the action itself takes 3 hours. Also, the action is in a damage repair procedure that is less well structured and trained upon. For the 8 hour case, this might not be significant, but for the 4 hour case, such uncertainties are potentially important. ZRX03 is a conditional Level 2 event that was set to 1.0 in the Internal Events PRA and that value was carried over to the fire PRA consistent with the EPRI/NRC Fire HRA Guidelines, NUREG-1921. There is not sufficient justification available to reduce the HEP. The values resulting from the detailed analysis of the other HFEs identified by the peer review team were provided as input to the Fire PRA model and the Fire HRA Notebook. Risk significant HFEs including those set to fail were all reviewed during 12/12 - 16/2011 cutset review and refinements were made as appropriate. Section 4.2.3 of the Fire HRA Notebook and particularly Table 4-6 has been updated to reflect these changes.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HRA-C1	HRA – Human Reliability Analysis	Closed	The FPRA HFE dependency analysis relied on the dependent combinations identified by the internal events PRA. The FPRA HFE dependency analysis did not review the PRA results for combinations relevant to fire scenarios (which may be different combinations from internal events due to differences in risk significance). (F&O ID 4-18)	Dependency groups were re-evaluated for fire in several ways; 1) By adding new fire HFEs to the existing dependency groups (as identified in the Operator.caf fault tree), 2) By re-evaluating the THERP dependency formulas based on the dependency factors (such as crew, location, timing) for the HFEs in these re-defined dependency groups, and 3) By reviewing cutsets with HEPs set to 0.1 (as was done for the original internal events dependency analysis) and set to 0.9 (as recommended by the peer review committee). The 0.9 cutsets were not very informative, but the 0.1 cutsets allowed the analyst to determine which HFEs were the predominant initial HFE in the sequence of multiple HFEs in the same cutset. A count was made of the number of times a given HFE appeared as the initial HFE in the sequence of HFEs in the same dependency group. The HFE that appeared the most times was used as the basis for the group's THERP dependency formula and the resulting HEP was used in the fire PRA model. The THERP dependency assignment trees were used to re-evaluate dependency between the actions given fire influences on the factors and the dependency assessment was increased from Low to Moderate for one of the dependency groups. The HFEs used for the THERP dependency calculations were changed in three cases. New dependency values were calculated on this basis and were provided as input to the fire PRA model. Section 4.2.5 of the Fire HRA Notebook has been updated to document these assessments, calculations and results.
HR-G7	HRA – Human Reliability Analysis	Closed	The degree of dependence between joint HFEs in the same cutset in the FPRA were carried over from the internal events PRA, without consideration of relevant fire-related effects. (F&O ID 4-19)	See response to SR HRA-C1

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HRA-C1	HRA – Human Reliability Analysis	Closed	The following base model HFEs are documented in the HRA notebook as set to failure, but the FireInitiatorHRA File instead assigns an HEP value: ZDPC1F31DCOPRATR, and ZDPC2F31DCOPRATR, Response to a question posed to NMP1 indicated that the first two HFEs no longer will be set to failed. These events were recently amended and quantified to reflect procedure changes and a modification (as documented in Appendix F) to physically stage the portable charger such that these HFEs can now be credited. (F&O ID 4-21)	ZDPC1F31DCOPRATR and ZDPC2F31DCOPRATR have been re-evaluated since the peer review, including operator interviews to discuss the feasibility of the actions. On this basis, ZDPC1 has been credited and quantified with an HEP, but ZDPC2 has again been set to 1.0. The Fire HRA Notebook and associated quantification files have been amended to reflect these values.
HR-G7	HRA – Human Reliability Analysis	Closed	The calculation for the probability of dependent HEPs utilizes the THERP (NUREG/CR-1278) methodology for calculating dependencies, but doesn't follow the methodology. The THERP calculation method for multiple HEPs is exemplified as follows: If the dependent HFE combination is P(A) and P(B), where P(A) = 5E-3 and P(B) = 2E-3 where the dependence is moderate, THERP calculates the *dependent HFE as:* P(A *and* B:MD) = 5E-3 * (1+6*2E-3) / 7 = 7E-4. If a third HEP appears in the cutset with an HEP of 2E-2 with low dependence (P(C)), a similar approach is taken, this time substituting the value for P(A and B) for P(A) and P(C) for P(B) in the equation for low dependence: P(A and B and C:LD) = P(A B) * (1+19*P(C))/20 = 7E-4 * (1 + 19*2E-2)/20 = 5E-5 Note the THERP directs performing the above calculations with the HFEs sorted by the time that they would occur. P(A) is the HEP that occurs first in the timeline, and P(B) occurs later, and so on. The above are examples, and note that the fire specific HEPs of course would be used in all of the equations. The NMP1 calculations don't follow the approach described above. (F&O ID 4-22)	The NMP1 HFE dependency strategy does not model cascading dependent HEPs. HFE dependency groups are created and it is assumed that once the second dependent failure occurs, all other HEPs also fail. The dependent group is treated in the model like a "beta" in equipment CCF and the gammas, deltas, etc. are not developed. The approach outlined in THERP (NUREG/CR-1278) is used for the second dependency and the remaining combinations are treated using the *ZQQQQ dependent event. The NMP1 HFE dependency* approach is conservative in this regard, since modeling dependency among groups of three or more causes significant uncertainties due to the variety in PSFs and timing and the subjective nature of the Low-Moderate-High category assignment process. In addition, most cutsets would have some intervening success, which creates another uncertainty regarding the actual dependency of greater than two operator failures within a cutset.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HR-E3	HRA – Human Reliability Analysis	Closed	HRA Notebook HRA Calculator Operator Interview insights documents some operator review, but does not describe further talk-throughs or detailed review of sequences with the operator. NMP was able to provide somewhat more detailed notes that clearly showed general review of actions for changes due to fire with ops personnel. (F&O ID 5-14)	Additional interviews with Operations and Training were conducted during the week of 12 December 2011 and the Fire HRA Notebook operator interview documentation was expanded into a new Appendix so that it is more consistent with the style and content of Appendix C of the internal events HRA notebook.
HR-E3	HRA – Human Reliability Analysis	Closed	HRA Notebook HRA Calculator Operator Interview insights and additionally provided notes documents operator reviews performed for the Fire HEPs, but does not talk through of each action in step by step detail with Ops and Training personnel to show that it is consistent with the plant observations and operator training procedures. (F&O ID 5-15)	Additional interviews with Operations and Training were conducted during the week of 12 December 2011 and the Fire HRA Notebook operator interview documentation was expanded into a new Appendix so that it is more consistent with the style and content of Appendix C of the internal events HRA notebook.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HR-A1	HRA – Human Reliability Analysis	Closed	N1-HRA-F001 Individual Greyed out items in Table 2-1 are not credited for fire based on being screened as not valid for fire scenarios. Based on discussion with NMP personnel they should all be 1.0 in FRANX, but instances were found that are not set to 1.0 FRANX. *Examples:* ZAI01_AIOPERATOR replaced with ZAI01F21AIOPRATR=0.05 ZDPC1_DCOPERATOR replaced with ZDPC1F31DCOPRATR=0.004 ZFF01_JMPRSINSTFL replaced with ZFF01F21JPRSINSTL = not in BE database ZFF02_DWVOPNFAIL replaced with ZFF02F21DWVOPNFL = 0.5 Some events could also not be found in the nmp1.caf *fault tree:* ZDC01_DCOPERATOR In the case of ZOD09F31ODOPRATR, it was not intended to assign an HRA value other than 0.99 (a 0.1 value was assigned in the FPRA model), so NMP1 will check in the FireInitiatorHRA File and amend as appropriate. (F&O ID 5-16)	ZAI01_AIOPERATOR was inadvertently grayed out in Table 2-1 instead of ZA101_A1OPERATOR; this has been corrected in the Fire HRA Notebook. ZDPC1_DCOPERATOR was incorrectly assigned an HEP less than 1.0; the 4 hour action time window makes this action infeasible; this has been corrected in the fire PRA model input and in the ES and Fire HRA Notebooks. ZFF01_JMPRSINSTFL and ZFF02_DWVOPNFAIL were inadvertently grayed out in Table 2-1; this has been corrected in the Fire HRA Notebook. ZDC01_OPERATOR is contained in the Operator.caf fault tree. ZOD09_ODOPERATOR is the event name in the CAFTA model; ZOD0931ODOPRATR is the associated designator that is used to assign HEP values using the FRANX tables. The HEP for ZOD09 was modified as a result of cutset reviews the week of 12/12-15/2011; this was also reflected in the updated Fire HRA Notebook. The entire Table 2-1 was reviewed and corrections were incorporated in the updated Fire HRA Notebook.
HR-B1	HRA – Human Reliability Analysis	Closed	It is unclear from the documentation of Table 5-1 how the Internal Events carryover HEPs were modified for the FPRA without directly comparing the two HEPs line by line. IE Fire HEPs do not appear inappropriately modeled in the HRA calculator based on the descriptions given in Table 5-1. Because the qualitative discussion does not directly translate to changes made in the HRA calculator, it is difficult to determine if all Fire Effects have been appropriately incorporated. (F&O ID 5-17)	Table 5-1 discusses the recovery actions modeled in the Fire PRA. The table was not very explicit in discussing what changes were made from the internal events HEPs to develop the fire HEPs. This has been clarified with a modified Table 5-1 in the updated Fire HRA Notebook that better summarizes the specific changes made to reflect fire effects (similar to what is shown in the detailed analysis fire HEP table 4-6).

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HR-G7	HRA – Human Reliability Analysis	Closed	Table 4-11 in the HRA Notebook shows the calculation of dependent HFEs and doesn't look complete. MCR (fire in MCR) actions aren't included, for example. (F&O ID 4-23)	The dependent HEP calculations were re-evaluated to include specific values for the case of fire in the MCR and these values were provided as input to the fire PRA model. The new calculation basis has been included in Table 4-13 in the updated fire HRA notebook.
IGN-A7	IGN – Fire Ignition Frequencies	Closed	The IGN notebook includes the following Batteries: Zone B2A – 3187, 3188, Zone B2B – 3193, 3194 Section 2.6 includes an assumption that "Figure 1 shows the two battery banks in Battery Room #11, which are counted as one battery set within the fire zone." A question was asked about this, and the response was to match up the assumption and the IGN counts. However, in the present counts, both sets of batteries in each room are still counted. (F&O ID 1-16)	The database containing the ignition source count was revised so that each battery set is counted correctly and consistently. Also, it was verified that batteries in other areas are consistently counted.
IGN-A7	IGN – Fire Ignition Frequencies	Closed	Space heaters were not counted as ignition sources at NMP-1. For example, there are numerous space heaters in the Turbine Building that do not appear in the FMDB either. Per NMP1 question response, Observations made during the walkdowns indicated that these space heaters are not in service during normal power operation. Further, it was assumed that the electric motors associated with these heaters were less than 5 hp. Plant specific fire events indicate a space heater fire occurred at NMP. Since space heaters are not limited by the 5 hp limit (mentioned in the question response), the space heaters should be counted as potential ignition sources, unless permanently removed from service. (F&O ID 1-4)	The technical specifications for the heater (Vendor Manual N1W200000HEATER002) indicate that they are less than 5 hp and therefore they are not included in the model as an ignition source. A review of the plant specific fire events suggests that a fire started by a space heater was non-challenging per the criteria in Appendix C of NUREG/CR-6850. The Ignition Frequency notebook (Section 2.6, Bin 14-Electric Motors) was revised to include this explanation.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
IGN-A7	IGN – Fire Ignition Frequencies	Closed	The Electric Motors in the ignition counts includes a total of 52 motors. Several refrigerators were included, which are typically not greater than 5 hp. Also; many of the motors appear to be less than 5 hp (multiple MOVs, H2, door motors, multiple crane motors). Counting of enclosed MOV motors and motors greater than 5HP is not consistent with the methodology described for BIN 14 in the IGN Analysis. (F&O ID 1-5)	The motors were reviewed and only those greater than 5 hp were included in ignition source counting. Ignition Frequency notebook was updated. The FMDB was updated. The changes in the ignition source database were automatically reflected in the scenario frequency and subsequent quantification. It should be noted, that some motors with unknown horse power were kept in the model.
IGN-A5	IGN – Fire Ignition Frequencies	Closed	According to Section 2.1 of the Fire Ignition Frequency Notebook the generic fire frequencies are updated on a reactor-year basis. However, the plant availability is not included anywhere in the frequency calculation or the fire quantification. (F&O ID 2-21)	The generic frequency values for all bins were multiplied by plant availability. Plant Availability (@AVAIL = 0.88) was developed and documented in the NMP1 Initiating Event (IE) Notebook. The ignition frequency notebook (Sections 2.1 and 2.7.3) was updated with discussion involving the use of the plant availability.
IGN-A7	IGN – Fire Ignition Frequencies	Closed	N1-IGN-F001 Counting Guidance: S2.6 Describes binning of ignition sources. Walkdowns data related to the counting of ignition sources appears in the FMDB comments for Task 1 Partitioning. The walkdown documentation describes compartments where no ignition source counts were performed due to inaccessibility (At least 9 areas affected). Based on response to Peer Review questions, no alternate method was used to identify ignition sources in these areas, including drawing review, pictures, operator interviews, etc. The rationale for this was that the areas would be screened anyway, and the additional counts in these compartments would dilute fire frequency in other areas of the plant. (F&O ID 5-12)	The IGN notebook was revised to indicate that drawings were used to identify ignition sources in those zones that are inaccessible to walkdowns due to high radiation.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
IGN-A7	IGN – Fire Ignition Frequencies	Closed	MOS factors: S2.7.2 describes MOS factors and the use of Table 6-3 from NUREG/CR-6850. Review of Appendix C fire compartment values for MOS indicates that values may have been misapplied, or have been applied without appropriate documentation of justification. MOS walkdown documentation is provided in the comments field of the partitioning task in the FMDB. Some of these are described as having 'conservatively low' factors applied; however this inappropriately weights other areas higher. Examples of possibly inappropriate factors include: AB 252 SAS Equipment Room - Occupancy is set to low (1) based on requiring a keycard, but generally SAS areas are continuously manned. Discussion with plant/security personnel should be used in assigning the factors. (General) in many cases, compartments were not observed during walkdowns due to inaccessibility. In these cases 'conservatively low' values were picked so as not to dilute areas where more FPRA cables are present. This results in undercounting risk in the unobserved areas and overcounting risk in other areas. Alternate methods of estimating these values should be utilized. (General) Equipment rooms/Equipment Areas - A number of these are binned Low (1), even though they may have large amounts of electrical or mechanical equipment. Areas with large amounts of equipment generally have large amounts of maintenance. (General) Records areas, library, TSC, File Rooms - For many of these, Storage is binned Low (1) meaning no combustible or flammable materials are stored. Would expect a records storage area to have a large amount of paper/combustibles. Occupancy is binned Low (1) – With the exception of locked areas, this may also be higher than low depending on occupancy by doc services personnel, plant personnel, etc. Example: Peer review walkdown of C2 found a number of ongoing maintenance projects, and a computer room with transients (fans, trash cans, computers, etc) and storage. Ranking of 1 is not appropriate.	Combustible storage influence factor for each fire zone was reviewed, compared against combustibles listed in the Unit 1 Combustible Loading Calculation, and revised where appropriate. All influence factors (storage, occupancy, and maintenance) were reviewed with NMP1 staff (1/13/12) to ensure that influence factors were assigned consistently throughout the plant. The database maintaining the influence factors was updated with corrected values. The IGN FPRA notebook was revised (Section 2.7.2) to explain that plant personnel reviewed the MOS influence factors.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
			(General) Shops - Electrical/Mechanical Shops have maintenance binned Medium (3). It is expected that although the areas would not have a large number of work orders performed on equipment belonging to the shop, there would be a lot of maintenance performed on equipment brought into the shop for repairs. (General) DG Building rooms, diesel Fire Pump room, TG Bay - These Rooms have no MOS factors above Medium (3) and are mostly Low (1). NUREG/CR-6850 suggests (p.22) 'If maintenance activity of a compartment includes liquid combustible/flammable material (e.g., diesel fuel, lubricating oil), the compartment should be rated as 'high.' RSSB Elev. 261 Radwaste Control Room - Area is Binned Low (1) for occupancy. Radwaste Control rooms are generally continuously manned. (General) - TB Laydown and large open floor areas, Rx Building General areas - Most of the TB open areas have storage factors of Medium (3) or Low (1). These areas generally include designated laydown areas and pre-outage staging areas, which would have higher storage factors. Example: Peer Review walkdowns of the area shows Area R4B had a laydown area containing numerous rubber hoses which appeared to be stored there. Storage ranking of 1 is not appropriate. Walkdowns in T3B also found that the side rooms (labs) off of the main hall had large amounts of storage and are appear to be frequently occupied. The treatment of storage of 3 appears appropriate for the overall area, but the treatment of the area should be discussed in detail in the IGN analysis since there is a great difference from room to room in the amount of storage. PW-18 DG Missile enclosure has occupancy & storage of 0. Even though it is inaccessible at power, transients may left in during an outage and remain at power. (General) EXT area, - This area is binned 1,1,1 for MOS. These areas have high potential for storage and generally have high occupancy. (F&O ID 5-2)	

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
IGN-A7	IGN – Fire Ignition Frequencies	Closed	During the walkdowns supporting the Peer Review assessment, the reviewers noted the following items relating to ignition sources and counting: Electrical Cabinets with designation DTB-XXX: These cabinets are all over the plant, generally on the side walls. Treatment of the cabinets is not specifically addressed in the IGN analysis; however, based on the counting guidance it would appear these should be included as BIN 15 items. The cabinets appear to have no ventilation and have sturdy latches; therefore they could likely be called closed cabinets with the proper justification. Examples: R3A - DTB-674; C1 - DTB-3, DTB-4; C2 - DTB-12 Ignition sources were noted that were not captured in the IGN analysis: T3A - HDS-UPS175A, HDS-UPS175B, HDS-UPS175D - Appeared to be newly installed cabinets, non-vented but not well sealed, some with many screw holes in the backs. C1 - Cabinet A-36 - 4' x 5' cabinet, non-vented, not well latched, no controls. C2 - FP Alarm Panel near stairwell at the entrance (could not be verified on the ignition source list) C2 - JB-RF Reflash panel C2 - SPDS Signal Isolator Cabinet - 2' x 3' with vents on front. Unlabeled Items - Items without MEL labels were difficult to pick out during the walkdowns, since in cases where there are multiple unlabeled items of the same type, it is difficult to determine which is the one assigned to the Ignition source ID number. If the numbers remain this way, it may be difficult for maintenance and update of the FPRA. (F&O ID 5-32)	Missing sources in T3A and C2 were added to the database. Text was added to the Ignition Notebook (Section 2.6, Bin 18) to explain that DTB-### cabinets are junction boxes and the frequency is apportioned based on cable loading. The Ignition Notebook (Section 2.6) was revised to indicate that equipment in areas inaccessible due to high radiation were identified using drawings.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
MU-E1	IGN – Fire Ignition Frequencies	Closed	Plant Walkdowns conducted by the Peer Review Team found items installed in the plant which have not been accounted for in the Fire PRA analysis. (See F&O 5-32) Programmatic controls requiring review and update of items which can affect the Fire PRA have not yet been implemented. Discussion with NMP1 Personnel indicated that a review of ECs is planned for in the future. (F&O ID 5-35)	Section 2.0 of the IGN notebook was revised to include an effective date, which is a date that reflects the state of the plant. The following statement was added "The fire ignition sources listed in this report reflect the existing state of the plant as of December 31, 2011, including all planned and funded modifications to the plant."
IGN-A7	IGN – Fire Ignition Frequencies	Closed	IGN notebook describes for BIN 16 that bus ducts utilize the approach in FAQ-0035. FAQ 07-0035 describes a method for counting segmented bus duct by either linear foot, or weld sections. If either of these methods were employed, a higher count related to the number of linear feet or sections of duct would be expected than the current total of 8 counts. Based on responses to peer review questions, the bus duct count of 8 is selected based on engineering judgment that an equal amount of the four bus ducts appear in the two turbine building areas T3B and T4B. Bus ducts were counted only inside the buildings, not in the EXT. It is assumed that the fires in the ducts to the transformers are captured by the frequency of transformer fires, which are postulated propagating to the turbine building in the FPRA. No documentation of this departure from counting methods is described in the FPRA documentation, nor is a basis provided in the form of bus duct drawings used to compare relative length of bus ducts. The lack of counting bus duct in EXT reduces the risk associated with the transformers in this area, and also increases the risk in the TB areas. (F&O ID 5-9)	Lengths of bus ducts in Turbine building and in EXT were estimated from drawings and entered into FMDB. The discussion for Bin 16 (Section 2.6) in the Ignition Frequency notebook was updated.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
IGN-A9	IGN – Fire Ignition Frequencies	Closed	Appendix E of the Fire PRA Notebook Ignition Frequency provides a summary of the transient and Fixed Ignition Frequencies for each fire compartment/zone. All zones include a transient frequency (not zero). Review of the detailed data sheets show that the transient summary includes junction boxes and self-ignited cable fires (typically considered fixed ignition frequencies (see suggestion). (F&O ID 1-1)	This is classified as a suggestion. This is an editorial comment with no impact in the risk quantification. The contribution of these ignition sources is correctly represented and modeled in the FPRA. The worksheets label them as "transients" because the frequency is not apportioned with equipment counts like other fixed ignition sources (e.g., pumps).
IGN-A7	IGN – Fire Ignition Frequencies	Closed	NUREG/CR-6850 Bin 8 description (Page 6-16) states that starting air compressors are considered part of the EDG Bin. However, these are counted as Bin 9 in the D2A ignition counts. Zone OG1 includes "Off Gas Chiller Compressor (3)" which does not appear to meet the definition for compressors in Bin 9. Zone R3B also includes a "Door air compressor control" which also does not look like a compressor. (F&O ID 1-15)	Compressors CMPR-96-02, -03, -29, -30, which are in compartment D2A and D2B, were re-classified in the analysis because they are included in bin 8, i.e. they are counted as part of the diesel generator. Bin # 9 is appropriate for the off-gas chiller compressor, since there is not a separate category for chillers. Finally, the door air compressors were removed from analysis as this is a small compressor with no targets within the zone of influence and does not meet the definition of compressors to be counted as ignition sources in NUREG/CR-6850.
IGN-A9	IGN – Fire Ignition Frequencies	Closed	Appendix E of the Fire PRA Notebook Ignition Frequency provides a summary of the transient and Fixed Ignition Frequencies for each fire compartment/zone. All zones include a transient frequency (not zero). Review of the detailed data sheets show that the transient summary includes junction boxes and self-ignited cable fires (typically considered fixed ignition frequencies (see suggestion). A detailed review of each zone indicated that only the torus (R1) includes a zero transient fire frequency. Area R1 can still have a non-zero transient fire frequency, based on transients being left in the torus area during an outage, and no-inerting occurs. This is considered low likelihood and low risk, but a requirement of the standard. (F&O ID 1-2)	The footnote to Table 2 in the IGN notebook, which gives the influence factors that are used to calculate the transient frequency, is revised as follows: "Fire Zone R1 is filled with inert gas during power operations; therefore, access is prohibited by design. During an outage, it is possible that combustible transients could be left in the fire zone; however, this is considered a low likelihood. A zero (0) value for influence factors is given to R1.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
IGN-A1	IGN – Fire Ignition Frequencies	Closed	Per section 2.3 of the NMP1 IGN notebook, the generic frequencies provided in Reference 3 (supplement 1 to NUREG/CR-6850) are sufficient for this analysis and is assumed to be used. A majority of the inputs identified in the FMDB table "tblGenericFrequencies" confirms that the Supplement 1 are used (see attached in next page) except the discrepancy associated with Bin 40 for the iso-phase bus duct (8.24E-4). The generic frequency for Bin 40 for iso phase bus ducts should have been updated to 1.85E-03 given in Table 7-3 of Supplement 1 to NUREG/CR-6850. (F&O ID 2-28)	The generic frequency for Bin 40 was changed to 1.85E-03 in tblGenericFrequencies in the fire modeling database (labeled as "GenericFrequencyNUREG" in the fire modeling database).
IGN-A4	IGN – Fire Ignition Frequencies	Closed	N1-IGN-F001 The reviews in Appendix B performed do not determine whether the plant experience during the generic data collection period which was prior to 2000 applies, since no plant specific reviews of data prior to 2000 were performed. Section 2.2 states that plant fire events are assumed to have been reported and recorded in the EPRI Fire Events database, and that the generic data is applicable to the plant experience. (F&O ID 5-1)	A sampling between dates 12/31/1999 to 6/20/2008 was selected to meet this requirement. This sampling is considered representative of the NMP1 experience. Reviews of fire events suggest no unusual pattern that can be attributed to a specific ignition source type. Plant-specific updates are not necessary if plant records do not reveal an unusual pattern of fires in terms of number of events and types of ignition source (see Chapter 6 of NUREG/CR-6850). Therefore, the generic frequencies provided in Reference 3 are sufficient for this analysis.
IGN-A4	IGN – Fire Ignition Frequencies	Closed	IGN Notebook The reviews in Appendix B performed do not describe the criteria used in the screening process. Response to peer review questions indicate that the screening is *based on the fire being a challenging fire as described in* NUREG/CR-6850. The screening performed is consistent with the provided response. Also, the Fire Events screened were updated to include the events from the BWROG inquiry on fire events. Section 2.2 which describes the method for screening does not fully describe the range of events selected for Appendix B. (F&O ID 5-10)	Documentation was updated as suggested.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
IGN-A7	IGN – Fire Ignition Frequencies	Closed	During the walkdowns supporting the Peer Review assessment, the reviewers noted the following: Unlabeled Items - Items without MEL labels were difficult to pick out during the walkdowns, since in cases where there are multiple unlabeled items of the same type, it is difficult to determine which is the one assigned to the Ignition source ID number. If the equipment cannot be correlated with the MEL numbers, it may be difficult for maintenance and update of the FPRA. (F&O ID 5-33)	This is classified as a suggestion; the addition of a location description based on this comment will be evaluated for the next Fire PRA update.
PP-C2	PP – Plant Partitioning	Closed	Exclusion of locations contained within the licensee controlled areas are not specifically addressed by the plant partitioning notebook. The notebook discusses disposition of Unit 1 and Unit 2 specific facilities located within the Owner Controlled area however no mention is made of facilities located licensee-controlled area located outside of the owner controlled area. (F&O ID 3-2)	Added new Figure 2 to PP notebook that shows NMP1 and excluded facilities. Added a few clarifying sentences to page 2. Added justification for exclusion of facilities.
PP-B5	PP – Plant Partitioning	Closed	N1-PP-F001 S3.0 States that no credit is taken for active barriers. However, PRA documentation does not describe the boundaries used, and references only the FHA in the UFSAR. UFSAR Appendix 10A Review of UFSAR 10A-20 appears to credit a water curtain for protecting the TB wall from fires in the transformers. The water curtain was implicitly credited as saving the boundary between the TB and transformer through incorporation of the USAR as the definition for zone boundaries. Based on the response to question 03-04 fire doors and fire dampers are credited fire barrier elements in the PB&P. Accordingly where credited these elements must be defined and justified as providing adequate separation. (F&O ID 5-3)	Page 7 of PP notebook was revised to justify the use of fire dampers and to clarify that the water curtain system is not an active fire barrier, as follows: "Regarding active fire barriers, fire dampers are credited as active separation elements, since they are rated either 1.5 or 3 hours as listed in plant reference C-42355-C Sh. 1 and 2. There is a manually-operated water curtain system that protects the turbine building from a transformer exposure fire. This is not considered an active fire barrier. The deluge system that protects the transformer is credited in the Fire PRA."

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
PP-B1	PP – Plant Partitioning	Closed	UFSAR Appendix 10A Review of UFSAR 10A and the Fire Zone drawings in N1-PP-F001 Appendix C does not clearly define the boundaries used to partition the fire zones when the partitions are not fire rated boundaries. These non-rated walls are utilized as partitioning elements but are not described in the PRA analysis to determine which walls, seals, or dampers are credited in the analysis. If localized features (fire barriers, wrap, etc.) are used to define the boundaries, it is unknown because the actual boundary used is not clearly defined. It cannot be clearly determined whether any fire zone boundaries overlap, since the non-rated boundaries utilized are not clearly defined. N1-ES-F003 Appendix B The multi-compartment analysis performed describes the barrier type which is applied between compartments, though still does not define the location of the physical boundary, i.e. which wall, which seal, which damper. Examples of this include: • B-40143-C SH5: Between RS2E and RS2C at Mw/20.3 • B-40144-C Sh5: Between AB4A, AB4B, AB4C. (F&O ID 5-4)	The PP notebook (Section 3) indicates that no fire zones are overlapping. The PP notebook was revised (Section 4) to indicate that the multi-compartment report (Table 3) shows what types of barriers are credited for each barrier.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
PP-A1	PP – Plant Partitioning	Closed	UFSAR Appendix 10A Covers safety related SSD areas. Does not appear to clearly discuss what is in/out of the boundary beyond the drawings. The EXT area is defined only as that which is not Unit 2 but outside of Unit 1. It is unclear what could be considered Unit 2 and outside the GAB, and what is not unit-related and would be included as EXT. From this it appears that the documentation does not clearly evaluate areas that were outside the FHA analysis to determine whether they needed to be included in the GAB. An example of where this causes confusion is the Access Passageway connecting the 277' of the Administration Building of Unit 1 to Unit 2. At the entrance to the Administration Building from the passageway, the wall is shown to have a 3 hour fire rating on the attached drawing, B40144C-005 (near the intersection of Column Lines B and 25). It is unclear whether the passage is considered U1 (EXT) or U2 (Outside GAB), except for its lack of being addressed in the remainder of the U1 analysis. (F&O ID 5-5)	New Figure 2 was added to PP notebook to show EXT areas, NMP2 areas, and those within the global analysis boundary (GAB). Revised text to reference the figure and to justify exclusion of areas not included in GAB.
PP-B7	PP – Plant Partitioning	Closed	N1-PP-F001 S4.0 Describes that walkdowns were performed, and presented in the FMDB. FMDB Task 1 Comments Field: Compartment walkdowns performed focus on Task 6 MOS factors. They describe the compartment function in general but do not describe partitioning features nor do they justify whether the walkdowns agreed that credited partitions match the FHA zone boundaries. (F&O ID 5-7)	The descriptions of the barriers were improved in the fire modeling database (task 1 comment field), where needed. The PP notebook explains that confirmatory walkdowns verified the conditions of the credited partitioning elements between the fire zones, as described in the UFSAR, and no deviations were adopted. Reference to the walkdown notes, in which the plant partitioning elements were addressed, was added to Section 4.0.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
QNS-B1	QNS – Quantitative Screening	Closed	A comparison between App. B of PP notebook and Table B-1 of the QNS notebook identified that fire compartments FBZT261S and R1 are not included in QNS. Based on NMP1 staff responses, R1 is the containment, which is inerted and no fires are postulated in there. FBZT261S is an imaginary zone for Appendix R purposes that is not marked on the floor. It is no longer treated as a zone since it is within T3B in the turbine building. The impacts/targets in this zone are already accounted for. (F&O ID 2-1)	The descriptions in Appendix B of the PP notebook were revised to explain that (1) R1 is the containment, which is inerted during power operation and no fires are postulated, and (2) FBZT261S is not treated as a zone since it is within T3B in the turbine building and the impacts/targets are already accounted for. No change was necessary in QNS.
PP-C1	PP – Plant Partitioning	Closed	The global plant analysis boundary is defined as containing the NMP Unit 1 primary buildings defined as the Reactor, Turbine, Radwaste Solidification Storage, Waste Storage, Screenhouse/Pumphouse, Offgas, and Administrative Buildings. Areas located outside of these primary buildings are identified as fire compartment EXT for exterior. The specific boundary with respect to the EXT fire compartment is not well defined. This fire compartment is said to contain Unit 1 FPRA credited equipment only and excludes Unit 2 SSCs. Specifically the EXT fire compartment is defined as being designed 'as a catch-all for equipment outside of the plant structures.' The approach used to 'exclude' Unit 2 and non-Unit 1 associated SSCs is discussed in detail in Section 2.0 however documentation of the specific global analysis boundary is not provided. Accordingly the boundary of the EXT fire compartment and global plant analysis boundary that separates Units 1 and 2 is defined by function rather than construction. This SR is considered met however an F&O has been generated to more clearly define the global plant analysis boundary physically rather than by function. (F&O ID 3-1)	Figure 2, which clearly shows EXT, was added to the PP notebook. In addition, Section 2.0 of the PP notebook was revised to discuss the fact that the buildings within the global analysis are protected from the fire hazards in the EXT by distance as well as fire resistive barriers.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
PP-A1	PP – Plant Partitioning	Closed	The NMP-1 response to Peer Review question 05-03 includes some provided justification for selection of certain global boundaries, including the need to include switchyard, Access Passage, and other Unit 2 areas as part of the GAB. Information in the plant response should be included as part of the PRA documentation. (F&O ID 5-6)	The NMPNS response to peer review question 05-03 was added to section 2.0 of the PP Notebook.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
FQ-A2	ES – Equipment Selection	Closed	The initiating event assigned to each FRANX scenario is identified in the FRANX scenario table. The approach is described in the Plant Response Model report, section 3.0. This report mentions the following clarification: Initiating events such as %LOSP (loss of offsite power), %LOF (loss of feedwater), %LOC (loss of condenser), %MSIV (main steam isolation valve closure), %IORV (inadvertent open relief valve), %A2X (loss of Powerboard 102), %A3X (loss of Powerboard 103), and %RWX (complete loss of Reactor Building Closed Loop Cooling) are not assigned as induced initiators at the compartment level because the equipment that could cause these initiators is modeled in the fire PRA. Assigning these initiating events in fire scenarios where the corresponding equipment has not been damaged by fire would be overly conservative. However, without the proper assignment of each IE to the specific fire scenario where these events may occur, the FPRA may under-predict the overall CDF due to failure to model the proper timing and success criteria for the fire sequence. For example, a loss of feedwater, feedwater overfeed or IORV opening would have different operator action timing (available time) that a reactor trip followed by a loss of feedwater, feedwater overfeed or IORV opening. As another example, the MSO expert panel report documented the evaluation of feedwater overfill issue and the resolution is to add logic for FW overfill in Fire PRA. The logic under gate OVERFILL reflects the changes documented in PRM notebook. However, no new FW overfill initiator was added. As a result, the PRM cannot address the impact caused by FW overfill initiator plus the functional failures resulted from it as reflected in gate OVERFILL. Particularly since the loss of FW initiator is not modeled under gate IE-XXFLF, when the fire sequence choose %LOF as initiator for C3 PNL F scenarios, the overfill issue is not covered. Therefore, as noted in the question, in this scenario, the overfill condition cannot propagate up through the gate CNFLF to cause loss of Ecs. (F&O ID 1-24)	Based on the response to Finding 1-10 (addressed in the Equipment Selection (ES) Notebook), no new approach for assigning the appropriate initiating event to each fire scenario is required. The response analysis for actions in a post fire event are deemed to be sufficiently conservative to allow for any minor timing differences that may result from differences in the sequential failures (i.e. a reactor trip followed by a loss of feedwater versus a loss of feedwater induced trip) in any accident sequence. The initiator logic associated with the OVERFILL sequence of events does address initiators that could result in the disruption of the feedwater/steam flow cycle without a complete loss of feedwater. Complete loss of feedwater should not result in the overfill condition. The internal events PRA identified 4 initiating events that could lead to overfill (Loss of condenser, MSIV Closure, Loss of Instrument Air, and Rupture of Variable Leg) and total loss of feedwater was not one of them. Therefore, the fire scenario identified and assigned the Loss of Feedwater initiator would not result in overfill, and this part of the fault tree is not relevant to those sequences.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
PRM-A1	PRM – Plant Response Modeling	Closed	Per Utility Response to a question on the MSIV failure modeling; There are two types of scenarios modeled for MSIV failure to close: (1) TBVs opening and (2) TBVs not opening. Note also that the BOC model includes both ISLOCA and Other LOCAs outside containment. For fires these are not pipe breaks. 1) Circuit analysis feedback indicated that TBVs did not have cables (hydraulic) thus fire does not cause them to open or stay open. If these were to spuriously open it would be similar to large LOCA outside containment and CDF could happen very quickly, but the frequency is very low for this sequence. 2) Assuming the TBVs are working properly (this is what is showing up in results), the model also address the fact that there is some leakage believed to be in the small LOCA range, but no plant data was ever obtained as to how small this leakage is. Failure of MSIVs to close with loss of feedwater in the accident sequence model is treated as a small LOCA outside containment. Without feedwater and with a small LOCA, RPV level will drop actuating Ecs, but level will continue to TAF requiring emergency depressurization and loss of the Ecs. This logic and failure to provide makeup at low pressure is modeled as core damage. Operator actions to close MSIVs is included in the model (locally venting air from MSIV is included in Level 2 as preventing LERF). (F&O ID 1-9)	A detailed circuit analysis of the Main Steam Isolation Valves failure was performed which allowed certain cables to be screened out. A requantification of the model revealed that the frequency of the sequence associated with the MSIVs failure to close with turbine bypass valves (TBVs) working properly had been reduced such that the conservatism in the model was acceptable.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
PRM-A1	PRM – Plant Response Modeling	Closed	Queries have been built in FRANX database to identify potential inconsistencies: A large number of cables (e.g., 1S-580, 11DV-57A, 11DV-57A, 171-176, etc.) are not included in the 'Cable_to_Component' table. A check in 'ZONETAG' table shows that all these cables are mapped to a number of BE's. Several check of these BE's have also been performed to verify that they are modeled in fault trees. Site response indicated that all the cables that have no mapping in the query were verified, including the ones listed as examples in the question statement. In addition, ES Notebook table D-1, item 100 mentions PCV-33-39 needs to spuriously open when IV-33-04 spuriously operates in order to get a LOCA. However, PCV-33-39 does not have cables. These removed cables are associated with future plant mods that have been incorporated in the FPRA as listed in the FQ draft notebook. The mods are simulated in the FPRA by not mapping the affected cables to the basic events. The two tables (tbl_MOD_Cables_to_Nullify and tbl_MOD_CableToBE_to_Nullify) are included in FRANX support database that we use to ease quantification. However, these table results have not been documented. On the other hand, the FPRA model does not reflect the as-built as-operated plant conditions due to this approach, which is similar to issues covered in F&O 4-3. (F&O ID 2-13)	Proposed modifications have been reflected in the Fire PRA model to simulate the post transition "as built as operated plant". As part of the NFPA transition process, including the fire risk evaluation applications, the proposed modifications will either be approved and implemented or the model will be revised to remove the modification. Two tables (tbl_MOD_Cables_to_Nullify and tbl_MOD_CableToBE_to_Nullify) were added to the FQ notebook.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
SY-A4	HRA – Human Reliability Analysis	Closed	Plant walkdowns and interviews with knowledgeable plant personnel (e.g., engineering, plant operations, etc.) for the new system model changes are not evident. NMP response states this was not necessary for the minor system changes made due to fire. However, the system model changes are extensive. The system model changes should satisfy system model requirements as specified in HLR-SY-A & B SRs to ensure model integrity. (F&O ID 2-17)	The only major system logic changes were made to facilitate cable mapping, operator actions and recoveries and fire PRA unique equipment failure modes. The system model changes that were made satisfied the system model requirements specified in HLR-SY-A&B. Interviews with knowledgeable plant operations personnel were conducted by the Fire PRA team to address the system logic changes. Walkdowns were performed by plant personnel as an iterative process as questions arose and were incorporated directly as system model changes. Since the system model developers for the fire PRA were also involved in the Fire HRA task 7.12, the interviews served the dual purposes of informing both the PRM and HRA tasks. Additionally, plant electrical experts were directly involved with modeling changes associated with cable mapping logic. The Fire HRA Notebook section 4.1.4 discusses the plant personnel interviews conducted during the weeks of 11/22/10, 6/27/2011 and 9/25/2011. Information from these interviews was utilized to modify the system models as appropriate to reflect equipment and operator interactions and was also factored into the detailed analysis for HEP quantification. Summaries of the interview information are included in the Operator Interview Insights field of the HRA Calculator files provided as Appendix D to the Fire HRA Notebook. In response to Peer Review findings 5-14 and 5-15, additional interviews with Operations and Training were conducted during the week of 12 December 2011 and the Fire HRA Notebook operator interview documentation has been expanded into a new Appendix so that it is more consistent with the style and content of Appendix C of the internal events HRA notebook. Based on the extent of plant walkdowns and involvement by knowledgeable plant personnel, no further walkdowns or interviews are considered necessary at this time.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
SY-A3	PRM – Plant Response Modeling	Closed	Dependencies, instrumentation & control, and other requirements in this SR were not fully addressed in system model updates for fire. Please note the following are only examples. Addressing these specific examples does not justify the missing steps required by the related SRs. A. For example, breakers R10 and R40 and motor operated disconnect switch 8106 have been added to the model (also shown in simplified one line diagram). However, the control for these breakers does not seem to be modeled. Neither have the dependencies been modeled (e.g., DC control power to the breaker). B. A second example is the containment isolation model change described in PRM notebook Appendix C. A number of valves have been added. These valves do not seem to be included in PRA SYSTEM NOTEBOOK - SY.19 – Containment Isolation. A check of fault tree shows that the added logic is under gate ISMOV83109, which includes power and control dependencies. Operator action is modeled under gate ZIS03, which seems that operator cannot close it without power. Overall, the dependencies may not have been fully investigated and documented. C. The power dependencies for annunciators modeled in fire PRA for HRA impacts have not been modeled. D. The power dependencies for indications modeled in fire PRA to support operator actions have not been modeled. (F&O ID 2-3)	The system and component attributes listed in SR SY-A3 were considered in the Internal Events PRA and documented in Section 2.0 of the system notebooks. As components were added to the Fire PRA model these attributes were also considered, as applicable, with the exception of the annunciator and operator action indications as explained in the response to examples C & D below. The specific examples of the modeling of instrumentation & control and power dependencies were reviewed and necessary changes were made as follows: A. As part of the cable selection for breakers and motor operated disconnects, control power and interlock dependencies are considered and the necessary cables are modeled from the control power source to the device. For the specific example, breakers R10 and R40 and motor operated disconnect switch 8106, the control power was considered but was screened out as not required since the required position of these components is closed and they are normally in the closed position. DC Control power is not needed to maintain these components in the closed position. For breakers R10 and R40, the following control power cables were modeled: 11B-29, 11B-29A, 11B-29B, 12B-29A, 12B-29B, 14B-26, 1C-90, and 1S-1879. For motor operated disconnect 8106, the following control power cables were modeled: 11B-29, 11B-29A, 11B-29B, 12B-29A, and 1S-1879. B. It was determined that the fire PRA changes were identified in Section 3.8 of the SY.19 Containment Isolation Notebook. Also, it was determined that ZIS03 operator action is a local action to close containment isolation valves and does not require support systems. Given the spurious opening of these valves, local closure is required per procedure. C. Power dependencies for annunciators were not in the model. These have been added to the model and were found to not significantly impact the results. D. Power dependencies for operator action indications were also not in the model. These were added to the model and were found to not significantly impact the results.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
				The Equipment Selection (ES) Notebook Section 2.1 contains a description of the consideration of system dependencies in the selection of components for the Fire PRA model and the Fire PRA equipment list. A description of the addition of power dependencies for annunciators and operator action indications was added to Appendix C of the Plant Response Model (PRM) Notebook.
AS-B3	PRM – Plant Response Modeling	Closed	MSO scenario CRD-4 as documented in the PRM notebook identified potential flooding impacts in the northeast corner room. This potential flooding have not been examined for possible flooding impacts to equipment credited by the FPRA. This is one example, but there could be others. (F&O ID 4-8)	The MSO table was reviewed to locate any phenomenological impacts to systems and accident progression phenomena. The following MSOs were identified and evaluated for the impact on the Fire PRA. Scenario 2b identified hotwell impacts and environmental concerns in the turbine building; however, these impacts are already modeled in the Fire PRA as discussed in Appendix D of the PRM notebook. Scenarios CRD-3 and CRD-4 identified that if the scram discharge volume vent or drain AOVs fail to close there could be flooding impacts in the northeast corner room. Upon further review it was determined that the scram discharge volume LOCA is modeled in the NMP1 PRA and includes the impact of corner room flooding (See Section 4.2.5 and Appendix J of the IE Notebook). The Fire PRA scenario uses the same logic (%HELBSDC logic in BOC event tree). Scenario 4b, 4c, 4d identified containment overpressure impact on NPSH for the CTS and CS pumps. However, as described in Appendix C of the PRM notebook modeling the containment overpressure failure mode was not required.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
				Scenario NEW 2010 identified that inadvertent operation of the containment sprays could cause a rapid drop in containment pressure where the vacuum breakers are unable to equalize the pressure. Calculation, S0-GOTHIC-VB001 Rev 00, GOTHIC Containment Response with Various Wetwell-Drywell and Reactor Building-Wetwell Vacuum Breaker *Configurations, concludes that the inadvertent operation* of all 4 loops of containment sprays during normal plant operating conditions coincident with spurious closure of all (3) Reactor Building-to-Wetwell Vacuum Breakers will result in an exceeded containment vacuum design limit condition. However, as discussed in the Technical Specification Basis 3.3.6 and 4.3.6, a safety factor of 1.7 exists which provides adequate margin for the Torus and Drywell for this beyond design basis event. Based on the safety factor of 1.7 that exists, NMP1 is within its feasibility to withstand this event outside the design basis event. A discussion of the review of MSO scenario phenomenological impacts was added to Section 4.0 of the PRM notebook.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
PRM-B7	PRM – Plant Response Modeling	Closed	Multiple spurious operations events can introduce the need for new success criteria and this list has been examined. Several open items remain regarding success criteria for specific MSO scenarios as noted: - Failure of SCRAM MSO Scenarios 1a and 1b – These are dispositioned in Appendix D as not modeled. Condition reports have been generated and dispositioned. The modeling comments indicate 'need to complete Action.' - Feedwater overfill MSO scenario 2ak – No review is documented of equipment qualifications for the ERVs to ensure no adverse component impacts / structural issues that could affect accident sequences. - Multiple ERVs MSO Scenario 3a - currently addressed by small LOCA in the PRA model. Opening of multiple or all ERVs may be more appropriately addressed by medium / large LOCA. The disposition is that the consequences are the same regardless of the number of ERVs that open. However, no discussion is made of the timing differences that could affect human interactions and potential plant damage state differences. - Non-synchronous paralleling of EDGs MSO Scenario 5f – This MSO scenario was left open- Makeup from EC Makeup Tank MSO Scenario EC-4a – This scenario results in a quicker loss of inventory will be reflected in the time available for this manual action in the HRA analysis. The comment indicates timing 'will be' reflected. Has this been included in the HRA analysis? - EC Cross-connect valve on makeup line MSO Scenario EC-9 - The inventory in the EC takes more than one hour to deplete, therefore requiring make-up. This should provide sufficient time for opening this valve. In addition, its function is only needed in conjunction with failure to provide make-up to EC from its primary source. However, this action is not modeled in the FPRA and therefore success criteria such as event timing have not been addressed.	The specific examples were reviewed and recommendations were developed as described below. Additional information was added, as applicable, to the modeling comments for these scenarios in Appendix E of the PRM notebook. Failure of SCRAM MSO Scenarios 1a and 1b – These are dispositioned in Appendix D as not modeled. Condition reports have been generated and dispositioned. The modeling comments indicate "need to complete Action." Response: For MSO scenarios 1a and 1b, NMPNS is taking the position presented by the BWR Owner's Group document BWROG-TP-11-011 titled 'BWROG Assessment of Generic Multiple Spurious Operations (MSOs) in Post-Fire Safe Shutdown Circuit Analysis for the Operating of BWR Plants'. Given the unlikely set of circumstances required for this condition to occur and to remain in effect until such time that it could pose a beyond design basis concern to the reactor, the risk associated with this issue is judged to be low. Additionally, given the fact that there are multiple barriers (circuit failure characteristics, design features, procedural guidance and rigorous operator training) in place to prevent, and mitigate the consequences of this condition, the safety significance of this issue is also judged to be very low. The disposition and modeling comments in Table E-1 of the PRM notebook have been updated to reflect this position. Feedwater overfill MSO scenario 2ak – No review is documented of equipment qualifications for the ERVs to ensure no adverse component impacts / structural issues that could affect accident sequences. Response: Feedwater overfill modeling was enhanced to support the spurious operations of electric pumps failing to trip on high level, flow control valves spurious open and fail to close on high level, failure to declutch shaft driven pump and flow control valve due to fire that can cause this event. In addition the model includes logic with regard to preventing overfill (e.g., FCV -29-137 and FCV -29-141 closure, pump trip etc). Fires that cause overfill are also conservatively treated as failing feedwater as an injection system and operator action is required to recover an emergency condenser (operator

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
			- MSO Scenario CRD-4 - As noted in Appendix B under High/Low Pressure Interfaces, if the scram discharge volume (SDV) vent (IV-44.2-15 and 16 in series) or drain (IV- 44.2-17 and 18 in series) AOVs fail to close (spuriously kept open), there could be flooding impacts in the northeast corner room. These flooding impacts have not been addressed by the FPRA. Inadvertent operation of containment spray New 2010 MSO scenario - Inadvertent operation of the containment sprays can cause rapid drop in containment pressure where the vacuum breakers are unable to equalize the pressure. The expert panel recommended that this scenario be further analyzed, e.g., with thermal hydraulic calculations, to either confirm or dispose the scenario. Such analysis has not been performed. This is not currently in the NMP1 Fire PRA and cannot be modeled until the scenario is analyzed and the impacts are determined. High room temp detectors can cause SDC trip New 2010 MSO scenario - Added Room Coolers to SDC system model with gate 'SDROOMCOOL' directly under main system gate 'XXSDF'. For success criteria, ANDed room coolers (HTX-202-83, 84) such that both are required to fail. However, no thermal hydraulic basis (e.g., room heatup analysis) appears to be provided for this modeling. (F&O ID 4-9)	action ZFL01). If this is not successful, another operator action (operator action ZOD01) is required to open ERVs and emergency blowdown for low pressure injection even though they may already be open. This is considered to be conservative modeling (e.g., no credit for feedwater success given overfill and then requiring ERV's to be open by an operator action). Multiple ERVs MSO Scenario 3a - currently addressed by small LOCA in the PRA model. Opening of multiple or all ERVs may be more appropriately addressed by medium/large LOCA. The disposition is that the consequences are the same regardless of the number of ERVs that open. However, no discussion is made of the timing differences that could affect human interactions and potential plant damage state differences. Response: Small LOCA modeling, as defined in the accident sequence, is conservative with respect to always requiring an operator action to open ERV's for low pressure injection. Regardless of how many ERV's are spuriously opened by the fire, the operator action to open the ERV's is credited. In addition, all the injection systems for low pressure (e.g., core spray, raw water, and fire water) are capable of providing adequate injection for steam LOCA whether it is small or large as captured in the accident sequence. The potential non-conservatism is associated with the time available for operator actions to align fire water. For firewater alignment operator action, there is a separate HEP basic event for multiple ERVs spuriously opening (see operator action ZOR15). Non-synchronous paralleling of EDGs MSO Scenario 5f – This MSO scenario was left open. Response: This scenario was added to the model. Makeup from EC Makeup Tank MSO Scenario EC-4a – This scenario results in a quicker loss of inventory that will be reflected in the time available for this manual action in the HRA analysis. The comment indicates timing 'will be' reflected. Has this been included in the HRA analysis?

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
				Response: The PRA model has always addressed the reduced time and impact. Given loss of instrument air or spurious opening of LCV-60-17 or 18, the EC shells will overfill through the vents resulting in quicker loss of storage tank inventory. Long term makeup to the storage tank is required sooner. The PRA model accounts for this timing impact on the HEP for long term makeup (ZLT01, ZOU01 and 2). Another operator action (ZOMU1) addresses whether operators control EC makeup via manual valves in the LCV-60-17 and 18 flow paths. Success leads to use of ZOU01 and failure leads to use of ZOU02. EC Cross-connect valve on makeup line MSO Scenario EC-9 - The inventory in the EC takes more than one hour to deplete, therefore requiring make-up. This should provide sufficient time for opening this valve. In addition, its function is only needed in conjunction with failure to provide make-up to EC from its primary source. However, this action is not modeled in the FPRA and therefore success criteria such as event timing have not been addressed. Response: There has never been a need to credit this crosstie and it is not important to the PRA results. The reason for this is that firewater makeup to the EC storage tanks is via manual valves and operator actions. There is really nothing major to recover with this valve other than to provide some more time for long term makeup. MSO Scenario CRD-4 - As noted in Appendix B under High/Low Pressure Interfaces, if the scram discharge volume (SDV) vent (IV-44.2-15 and 16 in series) or drain (IV- 44.2-17 and 18 in series) AOVs fail to close (spuriously kept open), there could be flooding impacts in the northeast corner room. These flooding impacts have not been addressed by the FPRA.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
				Response: The SDV LOCA in the BOC event tree models this scenario for non-fire scenarios. This same event tree is utilized for fire induced events so it is included in the fire PRA. Inadvertent operation of containment spray New 2010 MSO scenario - Inadvertent operation of the containment sprays can cause rapid drop in containment pressure where the vacuum breakers are unable to equalize the pressure. Response: Calculation, S0-GOTHIC-VB001 Rev 00, GOTHIC Containment Response with Various Wetwell-Drywell and Reactor Building-Wetwell Vacuum Breaker Configurations, concludes that the inadvertent operation of all 4 loops of containment sprays during normal plant operating conditions coincident with spurious closure of all (3) Reactor Building-to-Wetwell Vacuum Breakers will result in an exceeded containment vacuum design limit condition. However, as discussed in the Technical Specification Basis 3.3.6 and 4.3.6, a safety factor of 1.7 exists which provides adequate margin for the Torus and Drywell for this beyond design basis event. Based on the safety factor of 1.7 that exists, NMP1 is within its feasibility to withstand this event outside the design basis event. High room temperature detectors can cause SDC trip New 2010 MSO scenario - Added Room Coolers to SDC system model with gate 'SDROOMCOOL' directly under main system gate 'XXSDF'. For success criteria, ANDed room coolers (HTX-202-83, 84) such that both are required to fail. However, no thermal hydraulic basis (e.g., room heatup analysis) appears to be provided for this modeling. Response: This dependency was added as a result of the fire PRA MSO evaluation (see SY.15 Shutdown Cooling System Notebook). No thermal hydraulic analysis was required as the room coolers are designed for the heat loads anticipated in the rooms.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
PRM-B10	PRM – Plant Response Modeling	Closed	A search for internal event components not included in ES equipment selection shows that the following Bes are set to fail per NMP1.rr file: BV_122_04__VAZN1 BV-122-04 spuriously Open BV_94_164__VAZD1 BV-94-164 AOV Fails to Open on Demand BV_94_206__VAZD1 BV-94-206 AOV Fails to Open on Demand Only BV-122-04 was not included in the ES table G-1, Fire PRA Equipment List. However, in the FRANX model, BV_122_04__VAZN1 was changed to BV_122_04__VAZN3 with a failure probability of 0.01. The description for this change states that BV-122-04 Spuriously Open (mod for fire with cables not assigned). This appears that BV-122-04 may have been treated with a plant mod to remove all cables. Such change is not documented in the PRM notebook. In the later task CF, this valve is further evaluated to ensure that a hot short that clears within 15 minutes can be tolerated without adverse consequences. A note has been added to Table 14 in NMP notebook N1-CF-F001, Fire PRA Notebook: Circuit Failure Mode Likelihood Analysis (CF). Thus this is mainly a documentation issue that the PRM notebook should clearly document the treatment for systems and equipment that were included in the Internal Events PRA but were not selected in the ES element. (F&O ID 2-15)	The modeling for BV-94-164, BV-94-206, and BV-122-04 was reviewed and it was determined that all three valves do not have cables evaluated and should have a failure probability of 0.01 set for all fire events. These valves were determined to meet the criteria for three-phase proper polarity hot shorts defined in EPRI 1019259 (NUREG/CR-6850 Supplement 1, Fire Probabilistic Risk Assessment Methods Enhancement), Section 16.2 and discussed in Section 2.3.1.4 of the Circuit Failure Likelihood (CF) notebook. This type of hot short is expected to have a limited duration (less than 15 minutes). These valves have also been evaluated to ensure that a hot short that clears within 15 minutes can be tolerated without adverse consequences. The fire PRA equipment list in Table G-1 of the Equipment Selection Notebook, Appendix C of the Plant Response Model (PRM) notebook, and Section 2.3.2.4 of the Circuit Failure Likelihood (CF) notebook were revised to be consistent with this information.
PRM-B2	PRM – Plant Response Modeling	Closed	NMP1 internal events F&O suggestions have not been reviewed, which may become significant in fire PRA. (F&O ID 2-19)	The suggestions from the Internal Events PRA peer review were reviewed and it was determined that they fell into one of three categories. 1) The suggestion involved Internal Events PRA documentation only; 2) The suggestion was already addressed in the Internal Events PRA rollout; or 3) The suggestion had been addressed in revisions to procedure CNG-CM-1.01-3003, PRA Configuration Control. In each of these categories, no impact on the Fire PRA was identified. The PRM discussion of Internal events F&Os in Section 2.0 was expanded to include a discussion of the review of Internal Events PRA peer review suggestions.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
SY-A2	PRM – Plant Response Modeling	Closed	The existing NMP1 system models have been modified to address fire PRA changes. The changes are documented in both system notebooks and the NMP1 PRM notebook N1-PRM-F001. However, a number of system notebooks have not been fully updated yet. (F&O ID 2-2)	As the Fire PRA was being developed, changes to the system notebooks have been drafted. At the time of the peer review, these updated system notebooks had not been reviewed and approved. These updates will be finalized in conjunction with the next update to the Internal Events PRA model using normal NMPNS processes.
SY-B1	PRM – Plant Response Modeling	Closed	New components added in fire PRA do not have CCF modeled. A question has been sent to NMP1. The responses states that they cannot think of many cases without thorough review. Most items added were spurious operations (e.g., R10 and 40 mentioned above) for which CCF is not modeled. Also, when components were added by request of circuit analysis, CCF was not necessary modeled. An example would be the EDG Fuel Oil Transfer Pumps. These were considered part of EDG boundary from the internal events model and were only added to support circuit analysis. (F&O ID 2-4)	A list of the basic events that had been added to the Fire PRA was reviewed to determine whether common cause failure needed to be modeled. It was determined that the additions were either spurious operations where the risk contribution due to common cause failure would have been negligible or were fire induced failures which already had a small risk contribution and therefore, the risk contribution due to common cause failures would have been even smaller. A discussion of the treatment of common cause failure in the Fire PRA was added to Section 3.2 of the PRM notebook.
PRM-B8	PRM – Plant Response Modeling	Closed	A suggestion is made to further clarify the documentation of modeling changes for the following items. That is the new basic events added are documented, however, no discussion is made about where the basic events are placed in the FPRA model (e.g., gates / sequences location in the FPRA model), their impacts (what impact occurs to functions modeled in the FPRA, e.g., inventory control, containment heat removal) and logic (describe the fault tree logic as applicable). - Emergency Cooling Model (loss of inventory) - Feedwater Overfill Modeling - Shutdown Cooling Model (pump dead head) - Shutdown Cooling Model (trip on loss of room cooling) - MG 167 - EDG Fuel Transfer Pumps - EC Model (spurious isolation) - ERV Model (spurious opening) - Core Spray Model (Failure to Open) - Feedwater System - PB16 and PB17 Out-of-Phase Spurious Crosstie (F&O ID 4-10)	A statement has been added to Section 3.0 of the PRM notebook which refers the reader to the system notebooks for a more detailed discussion of the changes made as a result of the development of the Fire PRA.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
HRA-D1	HRA – Human Reliability Analysis	Closed	The PRM Notebook, Appendix B, states: Offsite power recovery basic events . . . are set to failure using the FireInitiatorHRA feature of FRANX (Appendix C) to ensure that such recovery during fires is 'evaluated prior to applying recovery'. Similarly, Appendix B states that DHR-REC and EDG recovery were set to failure so that recoveries for fires could be 'evaluated prior to apply recovery'. No evaluation is made by the FPRA to apply these recoveries. NMP1 responded that these statements in the notebook were outdated and potentially misleading. (F&O ID 4-20)	References to the evaluation of recovery events before they are applied were removed from Appendix C (formerly Appendix B). Description now just indicates that the recovery events are set to failure.
PRM-B5	PRM – Plant Response Modeling	Closed	A suggestion is made to revisit this SR and its referenced SRs if any new fire-specific initiating events are identified as part of resolving other F&O findings related to fire-specific initiating events identification and success criteria development. (F&O ID 4-6)	With the identified responses to the other Initiating Event related F&Os (1-10, 1-11, 1-24, and 6-7), no additional initiating events or accident progressions were identified.
AS-B3	PRM – Plant Response Modeling	Closed	A new potential fire-induced BOC pathway is modeled in the FPRA via the turbine bypass valves. The phenomenological impacts: loss of condenser, potential pipe break, and containment bypass for the LERF modeling are identified and modeled. However, the documentation would benefit from a more detailed discussion of the phenomenological impacts. (F&O ID 4-7)	A detailed circuit analysis of the Main Steam Isolation Valves failure was performed which allowed certain cables to be screened out. A requantification of the model revealed that the frequency of the sequence associated with the MSIVs failure to close with turbine bypass valves (TBVs) working properly had been reduced such that the conservatism in the model was acceptable. The frequency of the phenomenological impacts were similarly determined to be reduced, and therefore, no additional discussion is warranted.
QNS-C1	QNS – Quantitative Screening	Closed	Section 4.0 of N1-QNS-F001 documents that the highest risk fire areas are not screened. Section 2.2 documents the sum of the CDF and LERF are less than 10% of the estimated total internal events CDF and LERF, which meets capability category II. (F&O ID 6-3)	The QNS notebook describes the initial qualitative screening that was performed. It provided insights into the reasonableness of the Fire PRA model but no scenarios were screened based upon the results. Therefore, updating the notebook with subsequent model results is not necessary.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
SF-A4	SF – Seismic Fire	Closed	*The Seismic Fire FPRA Notebook does not mention the* presence or absence of mercury switches on fire detection or suppression systems. Because of their operation mode these type switches make their parent systems susceptible to spurious operation during a seismic event. Therefore identification of these type systems is important when considering seismic effects. In response question no. 03-05 site personnel indicated that relays containing mercury switches were identified during the IPEEE study and that the study recommended making improvements. In addition the IPEEE reported that the improvements were scheduled for completion by the end of the 1999 refueling outage. However no documentation concerning the resolution of these improvements are documented in the Seismic Fire FPRA Notebook. Accordingly it is not known if these type switches remain installed in plant fire detection or suppression systems. (F&O ID 3-9)	*It should be noted that recommendations were made* and referenced in the IPEEE study for replacing certain mercury type contacts associated with Cardox systems. At the time, these Mercury switches were found to be a potential seismic systems interaction (i.e., stops fans and closes rollup doors in EDG rooms). These improvements were scheduled for completion at the end of the 1999 refueling outage. These systems since then have been transitioned to manual operation, which significantly reduces the possibility of spurious release. In addition, the Fire PRA is currently assuming a modification that will re-design the circuits for the fans and doors.
SF-A5	SF – Seismic Fire	Closed	The FIRE PRA Notebook Seismic Fire (SF) includes a recommendation to improve the fire brigade training procedures, based on the review. However, the assessment does not include a summary of the evaluation of the existing training. (F&O ID 1-17)	The following statement has been added to the SF Notebook: "The practice at Nine Mile Point is to respond to fire alarms, discharge of fire suppression systems and fire events in general as described in the pre-fire plans (N1-PFP-0101). These pre-fire plans do not cover fire brigade activities following a seismic event. "

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
IGN-A10	IGN – Fire Ignition Frequencies	Closed	The Ignition Frequency Calculation provides an estimate of the uncertainty intervals for each of the Ignition Frequency Bins for each area. These intervals are based on the generic values in NUREG/CR-6850 and Supplement 1. The total scenario frequency uncertainty is provided in the uncertainty report, Table 4-4. These uncertainty values include the total interval when combining the ignition frequency, non-suppression and severity factors. The method for developing the uncertainty intervals is provided in 4.2.2 of this report, and the method for combining variances is included in 4.2.1. The data, formulas, functions, equations and queries for developing the uncertainty intervals is included in the Fire Modeling database, and is not fully documented in the uncertainty or ignition frequency report. The queries developed in the database were review, to the extent possible, to determine if they provide an accurate approach for developing uncertainty intervals for each scenario and ignition frequency. However, due to the amount of interconnections between numerous queries, functions and tables, the process was not traceable without further description and documentation. As a result, an F&O was developed to include this analysis in the uncertainty calculation, and to better document the process for developing uncertainty intervals. A review of the uncertainty intervals for various zones and scenarios was performed, and generally the results looked reasonable. However, several zones/scenarios did not appear to be accurate. For example, Scenario AB4D-Cmpt-10, is shown to have an EF of 38 (from the fire database query provided to the Peer Review team), which does not agree with the ignition calculation. Scenario T1-Cmpt-10, shows an EF of 1.6, again not agreeing with the ignition calculation (note; although the TB big fire EF is small, it appears to be around 2.5 when the various contributions are combined). Overall, about 20% of the ignition sources appear to be questionable (e.g. uncertainty calc table shows a large number of uncertainty factors at 1.0), although since the process is not easily traced, it is not clear the amount of error here. (F&O ID 1-14)	The uncertainty notebook was updated to include detailed explanations on how the uncertainties of fire scenarios are calculated, including propagation of uncertainties in generic ignition frequencies, severity factors, and non-suppression probabilities. A discussion was added to explain that the value of the error factor can be significantly affected by the incorporation in the fire frequency of adjustment factors such as severity factor and non-suppression probability, and also by summing ignition source frequencies together. As a consequence, the fire scenario error factors calculated in the Uncertainty report may be significantly different from the error factors associated with the contributing ignition sources given in the Ignition Frequency report.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
QU-E3	Quantification	Closed	Table 4-4 computes parametric uncertainty based on the fire scenario frequency uncertainty times the CCDP point estimate. Propagation of uncertainties requires inclusion of all basic events existing in the cutsets. The actual 5th and 95th percentiles associated with CDF need to account for correlations between fire scenarios that share the same ignition frequency bin. Hot short probabilities, fire non-suppression probabilities etc. would also each be correlated. (F&O ID 2-36)	The report was updated to include the plant-level CDF uncertainty range, accounting for the uncertainty about the CCDP and the fire frequency. This uncertainty range was calculated using the software package UNCERT, which accounts for correlations between basic events. Results are now shown in Table 4-10.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
CS-A1	CS - Cable Selection	Closed	Detailed circuit analysis was employed for the NMP1 FPRA to improve realism by screening out cables that would not cause the mal-operation in question. Component functional failures are identified for detailed circuit analysis through review of the FPRA results. A detailed circuit analysis for the majority of the basic events associated with spurious operation was performed. Of 943 basic events that required cables, 616 had detailed circuit analysis performed. Of these, components associated with the electrical system, core spray, containment spray, containment spray raw water, emergency condensers, main steam isolation, and the high pressure coolant injection mode of feedwater were the most prevalent systems that received detailed circuit analysis. However, this review finds risk significant fire-induced component failures still exist in the FPRA for which detailed circuit analysis has not been performed. Detailed circuit analysis has not been performed to a large enough extent to demonstrate that sufficient realism is provided for the PRA modeling: numerous fire scenarios have CCDPs of 1.0, and examples of risk-significant component functions have been identified for which no detailed circuit analysis has been performed, as listed below. These are merely examples (Fussel Vessely between 0.2 and 0.03): MSIV Failure to Close – Component IV-01-04 (MSIV) is mapped to 53 cables, many of which appears to be associated with solenoid valves. The MSIV logic is that it takes 2 spurious operations to fail the MSIVs open. MSIV failure to close appears to be the dominant issue for a number of the top CDF scenarios (e.g., in T3A, etc.). Detailed circuit analysis has not been performed to include the proper impact of the MSIV cable logic. The result is that a single cable failure is modeled as an MSIV failure.	In order to demonstrate sufficient realism is provided for the PRA modeling, perform detailed circuit analyses for risk significant component functional failures. Model the impact taking into account the combinations of failures that must occur to cause the failure and the probability of those failures (e.g., hot short probabilities). This was performed in the CS notebook N1-CS-F001. All potential conservative mapping was reviewed to ensure that the issues listed were the only required changes. *The resolution of this F&O was identical to F&O 2-5.*

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
			LI-36-76A GEMAC Instrument - LI_36_76AFZIIZL1 has 22 cables assigned and is risk significant. IV-39-05 Emergency Condenser Return has 83 related cables and is risk-significant. FCV-29-137 has 89 related cables and is risk-significant. (F&O ID 4-1)	
CS-A1	CS - Cable Selection	Closed	The cable selection assumes all components are in their normal positions, e.g., pumps, valves, relays, position switches, control switches, isolation switches, transfer switches. Guidance in NUREG/CR-6850 is that a system [or component] can be assumed to be in its normal configuration and not in an unusual line-up, such as during test and/or maintenance, as long as the time in these less-usual configurations is small (e.g., ~1%) relative to the time it is in its normal configuration. A review was not performed to determine the time spent in any less-usual configurations. (F&O ID 4-2)	The PRA Component notebook addressed this generic question. It is not a concern for the CS Notebook as the CS notebook only takes the plant lineup and information as it is assumed from the PRA and PRA component notebook. No further action required.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
CS-A1	CS - Cable Selection	Closed	Section 4.2 of NMP1 Fire PRA Cable Selection notebook states the following: Only those cables which transverse back to or reside *inside the plant that adversely affected a primary* component needed to be identified for electrical equipment whose associated cabling is partially outside the plant structure (e.g. offsite power switchgear and oil circuit breakers). The word 'plant' is vague in this discussion and causes confusion. The utility was queried regarding this paragraph and the utility clarified as follows: The word 'plant' is defined as the buildings associated with the 'Power Block.' In the context of the FPRA, the power block is defined in the plant partitioning notebook. Therefore, all the zones in the plant partitioning notebook are within the scope of the cable selection and cable routing work. Cable selection involved identifying all the required cables whose failure could impact each primary component regardless of their location. This assumption only applied to the few cables that were not numbered and therefore were not in TRAK2000 and remained outside the plant or power block. (F&O ID 4-5)	The Cable Selection Notebook was revised to be consistent with the plant partitioning notebook with regards to language pertaining to "the plant."

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
CS-C2	CS - Cable Selection	Closed	Section 4.2 of NMP1 Fire PRA Cable Selection notebook states the following: Only those cables which transverse back to or reside inside the plant that adversely affected a primary component needed to be identified for electrical equipment whose associated cabling is partially outside the plant structure (e.g. offsite power switchgear and oil circuit breakers). The word 'plant' is vague in this discussion and causes confusion. The utility was queried regarding this paragraph and the *utility clarified as follows: The word 'plant' is defined as* the buildings associated with the 'Power Block.' In the context of the FPRA, the power block is defined in the plant partitioning notebook. Therefore, all the zones in the plant partitioning notebook are within the scope of the cable selection and cable routing work. Cable selection involved identifying all the required cables whose failure could impact each primary component regardless of their location. This assumption only applied to the few cables that were not numbered and therefore were not in TRAK2000 and remained outside the plant or power block. (F&O ID 4-5)	The Cable Selection Notebook was revised to be consistent with the plant partitioning notebook with regards to language pertaining to "the plant."
CS-A11	CS - Cable Selection	Closed	N1-CS-F001 CS Roadmap describes no assumed routing. None is otherwise described in the CS notebook. FMDB Cables where routing is requested are returned from Datatrak in table ZoneTag in the FMDB. Reviewed some requested routings and these were returned as available in ZoneTag. Process for requesting routing, getting routing, updating ZoneTag is not well documented (F&O ID 5-8)	The cable selection notebook was enhanced with verbiage in regards to obtaining routing from TRAK2000 for inclusion into the ZONETAG report.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
CS-A3	CS - Cable Selection	Closed	A number of tables have been exported from DATATRAK database and show the data tables listed in CS notebook. The subcomponents are listed in table FPRA_SUBCOMPS. However, the cable association with sub-components are not documented. All the cables associated with sub-components are listed under the primary components. This treatment is conservative, which could potentially mask the model results. Similarly, the FPRA_INTLK table includes all interlocks for primary components. While the cables are included for the interlocks modeled as primary components, the final basic events mapped are still the primary components that the interlocks are mapped too. This treatment could also mask the model results. For example, MSIV valve IV-01-01 has following interlock: VSL-ISO-CH11-CKT1 VSL-ISO-CH11-CKT1 VSL-ISO-CH11-CKT14 VSL-ISO-CH11-CKT14 VSL-ISO-CH12-CKT1 VSL-ISO-CH12-CKT1 VSL-ISO-CH12-CKT14 VSL-ISO-CH12-CKT14 These interlocks are mapped to the following cables: 1M-56 1M-61 1M-63 1S-1059 1S-1081 1S-1082 1S-598 1M-24 1M-67 1M-68 1S-1060 1S-597	A review of the leading contributors was performed to determine the Basic Events that may require detailed circuit analysis or additional logic changes. The following BEs were identified for Logic Changes and/or detailed circuit analysis: SOV113_229BVSZD1 IV_01_01___VMZD1 IV_01_02___VMZD1 IV_01_03___VAZMD IV_01_04___VAZMD IV_05_12_OOVAZD1 IV_201_16FRVAZD1 This list was based on an "Ones Run" that was performed to identify those BEs that may have an over conservatism based on impact on the overall model. Logic changes were performed and addition Detailed Circuit Analysis completed to provide a further refinement of the model logic. This was incorporated into the current BE File. NOTE: The treatment/classification of subcomponents are defined in the Equipment Selection Notebook and the mapping of cables to subcomponents was described in the Cable Selection Notebook. See Resolution under F&O 1-13 for further clarification.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
			A number of the above cables fail basic event IV_01_01_11ECZHD, which is channel specific. However, a number of cables fail directly the following MSIV basic events: IV_01_01___VMZD1 IV_01_02___VMZD1 IV_01_03___VAZMD IV_01_04___VAZMD (F&O ID 2-5)	

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
CS-A4	CS - Cable Selection	Closed	Some FPRA events (i.e., IV_39_06_01ECZHD) affect what appear to be common relays (R37B/36B) and multiple components (39-08, 39-10R, 39-06). Based on several questions on this, it appears that many of the subcomponents are listed under other subcomponents. For example, RLY-36-06B-K17B is mapped under DPT-36-06B in the AREVA database (not in the CAFTA or FRANX files). These new subcomponents are included in the FRANX tables. However, the cables associated with the subcomponents are also further screened in the AREVA DB, and not all cables associated with the subcomponents are mapped to the major component. Again as an example, the DPT-36-06B cable 1F-156 is screened as not impacting all of the primary EC components impacted by DPT-36-06B. A series of tables were sent to the review team which clarifies the sub-component and cable relationships (see question 1-19), including FPRA_SUBCOMP, FPRA_COMPLIST, and others. Although these are not easily used, they do provide clarification on the above issues. These tables and the associated description of the tables and process needs to be included in the FPRA. Finally; with the mapping as performed, it appears the FPRA conservatively models the impact of cable failures. For example, with DPT-36-06B, the AREVA modeling shows that the cable failure 1F-151 does not impact IV-39-10 (valve would fail safe on failure of 1F-151). However, the cable does impact the DPT basic event DPT_36_06B_ITPLD under CAFTA gate EI3910SIG12 which can fail IV-39-10. So, by having the AREVA DB map the cables to both the primary components and the subcomponents, the screening of the cable from the primary component is negated. The above is just an example. The issue appears potentially large based on a review of the number of cables with 10 or more components associated with the cables. (F&O ID 1-13)	A model review was performed on subcomponents pertaining to transmitters in the Fire PRA Model that are receiving cables in the ZoneTag. This review concentrated on ensuring that the detailed circuit analysis that was performed for the primary component was not negated by adding the cables back into the logic through the subcomponent. The steps that were followed are: 1. A list of subcomponents (Transmitters) and associated logic was identified for review. 2. The subcomponent Basic Event was analyzed for proper cable selection for each place it was identified in the Fire PRA Model Logic. 3. The logic was then compared to the remaining logic for the primary component the transmitter was a subcomponent of. This ensured cables were not being re-introduced into the primary component's logic based on screening the cable from the primary component basic event and including it as part of the subcomponent basic event. 4. For any identified conservatisms, the model was revised to remove the need for cables from the subcomponent as needed. Additional reviews were performed to ensure the removal of cables did not negatively impact other logic where the subcomponent basic event may have been applied. From this review, only one group was identified where a conservatism applied from re-introducing cables from the subcomponents basic event. This group of basic events is associated with failure to isolate Loop 11 or 12 EC. The BEs for DPT-36-06A, B, C, D have been revised. NOTE: The treatment/classification of subcomponents is defined in the Equipment Selection Notebook and the mapping of cables to subcomponents is described in the Cable Selection Notebook.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
CS-A4	CS - Cable Selection	Closed	The ZONETAG table was provided as a response to peer review questions. 6239 records have no TOEVENT (MEL) field specified. For example, Cable 12-42 is mapped to component BKR-(12/1-9)52. However, this cable is also failing BE BKR_1R42_41CAZO1, which is introduced from ZONETAG table without TOEVENT (MEL) specified. The reviewer suspected that the cable information could be in error, since breakers BKR-(12/1-9)52 and BKR-(1R42)41 (modeled BE is BKR_1R42_41CAZO1) are parallel as shown in the fault tree under gate RCP14_BREAKERS. Response from NMP revealed that the cable association to BKR-(1R42)41 was in error and will be removed for next update to ZONETAG. (F&O ID 2-11)	During finalization of Detailed Circuit Analysis reviews it was determined that the population of this field is not required due to the cable being directly associated to the Basic Event in the Detailed Circuit Analysis form. This cable was added to the basic event from a detailed review and determined to be required to support the defined function of the basic event. This cable may not be associated to any primary component by which cable selection was performed. Added statement to response to state that the DATATRAK and ZONEATG reports are included with the CS Notebook. These reports are explained in the CS Notebook. A note was added to ensure the TOEVENT field has a justification for some records being blank.
CS-B1	CS - Cable Selection	Closed	The Nine Mile Point Unit 1 NFPA 805 Coordination Study addresses overcurrent coordination and protection study examines circuit overcurrent protection (common enclosure associated circuits) and found that there are no cable thermal damage concerns. However, the analysis findings are distributed throughout the document with no summaries provided. A suggestion is made to enhance the documentation. (F&O ID 4-4)	The Cable Selection Notebook was updated to include the discussion for excluding MHIF.
CS-C1	CS - Cable Selection	Closed	A number of tables listed in these sections are not documented as attachments in CS notebook N1-CS-F001, Fire PRA Notebook: Cable Selection Notebook. These tables have been provided as a responses to peer review questions, which have been verified to show the requested information. In addition, the FRANX tables are not consistent with the table definitions in Section 6.0. (F&O ID 2-9)	Documentation was revised to add the listed tables in CS notebook and update the table definitions to be consistent with FRANX tables.

Table V-1 Fire PRA Peer Review – Facts and Observations

SR	Topic	Status	Finding	Disposition
PRM-B2	PRM - Plant Response Model	Closed	The NMP1 internal event PRA model has been updated substantially after last peer review. A new peer review or self-assessment has not been performed. (F&O 2-20)	Sections 1-5 and 1-6 of the Standard address the subject of what requires peer review. The Standard requires peer review of upgrades, not maintenance changes, and the review would be focused on only the upgrade changes. In addition, procedure CNG-CM-1.01-3003, PRA Revisions, includes PRA upgrades, PRA moderate maintenance, and PRA minor maintenance. The internal event F&Os responses and PRM notebook described the changes that were made since the internal events peer review. NMPNS reviewed and characterized these changes as PRA minor maintenance and PRA moderate maintenance. Therefore, a new peer review or self-assessment is not considered at this time.
IGN-B3	IGN – Fire Ignition Frequencies	Closed	Documentation of the software and database files supporting the FPRA analysis is not clear. Appendix A of each report is reserved for documenting the software and files associated with the analysis. Most of the Appendix A entries do not describe the files used as intended. The files utilized to perform the analysis described in the FPRA Task notebooks are not described. Data tables and queries within the database files are also not well described, and include a number of old, obsolete, or otherwise unused objects. The FMDB is used to generate all mapping tables used in FRANX models. The technical bases behind the FMDB is documented in its technical reference and user manuals. However, detailed query structures and macros / VBA codes may not have been verified and documented. Errors could exist if the design of the database is not fully verified and tested, or the application of the tools in the database is different from what is expected during the design. Changes can be made to the files which would not be captured or documented as changes in the notebooks. (F&O ID 5-11)	Databases that are used in the preparation of FPRA inputs are described in the main bodies of the notebooks and in appendices cited in the main bodies. The technical basis behind the FMDB is documented in its technical reference and user manuals. Cut set reviews were performed for the verification of the FPRA model. Cut set reviews can identify errors in all elements of the FPRA including the cable and raceway database content and structure, the fire induced risk model including human reliability, and the fire modeling database content and structure. Issues identified as a result of cut set reviews were addressed as a part of development of the FPRA model. Verification of the FPRA model is also performed by independent reviews of reports.

Table V-2 Fire PRA – Category I Summary

Table V-2 Fire PRA – Category I Summary

SR	Topic	Status
SY-A4	Plant walkdowns and interviews with knowledgeable plant personnel (e.g., engineering, plant operations, etc.) for the new system model changes are not evident. NMP response states this was not necessary for the minor system changes made due to fire. However, the system model changes are extensive. The system model changes should satisfy system model requirements as specified in HLR-SY-A & B SRs to ensure model integrity. (F&O ID 2-17)	The only major system logic changes were made to facilitate cable mapping, operator actions and recoveries and fire PRA unique equipment failure modes. The system model changes that were made satisfied the system model requirements specified in HLR-SY-A&B. Interviews with knowledgeable plant operations personnel were conducted by the Fire PRA team to address the system logic changes. Walkdowns were performed by plant personnel as an iterative process as questions arose and were incorporated directly as system model changes. Since the system model developers for the fire PRA were also involved in the Fire HRA task 7.12, the interviews served the dual purposes of informing both the PRM and HRA tasks. Additionally, plant electrical experts were directly involved with modeling changes associated with cable mapping logic. The Fire HRA Notebook section 4.1.4 discusses the plant personnel interviews conducted during the weeks of 11/22/10, 6/27/2011 and 9/25/2011. Information from these interviews was utilized to modify the system models as appropriate to reflect equipment and operator interactions and was also factored into the detailed analysis for HEP quantification. Summaries of the interview information are included in the Operator Interview Insights field of the HRA Calculator files provided as Appendix D to the Fire HRA Notebook. In response to Peer Review findings 5-14 and 5-15, additional interviews with Operations and Training were conducted during the week of 12 December 2011 and the Fire HRA Notebook operator interview documentation has been expanded into a new Appendix so that it is more consistent with the style and content of Appendix C of the internal events HRA notebook. Based on the extent of plant walkdowns and involvement by knowledgeable plant personnel, no further walkdowns or interviews are considered necessary at this time.
HR-E3	HRA Notebook HRA Calculator Operator Interview insights documents some operator review, but does not describe further talk-throughs or detailed review of sequences with the operator. NMP was able to provide somewhat more detailed notes that clearly showed general review of actions for changes due to fire with ops personnel. (F&O ID 5-14)	Additional interviews with Operations and Training were conducted during the week of 12 December 2011 and the Fire HRA Notebook operator interview documentation was expanded into a new Appendix so that it is more consistent with the style and content of Appendix C of the internal events HRA notebook.
HR-E3	HRA Notebook HRA Calculator Operator Interview insights and additionally provided notes documents operator reviews performed for the Fire HEPs, but does not talk through of each action in step by step detail with Ops and Training personnel to show that it is consistent with the plant observations and operator training procedures. (F&O ID 5-15)	Additional interviews with Operations and Training were conducted during the week of 12 December 2011 and the Fire HRA Notebook operator interview documentation was expanded into a new Appendix so that it is more consistent with the style and content of Appendix C of the internal events HRA notebook.

Table V-2 Fire PRA – Category I Summary

SR	Topic	Status
HR-G1	The following risk-significant HFEs set to failed. These actions have received scoping analysis. Since they are risk-significant, it is worthwhile to consider detailed analyses, if the actions are feasible: - ZECA3F23ECOPRATR, Fire HEP MCRA: Operators Fail to Locally Open IV-39-07, 09; FV = 7.08E-03, time margin 30% - ZECI1F23ECOPRATR, Fire HEP MCRA: Operators Fail to Close Manual Valves 05-31, 32, FV = 7.03E-03, time margin 32% - ZECI2F23ECOPRATR, Fire MCRA: Operators Fail to Manually Close FCV-39-15, 16 with Local HWO, FV = 7.03E-03 - ZHRA4F23HRAOPRAT, Fire HEP MCRA: Op Fails to Use East/ West Instrument Rooms (SBO), FV = 2.01E-02 - ZOD10F23ODOPRATR, Fire HEP MCRA: Op Fail to Open ERVs from RB Given Fire Induced ERV Circuit Failure, FV = 4.32E-02 - ZOLS1F23OLSOPRAT, Fire HEP MCRA: Op Fails to Implement DC Load Shedding FV = 2.20E-02 - ZOR12F23OROPRATR, Fire HEP MCRA: Op Fails to Align Fire Water FV = 3.71E-02 - ZOR13F23OROPRATR, Fire HEP MCRA: Op Fails to Align Fire Water (SBO 1 hour) FV = 1.99E-02 - ZDPC2F23DCOPRATR, Fire HEP MCRA: Op Fails to Use East/ West Instrument Rooms (SBO), FV = 1.34E-2 - ZIS02F23MSIVLOC1, Fire HEP MCRA: Operators Fail to Locally Vent Air from MSIVs, FV (LERF) = 1.9E-2 - ZRX03F21INJCTRCV, Scoping fire HEP: operator fails to align alternate inj sources in level 2 (conditional), FV (LERF) = 1.8E-2 (F&O ID 4-17)	Detailed analysis has now been performed for all the listed events except ZDPC2 and ZRX03. ZDPC2 is for aligning the portable charger for the case when the operator previously fails to load-shed DC. The HEP is set to 1.0 because the time gets tight (8 hours if they shed, but only 4 if they don't) and the action itself takes 3 hours. Also, the action is in a damage repair procedure that is less well structured and trained upon. For the 8 hour case, this might not be significant, but for the 4 hour case, such uncertainties are potentially important. ZRX03 is a conditional Level 2 event that was set to 1.0 in the Internal Events PRA and that value was carried over to the fire PRA consistent with the EPRI/NRC Fire HRA Guidelines, NUREG-1921. There is not sufficient justification available to reduce the HEP. The values resulting from the detailed analysis of the other HFEs identified by the peer review team were provided as input to the Fire PRA model and will be documented in the next revision of the Fire HRA Notebook. Risk significant HFEs including those set to fail were all reviewed during 12/12 - 16/2011 cutset review and refinements were made as appropriate. Section 4.2.3 of the Fire HRA Notebook and particularly Table 4-6 have been updated to reflect these changes.
CF-A1	Based on a review of the results, and performance of several ones-runs, there are a number of spurious operation probability events that are presently set to 1.0 that are potentially significant. Dominant to the results is the spurious opening (or failure to close due to hot short) of the MSIVs. However, other components show up in the results (e.g., BKR_R1042__CAZM1). Based on discussions with NMP1 PRA, the determination for the existing CF analysis was based on a previous model, and additional review for needed CF probability analysis has not been performed. (F&O ID 1-22)	Consideration of spurious opening of the air operated MSIVs (IV-01-03 and IV-01-04) was added to the CF notebook. In addition, to identify additional candidates for CF likelihood calculations, FRANX "Ones" runs were performed for fire scenarios with CCDP < 0.1 and CDF > 5E-7. The CCDP constraint is based on the fact that Ones runs have not been successful when CCDP > 0.1. The CDF constraint is meant to limit the number of scenarios *to a manageable number while capturing most of the remaining aggregate risk*. LERF Ones runs were also performed for the same set of scenarios. The cut set files for the Ones runs were combined into combined CDF and LERF cut set files. From these cut set files, importance measures were calculated. For top risk scenarios with CCDP > 0.1, Browser-based cut set reviews were performed using the regular FRANX Trues cut sets. By examining the candidates identified by the importance measures and the Browser reviews, four additional components were identified for CF likelihood calculations and the CF notebook was revised (Section 2.3.2.2) to include probability estimates for the additional components. The four components added

Table V-2 Fire PRA – Category I Summary

SR	Topic	Status
		are spurious closure of AC circuit breakers BKR-(16/008B)R1042/602 (POWER CIRCUIT BREAKER - 16A / 16B BUS TIE) and BKR-(17/007B)R1052/612 (POWER CIRCUIT BREAKER - A/B BUS TIE); and spurious closure of feedwater pump discharge block valves VLV-29-08 (11 FEEDWATER PUMP DISCHARGE BLOCK VALVE) and VLV-29-09 (12 FEEDWATER PUMP DISCHARGE BLOCK VALVE).
CF-A1	Events for high rad monitor alarm are not included in the CF report. Events are set to true in the analysis, but could be caused by a hot short where a CF probability can be assigned.This includes events RAMRN06A11_IIZL1, RAMRN06B11_IIZL1, RAMRN06A12_IIZL1, and RAMRN06B12_IIZL1. In addition, there are two annunciator basic events identified in the ES notebook which are not included in the CF report. (ANN_F1_2_7_IAZL1, ANN_F4_2_2_IAZL1). (F&O ID 1-21)	The spurious operation list in the CF notebook was revised to include the additional basic events mentioned in the finding. These events were considered for application of CF probabilities.
CF-A1	There are numerous components listed as CableCode = Blank and nothing under comment and blank under MEL. For example; RLY_K8616__RAZD1. Response to a question on this indicates that recent detailed cable mapping has been done and not all the cable codes appear to be populated. (F&O ID 1-6)	A review of the basic event table in the nmp1.rr database was conducted. For records having a blank in the cablecode column, a cablecode was added and a cablecode comment was provided as appropriate. The MEL field was verified to be populated where the basic event was tied to a specific component.
CF-A1	Review of Local Manual Actions credited in the FPRA indicate a number of valves have a 92-18 potential problem (no 92-18 evaluation). As such, credit for the manual actions, given circuit failures for the valves, should not be included in the FPRA without a 92-18 evaluation. Examples of valves include: IV-31-07, 31-08 (ZIN01_INOPERATOR), and MOV 201-17 (ZCV02_CVOPERATOR). This is a random sampling, and more valves are likely not evaluated. (F&O ID 1-29)	The valves that were credited for local manual action following circuit failures were evaluated for the potential for 92-18 type unrecoverable failures. For valves IV-31-07, IV-31-08, IV-39-09R, and IV-39-10R the PRA model has been revised to allow recovery only for damage to cables that do not cause unrecoverable failures. Proposed modifications will allow operator action to recover the valves by redesign of circuits to eliminate cables that cause unrecoverable damage. For valves IV-38-13, FCV-80-118, FCV-93-71, FCV-93-74, BV-93-25, and BV-93-27 it was determined that no credit for recovery was currently taken in the model. For valves IV-38-01 and IV-38-02 it was determined that they were not susceptible to the 92-18 type failure. For valves IV-39-07R and IV-39-08R the model was revised to preclude recovery for damage to cable pairs that could cause unrecoverable damage. For valves IV-83.1-09, IV-83.1-11, IV-201-07, IV-201-09, IV-201-17, and IV-201-31 it was determined that only selected cables could cause the 92-18 type failure; therefore, the model was revised to preclude recovery for those cables only. This information was incorporated in the Plant Response Model (PRM) notebook Appendix C.
HRA-A4	HRA Notebook HRA Calculator Operator Interview insights documents some operator review, but does not describe further talk-throughs or detailed review of sequences with the operator. NMP was able to provide somewhat more detailed notes that clearly	Additional interviews with Operations and Training were conducted during the week of 12 December 2011 and the Fire HRA Notebook operator interview documentation was expanded into a new Appendix so that it is more consistent with the style and content of Appendix C of the internal events HRA notebook.

Table V-2 Fire PRA – Category I Summary

SR	Topic	Status
	showed general review of actions for changes due to fire with ops personnel. (F&O ID 5-14)	
HRA-A4	HRA Notebook HRA Calculator Operator Interview insights and additionally provided notes documents operator reviews performed for the Fire HEPs, but does not talk through of each action in step by step detail with Ops and Training personnel to show that it is consistent with the plant observations and operator training procedures. (F&O ID 5-15)	Additional interviews with Operations and Training were conducted during the week of 12 December 2011 and the Fire HRA Notebook operator interview documentation was expanded into a new Appendix so that it is more consistent with the style and content of Appendix C of the internal events HRA notebook.

W. Fire PRA Insights

22 Pages Attached

Redacted Attachment W in its entirety.