Nebraska Public Power District

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NLS2012006
April 24, 2012

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

Subject: Nebraska Public Power District - Cooper Nuclear Station
Docket No. 50-298, License No. DPR-46
License Amendment Request to Revise the Fire Protection Licensing Basis to NFPA 805 Per 10 CFR 50.48(c)

Reference: Letter from Brian J. O'Grady, Nebraska Public Power District, to U.S. Nuclear Regulatory Commission, dated June 27, 2011, "Revised Submittal Date for 10 CFR 50.48(c) License Amendment Request and Request for Extension of Enforcement Discretion" (NLS2011057)

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District (NPPD) to request an amendment to Facility Operating License DPR-46 under the provisions of 10 CFR 50.4 and 10 CFR 50.90, to revise the Cooper Nuclear Station (CNS) Operating License consistent with the adoption of National Fire Protection Association (NFPA) Standard 805 as the CNS Fire Protection Licensing Basis, in accordance with 10 CFR 50.48(c). This request is made consistent with the commitment provided in the referenced letter. A concurrent related change to the CNS Technical Specifications (TS) is also requested.

The adoption of NFPA 805 as the CNS Fire Protection Licensing Basis is effected through the replacement of the standard Fire Protection License Condition (License Condition 2.C.(4) of the CNS Operating License) with the License Condition specified in Regulatory Guide 1.205, Revision 1. A corollary change to TS 5.4.1 is also included. NPPD has determined from the No Significant Hazards Consideration determination that these changes do not involve a significant hazard.

Enclosure 1 provides the NFPA 805 Transition Report. This report describes the transition methodology utilized and documents how CNS complies with the new NFPA 805 requirements. The report includes a description of the License Condition and TS changes, the basis for the amendment, the No Significant Hazards Consideration evaluation pursuant to 10 CFR 50.91(a)(1), and the Environmental Impact evaluation pursuant to 10 CFR 51.22. Attachment 1 contains a markup of the proposed changes to License Condition 2.C.(4) and TS 5.4.1. Attachment 2 contains the clean, retyped License and TS pages.

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NRR

The proposed TS changes have been reviewed by the necessary safety review committees (Station Operations Review Committee and Safety Review and Audit Board). Amendments to the CNS Facility Operating License through Amendment 241 issued February 16, 2012, have been incorporated into this request. This request is submitted under affirmation pursuant to 10 CFR 50.30(b). NPPD proposes an implementation date within six months after issuance of the NFPA 805 License Amendment.

By copy of this letter and its attachments, the appropriate State of Nebraska official is notified in accordance with 10 CFR 50.91(b)(1). Copies are also being provided to the Nuclear Regulatory Commission Region IV office and the CNS Senior Resident Inspector in accordance with 10 CFR 50.4(b)(1).

Should you have any questions concerning this matter, please contact Todd Stevens, CNS NFPA 805 Transition Project Manager, at (402) 825-5159.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: April 24, 2012
(Date)

Sincerely,

Brian J. O'Grady
Vice President – Nuclear
and Chief Nuclear Officer

/wv

Enclosure and Attachments

cc: Regional Administrator w/Enclosure and Attachments
USNRC - Region IV

Cooper Project Manager w/Enclosure and Attachments
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/Enclosure and Attachments
USNRC - CNS

Nebraska Health and Human Services w/Enclosure and Attachments
Department of Regulation and Licensure

NPG Distribution w/o Enclosure and Attachments

CNS Records w/Enclosure and Attachments

ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©[4]

Correspondence Number: NLS2012006

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
NPPD will implement the new NFPA 805 fire protection program to include procedure changes, process updates, and training to affected plant personnel as identified in Table S-3 of Attachment S to Enclosure 1.	NLS2012006-01	Within six months after issuance of the NFPA 805 License Amendment.

ENCLOSURE 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

Transition Report

Nebraska Public Power District
Cooper Nuclear Station
Docket Number 50-298

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

Nebraska Public Power District

Transition Report

April 2012

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Executive Summary

Nebraska Public Power District (NPPD) will transition the Cooper Nuclear Station (CNS) 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.

NPPD submitted the initial letter of intent in December 2005. The intended submittal date was revised in a September 2008 letter. Most recently, in a letter dated June 27, 2011, NPPD committed to submit the 10 CFR 50.48(c) License Amendment Request (LAR) by April 27, 2012.

The transition process consisted of a review and update of CNS documentation, including the development of a Fire Probabilistic Risk Assessment using NUREG/CR-6850 as guidance. This Transition Report summarizes the transition process and results. This Transition Report contains information:

- Required by 10 CFR 50.48(c)
- Recommended by guidance document Nuclear Energy Institute (NEI) 04-02 Revision 2 and appropriate Frequently Asked Questions (FAQs)
- Recommended by guidance document Regulatory Guide 1.205 Revision 1

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

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

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

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

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

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

List of Acronyms

AC	alternating current
ACP	Auxiliary Control Panel
ADAMS	Agency-wide Document Access and Management System
ADS	Automatic Depressurization System
AHJ	Authority Having Jurisdiction
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOV	air-operated valve
ASD	alternate shutdown
ASME	American Society of Mechanical Engineers
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CC	Capability Category
CCDP	conditional core damage probability
CDF	core damage frequency
CFAST	Consolidated Model of Fire Growth and Smoke Transport
CFR	*Code of Federal Regulations*
CNS	Cooper Nuclear Station
CRD	control rod drive
CS	Core Spray
CT	current transformer
DC	direct current
DG	Diesel Generator
DID	defense-in-depth
ECCS	Emergency Core Cooling System
ECST	Emergency Condensate Storage Tank
EDG	Emergency Diesel Generator
EEEE	Existing Engineering Equivalency Evaluations
EPRI	Electric Power Research Institute
ERFBS	electrical raceway fire barrier system
F&O	facts and observations
FAQ	frequently asked question
FDS	Fire Dynamics Simulator
FHA	Fire Hazards Analysis
FPA	Foote, Pagni, and Alvares
FP	Fire Protection
FPP	Fire Protection Program
FPRA	fire probabilistic risk assessment
FR	Federal Register
FRE	fire risk evaluation
FSAR	Final Safety Analysis Report
GDC	general design criterion/criteria
GL	Generic Letter
gpm	gallons per minute
HEP	human error probability
HGL	hot gas layer
HPCI	High Pressure Coolant Injection
HRE	higher risk evolution
HRR	heat release rate

List of Acronyms (continued)

HSS	high safety significant
HVAC	heating, ventilation, and air conditioning
IMC	Inspection Manual Chapter
IN	Information Notice
KSF	key safety function
LAR	license amendment request
LPCI	Low Pressure Coolant Injection
LERF	large early release frequency
M&TE	measurement and test equipment
MCB	main control board
MCC	motor control center
MCR	Main Control Room
MOV	motor-operated valve
MQH	McCaffrey, Quintiere, and Harkleroad
MSIV	main steam isolation valve
MSO	Multiple Spurious Operations
NBI	nuclear boiler instrumentation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NPO	non-power operation
NPPD	Nebraska Public Power District
NRC	Nuclear Regulatory Commission
NSCA	nuclear safety capability analysis
OMA	operator manual action
PC	Primary Containment
POS	plant operational state
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PWR	pressurized water reactor
QAPD	Quality Assurance Program for Operation – Policy Document
RA	recovery action
RAW	risk achievement worth
RCIC	Reactor Core Isolation Cooling
REC	Reactor Equipment Cooling
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RI-PB	Risk-Informed, Performance-Based
RIS	regulatory issue/information summary
RP	Radiation Protection
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RR	Reactor Recirculation
RWCU	Reactor Water Clean-up
SDC	Shutdown Cooling
SFPE	Society of Fire Protection Engineers
SPC	Suppression Pool Cooling
SR	Supporting Requirement
SRP	Standard Review Plan

List of Acronyms (continued)

SRV	Safety Relief Valve
SSA	Safe Shutdown Analysis
SSC	systems, structures, and components
SW	Service Water
USAR	Updated Safety Analysis Report
V&V	verification and validation
VAC	volts alternating current
VDC	volts direct current
VFDR	variance from deterministic requirements
yr	year
ZOI	zone of influence

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) has promulgated an alternative rule for fire protection requirements at nuclear power plants, 10 CFR 50.48(c), National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants - 2001 Edition" (Ref. 1). Nebraska Public Power District (NPPD) is implementing the methodology identified in Nuclear Energy Institute (NEI) document NEI 04-02, "Guidance for Implementing a Risk-informed, Performance-based Fire Protection Program Under 10 CFR 50.48(c)" (NEI 04-02, Ref. 2), to transition Cooper Nuclear Station (CNS) 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 CNS 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 established new Risk-Informed, Performance-Based (RI-PB) fire protection requirements. 10 CFR 50.48(c) incorporates by reference, with exceptions, NFPA 805, as a voluntary alternative to 10 CFR 50.48 Sections (b) and (f).

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" (Ref. 3), which endorses NEI 04-02, with exceptions, in December 2009.[1]

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

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

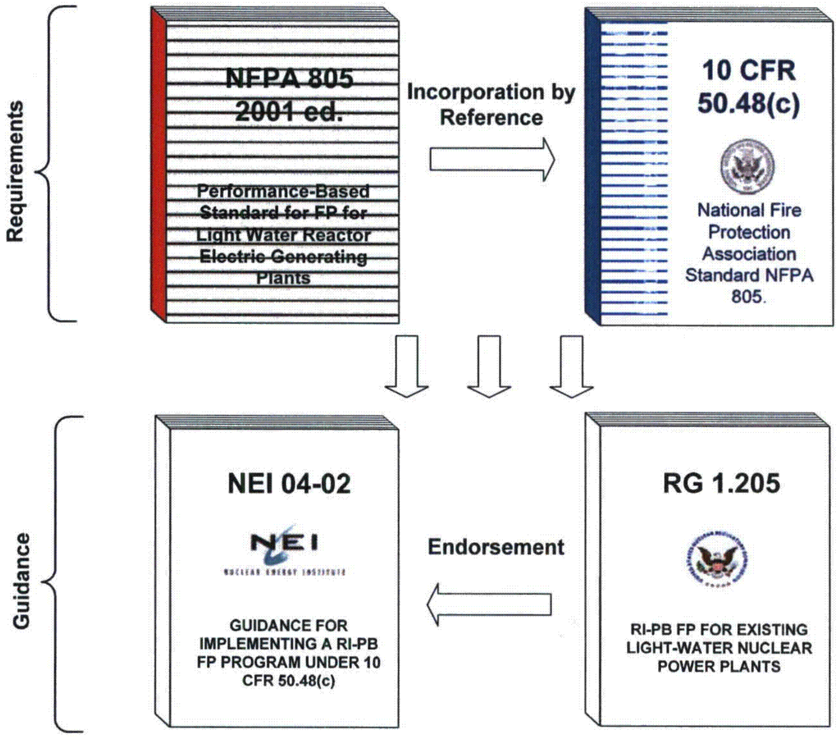

Figure 1-1 – NFPA 805 Transition – Implementation Requirements/Guidance

1.1.2 Transition to 10 CFR 50.48(c)

1.1.2.1 Start of Transition

NPPD submitted a letter of intent to the NRC on December 22, 2005 (Ref. 4), for CNS to adopt NFPA 805 in accordance with 10 CFR 50.48(c).

By letter dated March 7, 2006, the NRC acknowledged receipt of the letter of intent. The letter stated that the CNS discretion period would begin on December 31, 2005, and expire on December 31, 2007. The letter did not specifically grant the four-year enforcement discretion period requested in the letter of intent (Ref. 5).

The NRC granted a third year of enforcement discretion by Federal Register Notice 71 FR 19905, dated April 18, 2006. In accordance with NRC Enforcement Policy, the enforcement discretion period would continue until the NRC approval of the License Amendment Request (LAR) is completed.

On the basis of a revision to the Enforcement Policy (71 FR 19905), a three-year enforcement discretion period was granted to CNS by the NRC in a letter dated October 30, 2006 (Ref. 6). Therefore, the NRC considered the discretion period for CNS, which began on December 22, 2005, to expire on December 22, 2008.

On September 10, 2008, the NRC published in the Federal Register (73 FR 52705) a revision to its Interim Enforcement Policy regarding enforcement discretion for certain fire protection

issues, allowing licensees the option to request an extended enforcement discretion period for submittal of a LAR if they are pursuing transition to 10 CFR 50.48(c). This revision states that an additional period of enforcement discretion may be granted on a case-by-case basis, if a licensee has made substantial progress in its transition effort. This additional period of enforcement discretion, if granted, would end six months after the date of the safety evaluation approving the second pilot plant LAR review. The enforcement discretion would continue in place, without interruption, until NRC approval of the LAR to transition to 10 CFR 50.48(c).

By letter dated September 19, 2008, and in accordance with Federal Register Notice 73 FR 52705, NPPD requested that enforcement discretion be extended for CNS until six months after the safety evaluation is issued for the second pilot plant (Ref. 7).

By letter dated December 19, 2008, the NRC staff reviewed Nebraska Public Power District's request, and determined that the licensee had made substantial enough progress in its transition to NFPA 805 to grant the additional enforcement discretion. Accordingly, the enforcement discretion period was extended until six months after the date of the safety evaluation approving the second pilot plant LAR (Ref. 8).

On December 29, 2010, the NRC issued a safety evaluation for the second pilot plant (Ref. 9). Based on this document, the period of enforcement discretion for CNS was to end June 29, 2011.

On June 27, 2011, and in accordance with SECY-11-0033, NPPD committed to submit the NFPA 805 LAR by April 27, 2012 (Ref. 10). This was accepted in an NRC letter dated July 28, 2011, which extended enforcement discretion for CNS until NRC approval of the NFPA 805 LAR is obtained (Ref. 11).

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 (FPRA) using NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" (Ref. 12), as guidance and a revision to the Internal Events Probabilistic Risk Assessment (PRA) to support the FPRA
- 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 (FPP) to compliance 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 FPP 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 FPP, and resulting impact on the Licensing Basis.

2.0 OVERVIEW OF THE EXISTING FIRE PROTECTION PROGRAM

2.1 Current Fire Protection Licensing Basis

CNS was licensed to operate on January 18, 1974. As a result, CNS has a fire protection program based on compliance with 10 CFR 50.48(a), 10 CFR 50.48(b), 10 CFR 50 Appendix R (Sections III.G[2], III.J, and III.O[3]), Branch Technical Position APCSB 9.5-1, Appendix A, and the following License Condition.

CNS Operating License Condition 2.C.(4) states:

> The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Cooper Nuclear Station (CNS) Updated Safety Analysis Report and as approved in the Safety Evaluations dated November 29, 1977; May 23, 1979; November 21, 1980; April 29, 1983; April 16, 1984; June 1, 1984; January 3, 1985; August 21, 1985; April 10, 1986; September 9, 1986; November 7, 1988; February 3, 1989; August 15, 1995; and July 31, 1998, subject to the following provision:
>
> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

2.2 NRC Acceptance of the Fire Protection Licensing Basis

An NRC letter dated May 3, 1976, transmitted to NPPD a copy of Standard Review Plan (SRP) 9.5.1 "Fire Protection System" (Ref. 13). In a letter dated May 11, 1976 (Ref. 14) (and later clarified in a letter dated September 30, 1976 (Ref. 15)), the NRC requested NPPD compare the existing fire protection provisions at CNS with Appendix A of the SRP guidelines. The NRC also requested that Fire Protection Technical Specifications be proposed. The Fire Hazards Analysis (FHA) was developed in response to this request, and submitted to the NRC on December 17, 1976 (Ref. 16) (with a supplemental FHA submittals on March 31, 1977, and April 6, 1977 (Ref. 17 and 18)). Proposed Fire Protection Technical Specifications were submitted on February 4, 1977 (Ref. 19) (as revised on July 20, 1977, and December 19, 1977 (Ref. 20 and 21)). The NRC Staff reviewed these submittals and issued fire protection review questions, which NPPD responded to in letters dated May 11, 1978, June 21, 1978, August 16, 1978, December 11, 1978, and April 12, 1979 (Ref. 22, 23, 24, 25, and 26). On November 29, 1977, the NRC issued the initial Fire Protection Technical Specifications (Ref. 27). On May 23, 1979, the NRC issued the Safety Evaluation Report with License Amendment 56, which summarized the status of their evaluation of the FPP at CNS, and revised the CNS Fire Protection Technical Specifications (Ref. 28). By letters dated October 22, 1979, and January 16, 1980 (Ref. 29 and 30), NPPD proposed additional changes to the Fire Protection Technical Specifications, and submitted additional information concerning fire protection modifications. On November 21, 1980, the NRC issued License Amendment 66, which contained Supplement No. 1 to the Fire Protection Safety Evaluation (Ref. 31). This completed the staff's fire protection review for CNS.

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[2] In response to Generic Letter 86-10 Question 5.1.3, the NRC established that Section III.L is applicable to Paragraph III.G.3, where that option is chosen by the licensee.

[3] The 10 CFR 50 Appendix R Section III.O requirements for a reactor coolant pump oil collection system are not applicable to CNS since the containment is inerted during normal operation.

Pursuant to a scheduler exemption from the requirements of the previously issued 10 CFR 50.48 and 10 CFR 50 Appendix R, NPPD provided the NRC on June 28, 1982, the "Response to 10 CFR 50 Appendix R 'Fire Protection of Safe Shutdown Capability,'" Volumes I and II (Ref. 32). On December 2, 1983, NPPD submitted Volume III to that analysis (Ref. 33). On April 16, 1984, the NRC issued a Safety Evaluation which concluded that CNS met the requirements of Appendix R, Section III.G.3 and III.L (Ref. 34). In a letter dated July 29, 2002, NPPD provided to the NRC clarifying information contained in the April 16, 1984 Safety Evaluation, while reaffirming that the NRC's conclusion that CNS meets the requirements of Appendix R, Section III.G.3 and III.L remained valid (Ref. 35).

In letters dated May 9, 1985, and June 7, 1985, NPPD reported to the NRC the results of the re-developed Appendix R analysis and identified additional required modifications (Ref. 36 and 37). On August 21, 1985, the NRC issued a Safety Evaluation which concluded that the proposed modifications were in accordance with the technical requirements of Appendix R and were acceptable (Ref. 38).

In response to NPPD LARs, various Fire Protection Technical Specification changes were approved by the NRC:

- Amendment 82, dated April 29, 1983, regarding the Fire Protection Clean Water Supply (Ref. 39).
- Amendment 86, dated June 1, 1984, regarding submittal of 30-day Special Reports (Ref. 40).
- Amendment 90, dated January 3, 1985, regarding installation of additional fire detection instruments (Ref. 41).
- Amendment 98, dated April 10, 1986, regarding the addition of a Halon fire suppression system and fire detectors in the Service Water Pump Room (Ref. 42)
- Amendment 101, dated September 9, 1986, regarding usage of Fire Protection procedures (Ref. 43).
- Amendment 126, dated November 7, 1988, regarding Alternate Shutdown Capability (Ref. 44).
- Amendment 127, dated February 3, 1989, regarding miscellaneous changes to the FPP (Ref. 45).

Ultimately, as a result of the conversion to the Improved Standard Technical Specifications (License Amendment 178, dated July 31, 1998), the CNS Fire Protection Technical Specifications were relocated to the Technical Requirements Manual, a document incorporated by reference into the CNS Updated Safety Analysis Report (USAR) (Ref. 46).

In a letter dated December 16, 1994, NPPD identified to the NRC a number of commitment revisions, several of which affected the bases used to support specific exemptions granted by the NRC (Ref. 47). In this letter, NPPD also withdrew an exemption granted for the Critical Switchgear Rooms 1F and 1G that was determined to no longer be necessary. On August 15, 1995, the NRC issued a Safety Evaluation which revoked the exemption for the Critical Switchgear Rooms 1F and 1G (Ref. 48).

The following currently granted exemptions to 10 CFR 50 Appendix R, Section III.G are noted as having been previously approved by the NRC:

1. Exemption for lack of twenty foot separation between redundant Service Water Pumps in the Intake Structure as described in an NPPD submittal dated June 28, 1982. The NRC granted the exemption in a letter to NPPD dated September 21, 1983 (Ref. 49).
2. Exemption for lack of twenty foot separation free of intervening combustibles or one-hour barriers between redundant trains in the Cable Spreading Room as described in NPPD submittals dated June 28, 1982, March 18, 1983 (Ref. 50), and June 2, 1983 (Ref. 51). The NRC granted the exemption in a letter to NPPD dated September 21, 1983.
3. Exemption for lack of twenty foot separation or one-hour barriers between redundant trains in the Cable Expansion Room as described in NPPD submittals dated June 28, 1982, March 18, 1983, and June 2, 1983. The NRC granted the exemption in a letter to NPPD dated September 21, 1983.
4. Exemption for lack of one-hour rated fire barriers for redundant conduits and area wide automatic suppression system in the Reactor Building northeast corner at the 903'-6" Elevation as described in NPPD submittals dated June 28, 1982, March 18, 1983, and June 2, 1983. The NRC granted the exemption in a letter to NPPD dated September 21, 1983.
5. Exemption for lack of an automatic suppression system in the Control Building basement as described in NPPD submittals dated June 28, 1982, March 18, 1983, and June 2, 1983. The NRC granted the exemption in a letter to NPPD dated September 21, 1983.
6. Exemption for lack of a fixed suppression system in the Auxiliary Relay Room as described in NPPD submittals dated June 28, 1982, March 18, 1983, and June 2, 1983. The NRC granted the exemption in a letter to NPPD dated September 21, 1983.
7. Exemption for lack of a fixed suppression system in the Control Room as described in NPPD submittals dated June 28, 1982, March 18, 1983, and June 2, 1983. The NRC granted the exemption in a letter to NPPD dated September 21, 1983.
8. Exemption for lack of a three-hour barrier, or twenty feet of separation free of intervening combustibles combined with automatic suppression and detection for the redundant reactor vessel level and pressure instrument racks at the Reactor Building 931' Elevation, as described in an NPPD submittal dated March 18, 1983. The NRC granted the exemption in a letter to NPPD dated September 21, 1983.
9. Exemption for lack of a three-hour barrier, or twenty feet of separation free of intervening combustibles combined with automatic suppression and detection, and lack of alternate shutdown capability independent of the area, at the Reactor Building 903'-6" Elevation (excluding northeast corner), as described in an NPPD submittal dated March 18, 1983. The NRC granted the exemption in a letter to NPPD dated September 21, 1983.
10. Exemption for lack of three-hour barrier, or twenty feet of separation free of intervening combustibles combined with automatic suppression and detection, and lack of alternate shutdown capability independent of the area, at the Reactor Building 859' and 881' Elevations – Quadrants and Torus Area, as described in an NPPD

submittal dated March 18, 1983. The NRC granted the exemption in a letter to NPPD dated September 21, 1983.

3.0 TRANSITION PROCESS

3.1 Background

Section 4.1.2 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):

Phase 1: Preliminary Assessment and Letter of Intent

- Make preliminary determination of the activities that will be necessary to support the transition.
- Make initial determination of any changes to the plant or FPP that may be necessary.
- Establish a tentative schedule for completing all of the actions necessary for the transition.
- Submit a Letter of Intent to the NRC.

Phase 2: Analysis and License Amendment Request

- Conduct the transition activities to demonstrate compliance. Section 4.3 describes in detail how the current fire protection licensing basis can be used to support demonstrations of compliance with the requirements in NFPA 805.
- Determine extent to which the current fire protection licensing basis can be shown to demonstrate compliance with the new fire protection requirements.
- Determine any changes to the plant that will require a license amendment.
- Determine any alternative methods and analytical approaches that will be relied on to demonstrate compliance with the new fire protection requirements and will require a license amendment.
- Document the new fire protection licensing basis in a Transition Report.
- Update the schedule for completion of transition activities.
- Submit a LAR to the NRC.

Phase 3: Completion of Transition

- While the NRC reviews the LAR, complete all of the transition activities which do not require prior NRC approval, including plant changes which do not require a license amendment under the current license condition, procedure changes, and training.
- After the NRC issues the license amendment, complete any changes to the plant that required a license amendment.
- Rely on alternative methods and analytical approaches acceptable to the NRC to demonstrate compliance with the new fire protection requirements.
- Adopt the new licensing basis.

3.2 NFPA 805 Process

Section 2.2 of NFPA 805 establishes the general process for demonstrating compliance with NFPA 805. This process is illustrated 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 risk-informed, performance-based (RI-PB) methods. Consistent with the

guidance in NEI 04-02, CNS has implemented the NFPA 805 Section 2.2 process by first determining the extent to which its current FPP 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.

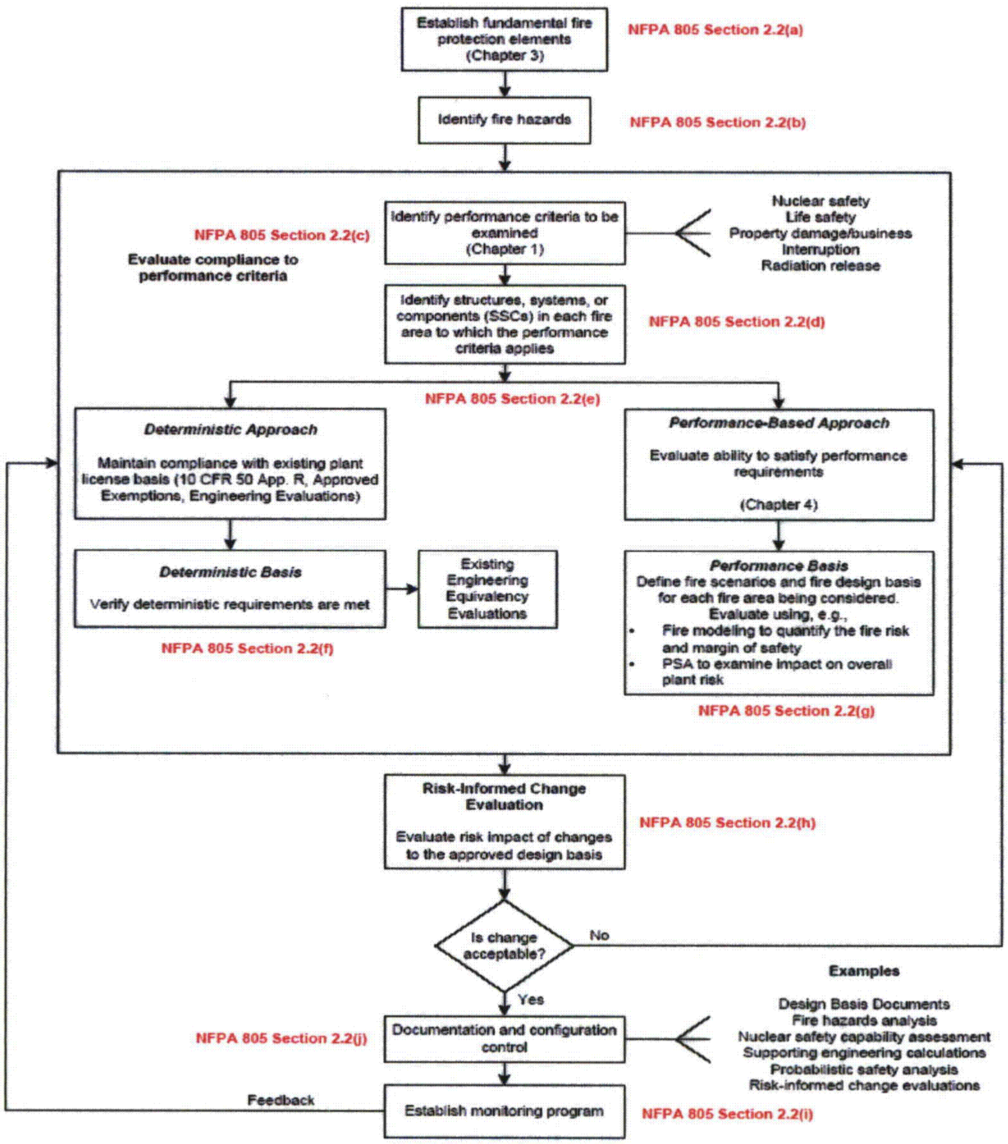

Figure 3-1 – NFPA 805 Process (NEI 04-02 Figure 3-1 based on NFPA 805 Figure 2.2)[4]

[4] 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.

3.3 NEI 04-02 – NFPA 805 Transition Process

NFPA 805 contains technical processes and requirements for a RI-PB FPP. 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 below in Figure 3-2.

Section 4.0 of NEI 04-02 describes the detailed process for assessing an FPP for the extent to which it complies with NFPA 805, as shown in Figure 3-2.

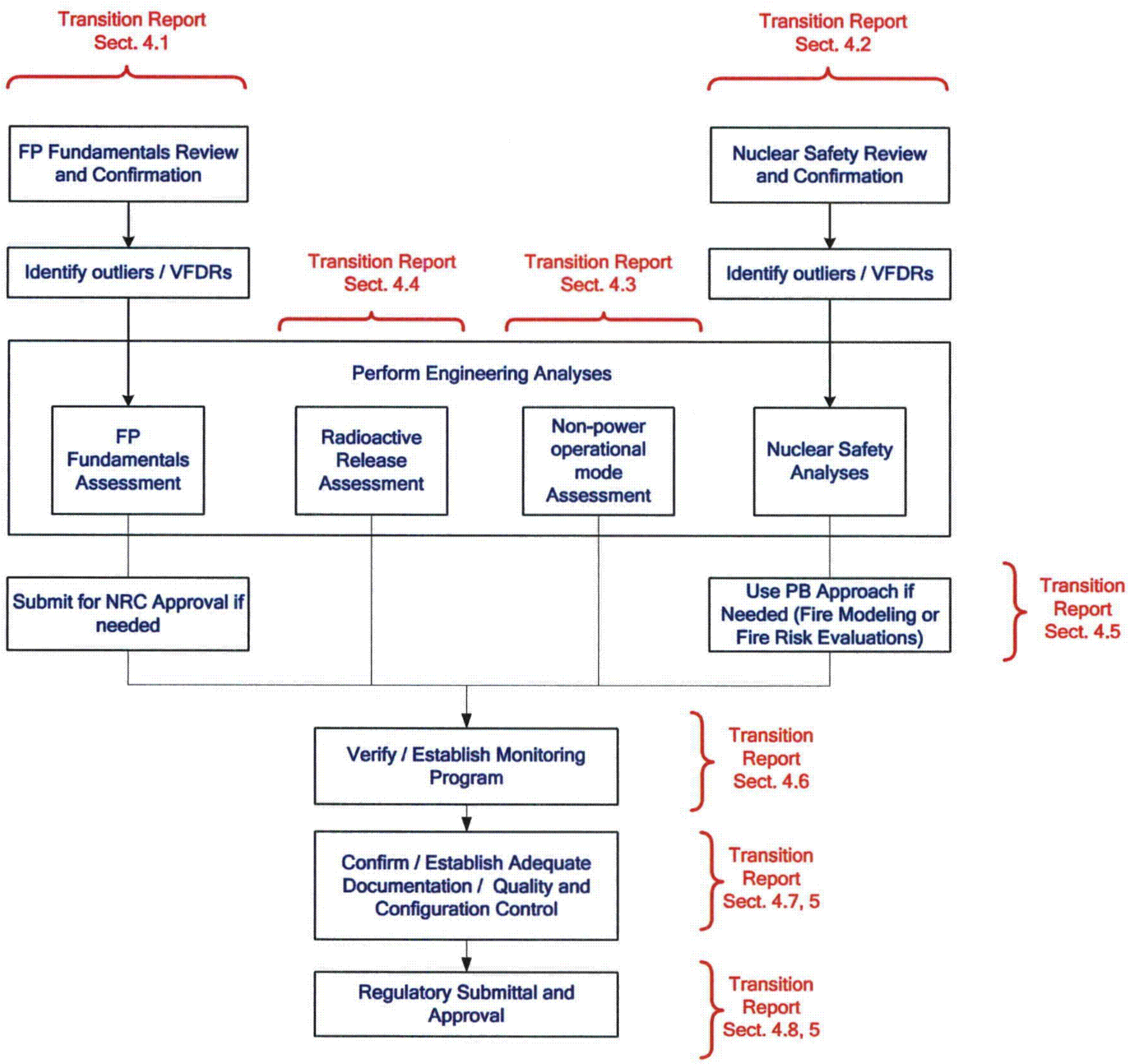

Figure 3-2 – Transition Process (Simplified) (based on NEI 04-02 Figure 4-1)

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 (Ref. 52) and in Regulatory Issues Summary 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 (Ref. 53).

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 non-pilot 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 FPP changes that would be necessary for compliance with NFPA 805. NEI 04-02 Appendix B, Section 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 CNS FPP to the requirements of NFPA 805 Chapter 3 was performed and documented in CNS Calculation NEDC 10-080 entitled "Transition of Fundamental Fire Protection Program and Design Elements". Calculation NEDC 10-080 used the guidance contained in NEI 04-02, Section 4.3.1 and Appendix B, Section B.1 (See Figure 4-1).

Each section and subsection of NFPA 805 Chapter 3 was reviewed against the current FPP, as described in Attachment A. Upon completion of the activities associated with the review, the following compliance statements were used:

- Complies - For those sections/subsections determined to meet the specific requirements of NFPA 805.
- Complies with Clarification - For those sections/subsections determined to meet the requirements of NFPA 805 with clarification.
- Complies by previous NRC approval - For those sections/subsections where the specific NFPA 805 Chapter 3 requirements are not met, but previous NRC approval of the configuration exists.
- Complies with use of Existing Engineering Equivalency Evaluations (EEEEs) - For those sections/subsections determined to be equivalent to the NFPA 805 Chapter 3 requirements, as documented by engineering analysis.
- 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).
- Complies with Required Action - Assigned to those NFPA 805 Chapter 3 sections/subsections determined to be met by the CNS FPP after completion of an action, to be completed after submittal of the NFPA 805 LAR.
- N/A – No compliance basis is necessary or applicable.

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.

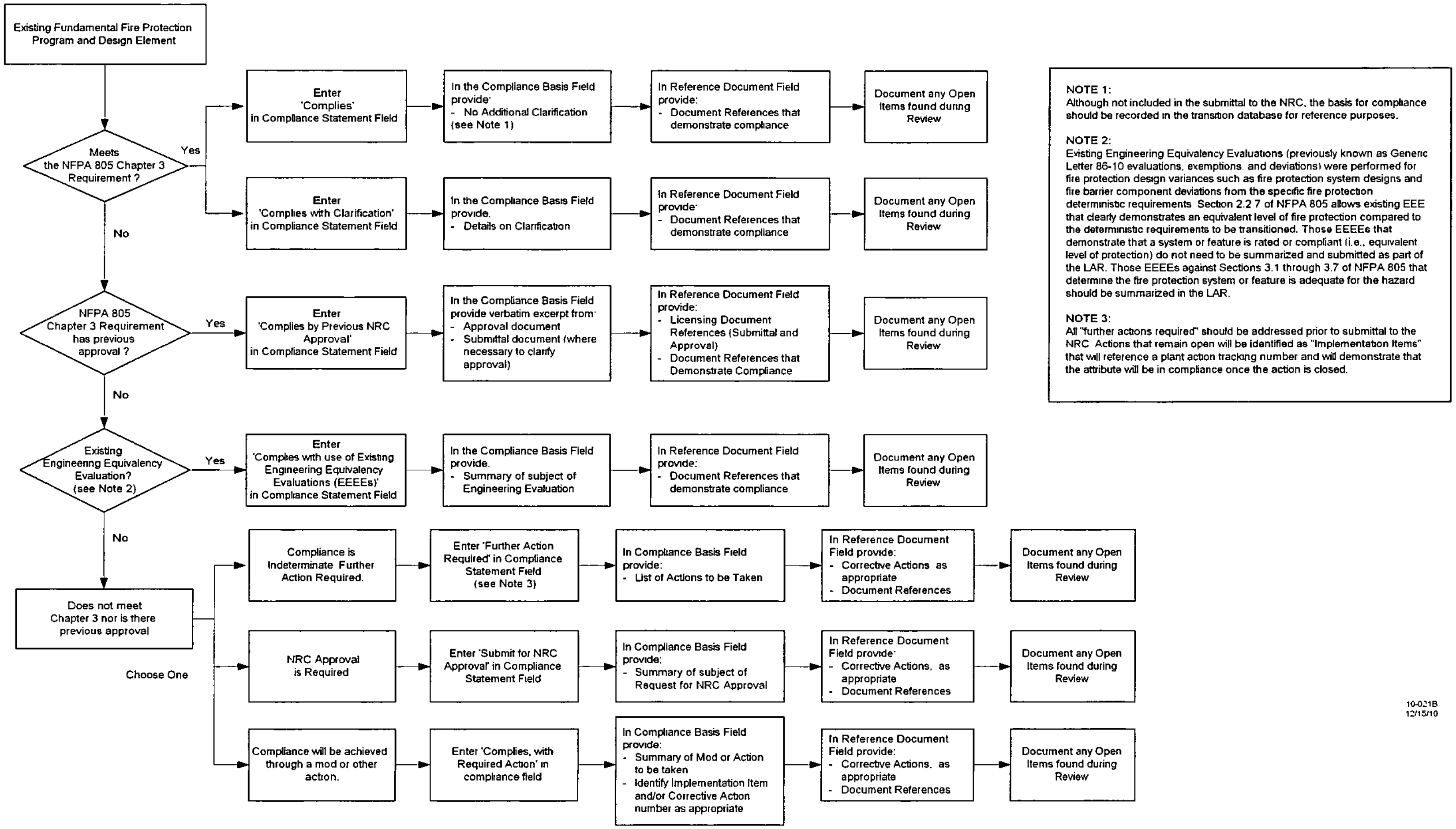

Figure 4-1 – Fundamental Fire Protection Program and Design Elements Transition Process (based on NEI 04-02 Figure 4-2) [5]

[5] 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 CNS either:

- Complies directly with the requirements of NFPA 805 Chapter 3.
- Complies with clarification with the requirements of NFPA 805 Chapter 3.
- Complies with a previously NRC approved alternative to NFPA 805 Chapter 3, and therefore the specific requirement of NFPA 805 Chapter 3 is supplanted.
- Complies through the use of EEEEs which are valid and of appropriate quality.
- Complies upon the completion of a required action. Implementation items are identified for those sections and/or subsections determined to meet the specific requirements of NFPA 805 after the completion of a modification or other action, such as a procedure change or a work request. (See Attachment S for details).
- None – No compliance basis is necessary or applicable.

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. NPPD 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:

- 3.3.1.2(1) – Approval is requested for the use of commercially available products which utilize small quantities of non-treated wood as an integral part of a finished product (e.g., tools, janitorial supplies, special fixtures, measurement and test equipment, and office type furniture).
- 3.3.5.1 – Approval is requested for certain wiring configurations above suspended ceilings that is not approved for plenum use, and not installed in conduit.
- 3.3.5.2 – Approval is requested for the use of plastic conduit in lieu of metallic raceways when installed embedded in concrete or below grade.
- 3.3.7.2 – Approval is requested for the configuration of the bulk storage hydrogen gas cylinders, which have their long axis pointing toward the Intake Structure.
- 3.5.3 (NFPA 20-1999, Section 7-5.2.4) – Approval is requested for the ability to stop the electric fire pump remotely from the Control Room.

- 3.6.1 (NFPA 14-1974, Sections 322, 442, 625, and 671) – Approval is requested for the use of lengths of hose at hose stations greater than lengths allowed by NFPA 14, standpipe outlet pressures exceeding 100 psi, the lack of approved extra heavy flanged pattern valves where the system pressure exceeds 175 psi, and the lack of water flow alarms on standpipe risers.
- 3.7 (NFPA 10-1975, Section 3-3) – Approval is requested for the placement of Class B fire extinguishers with respect to travel distance requirements for Class B hazards (flammable liquids) that are not of appreciable depth.
- 3.10.5 – Approval is requested for the lack of local keyed abort switches interlocked with the Diesel Generator (DG) CO_2 systems.
- 3.10.7 – Approval is requested for the lack of an odorizer on the DG CO_2 system.

The specific deviation and a discussion of how the alternative satisfies 10 CFR 50.48(c)(2)(vii) requirements are provided in Attachment L. NPPD 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 contain equipment required for nuclear plant operations, such as Reactor Building, Control Building, Turbine Generator Building, Diesel Generator Building, Radwaste Buildings, Intake Structure, and other structures (or areas) that are identified in the facility's pre-transition licensing basis.

These structures/areas 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 FPP changes. NEI 04-02, Section B.2 provides guidance on documenting the transition of Nuclear Safety Capability Assessment (NSCA) Methodology and the Fire Area compliance strategies.

4.2.1 Nuclear Safety Capability Assessment Methodology

The 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 (MSO)

The methodology for demonstrating reasonable assurance that a fire during non-power operation (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 for transitioning to NFPA 805 provided in NEI 00-01, Revision 1 (Ref. 54), Chapter 3, "Deterministic Methodology," as discussed in Appendix B-2 of NEI 04-02. The methodology is depicted in Figure 4-2 and consisted of the following activities:

- Each specific section of NFPA 805 2.4.2 was correlated to the corresponding section of Chapter 3 of NEI 00-01 Revision 1. Based upon the content of the NEI 00-01 methodology statements, a determination was made of the applicability of the section to the station.
- The plant-specific methodology was compared to applicable sections of NEI 00-01 and one of the following alignment statements and its associated basis were assigned to the section:
  - Aligns
  - Aligns with intent
  - Not in Alignment
  - Not in Alignment, but Prior NRC Approval
  - Not in Alignment, but no adverse consequences
- For those sections that do not align, an assessment was made to determine if the failure to maintain strict alignment with the guidance in NEI 00-01 could have adverse consequences. Since NEI 00-01 is a guidance document, portions of its text could be interpreted as 'good practice' or intended as an example of an efficient means of performing the analyses. If the section has no adverse consequences, these sections of NEI 00-01 can be dispositioned without further review.

The comparison of the CNS existing post-fire SSA methodology to NEI 00-01 Chapter 3 (NEI 04-02 Table B-2) for transitioning to NFPA 805 was performed and documented in CNS Calculation NEDC 10-039, Revision 1, "Nuclear Safety Performance Analysis Methodology Review."

Results from Evaluation Process

The method used to perform the SSA 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 for transitioning to NFPA 805 directly or met the intent of the endorsed guidance with adequate justification as documented in Attachment B.

It is important to note that the NRC, through the issuance of RG 1.205, endorsed Chapter 3 of industry guidance document NEI 00-01 as one acceptable approach to circuit analysis for a plant transitioning their FPP to NFPA 805. This endorsement was made initially on revision 1 and subsequently revision 2 of the NEI guidance document. To ensure that changes to the text in Chapter 3 that did occur between revisions did not alter the conclusions reached during the review, a gap analysis was performed in CNS Calculation NEDC 10-039, "Nuclear Safety Performance Analysis Methodology review EPM Report R1906-004-001." There were no additional justifications to address the impact of changes due to the gap analysis that required inclusion in Attachment B.

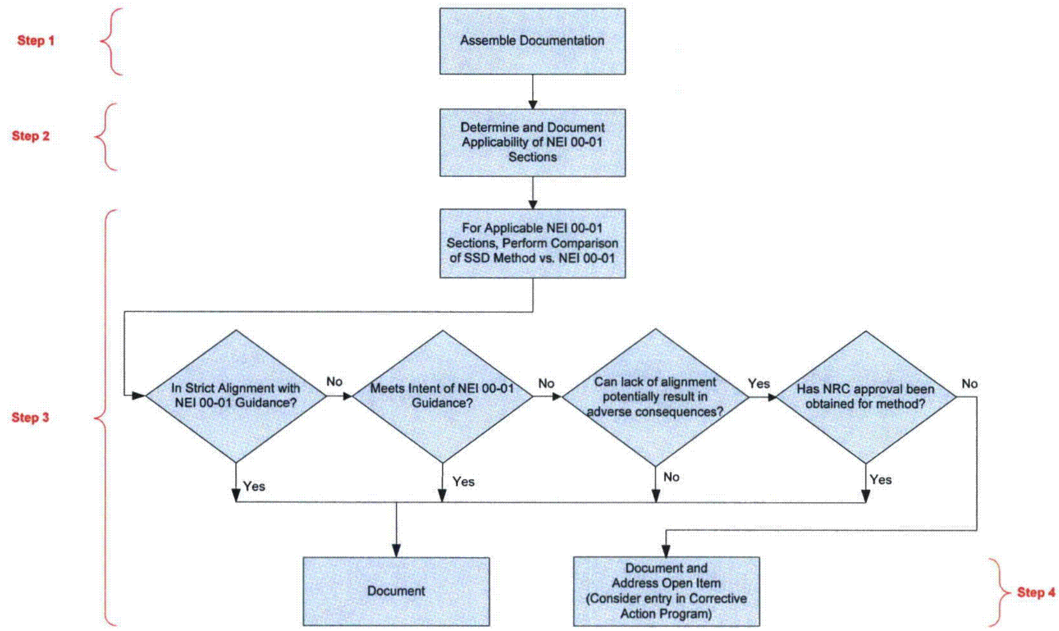

Figure 4-2 – Summary of Nuclear Safety Methodology Review Process (FAQ 07-0039)

4.2.1.2 Safe and Stable Conditions for the Plant

Overview of Process

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

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

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

a pressurized water reactor. For all other configurations, safe and stable conditions are defined as maintaining Keff <0.99 and fuel coolant temperature below boiling.

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

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

Results

Based on CNS Calculation NEDC 11-019, "Nuclear Safety Capability Assessment" (NSCA), the proposed NFPA 805 licensing basis for CNS is to ensure the plant can achieve and maintain the fuel in a "safe and stable" condition assuming a fire occurs during Mode 1 (Power Operation), Mode 2 (Startup), or Mode 3 (Hot Shutdown). The Mode 3 applicability for the analysis is defined as being up to the point at which the direct current (DC) breaker for the shutdown cooling suction valve is un-locked and closed, at which point spurious operation of these high/low pressure interface valves can occur due to fire damage to the valve control circuitry.

As part of the transition to NFPA 805, each fire area was evaluated for maintaining safe and stable hot shutdown conditions for a 24 hour coping period through either a "Deterministic Approach" (NFPA 805 Section 4.2.3) or "Performance Based Approach" (NFPA 805 Section 4.2.4). During the 24 hour coping period, the necessary systems and equipment have been evaluated to ensure their capability to achieve a safe and stable fuel condition as described within NFPA Section 1.3 and 1.6.56.

For the most limiting fire scenarios, the NSCA documents the availability of long term decay heat removal provided by water from the torus, with temperature maintained by the Residual Heat Removal (RHR) system operating in the suppression pool cooling mode. Initiation of the RHR system in the suppression pool cooling mode does not imply that the plant would then proceed all the way to cold shutdown. Following stabilization at hot shutdown, a long term strategy for reactivity control, decay heat removal, and inventory/pressure control would be determined based on the extent of equipment damage. If an assessment of the post-fire conditions indicated that placing RHR in the shutdown cooling mode would be advisable, then repair activities would commence, if necessary, in a safe and controlled manner to restore plant equipment necessary for reactor cooldown.

The mitigation strategies, damage assessment procedures, and repair of equipment to maintain safe and stable conditions are to be re-established within the scope of the NFPA 805 program. Based on the initial achievement of safe and stable conditions in conjunction with the availability of procedural actions, repair of equipment, and initial coping period, recovery of NSCA equipment that may be required beyond 24 hours is qualitatively deemed to have no significant measurable contribution to risk based on the following factors:

- Offsite resources (e.g., equipment, power, vehicles) that could be made available as backups to primary methods of prevention and mitigation.
- The expertise of the Technical Support Center and Emergency Operations Facility staff that would be in place and additional expertise that could be made available by phone or transported to these facilities.

- Existing Emergency Operating Procedures and other emergency response procedures in place to assist the plant operating staff with an option to proceed and implement actions and/or repairs for the plant to transition to, and enter, Mode 4 (Cold Shutdown), if necessary.
- Additional procedures that can be written and reviewed to perform alignments and equipment usage that may be beneficial and are not part of current plant practices or training. Such procedures and equipment usage can cover such a wide spectrum of possibilities that it is judged not useful to develop all possible contingencies at this time.

The above strategies with supporting procedures provides reasonable assurance that the fuel will be maintained in a safe and stable condition following a fire. The revision to existing procedural guidance is to be completed as part of implementation activities (Attachment S, Item S-3.3). Any changes that need to be made during implementation will be resolved using the change evaluation process.

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

- At-Power analysis, Modes 1-3. This analysis is discussed in Section 4.2.4.
- Non-Power analysis, which includes portions of Mode 3, Mode 4 and Mode 5. This analysis is discussed in Section 4.3.

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 operator manual actions (OMA) as recovery actions in the LAR (Regulatory Position 2.2.1 and NEI 04-02, Section 4.6). As a minimum, NEI 04-02 suggests that the assumptions, criteria, methodology, and overall results be included for the NRC to determine the acceptability of the licensee's methodology.

The discussion below provides the methodology used to transition pre-transition OMA and to determine the population of post-transition recovery actions. This process is based on FAQ 07-0030 (Ref. 55) 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 Control Room are not considered pre-transition OMA). Activities that take place at primary control station(s) or in the 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 (VFDR) (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 CNS Calculation NEDC 11-020. 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 CNS is summarized below.

As part of the NFPA 805 transition project, a review and evaluation of CNS 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 (Ref. 56). The Boiling Water Reactor (BWR) Generic MSO list dated May 2009 was utilized.

The approach outlined in Figure 4-3 (based on Figure XX 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. 2 Section F.4.1).
- Updating the Fire PRA model and NSCA to include the MSOs of concern.
- Evaluating for NFPA 805 Compliance.
- Documenting Results.

This process is intended to support the transition to a new licensing basis. Post-transition changes will use the RI-PB change process. The post-transition change process for the assessment of a specific MSO is a simplified version of this process, and does 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).

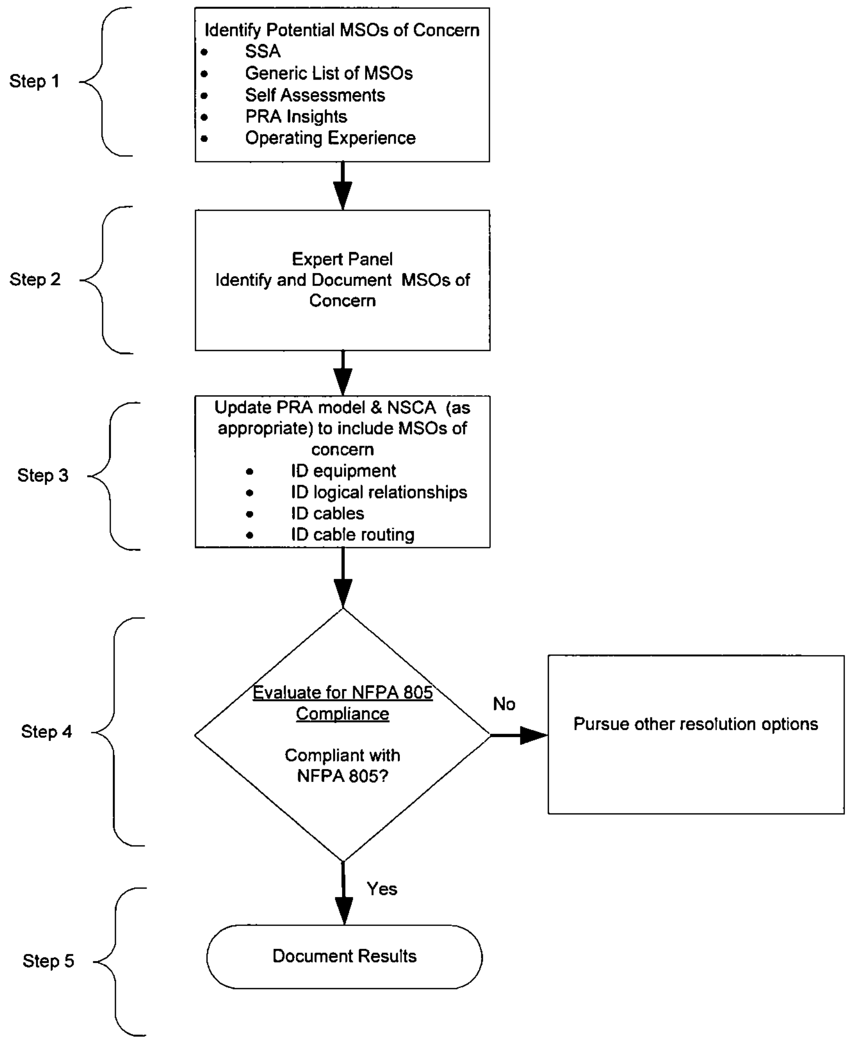

Figure 4-3 – Multiple Spurious Operations – Transition Resolution Process (Based on FAQ 07-0038)

Results

Refer to Attachment F for the process used by CNS 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 included the following determinations:

- The EEEE is not based solely on quantitative risk evaluations,

- The EEEE is an appropriate use of an engineering equivalency evaluation,
- The EEEE is of appropriate quality,
- The standard license condition is met,
- The EEEE is technically adequate,
- The EEEE reflects the plant as-built condition, and
- The basis for acceptability of the EEEE remains valid

In accordance with the guidance in RG 1.205, Regulatory Position 2.3.2, NEI 04-02, as clarified by FAQ 08-0054, "Demonstrating Compliance with Chapter 4 of NFPA 805" (Ref. 57), EEEEs that demonstrate that a fire protection system or feature is "adequate for the hazard" are summarized in the LAR as follows:

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

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

Results

The review results for EEEEs are documented by fire area in NEDC 12-008.

In accordance with the guidance in RG 1.205, Regulatory Position 2.3.2, and NEI 04-02, as clarified by FAQ 08-0054, Revision 1, EEEEs used to demonstrate compliance with Chapters 3 and 4 of NFPA 805 are referenced in the Attachments A and C as appropriate. In addition, none of the transitioning EEEEs require NRC approval.

4.2.3 Licensing Action Transition

Overview of Evaluation Process

The existing licensing actions (exemption requests / 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 (see Section 2.2):

- Exemption for lack of twenty foot separation between redundant Service Water Pumps in the Intake Structure.

  This is no longer required for transition because a RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

- Exemption for lack of twenty foot separation free of intervening combustibles or one-hour barriers between redundant trains in the Cable Spreading Room.

  This exemption is no longer required for transition because a RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

- Exemption for lack of twenty foot separation or one-hour barriers between redundant trains in the Cable Expansion Room.

  This exemption is no longer required for transition because a RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

- Exemption from the requirement for one-hour rated fire barriers for redundant conduits and area wide automatic suppression system in the Reactor Building northeast corner, 903'-6" Elevation.

  This exemption is no longer required for transition because a RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

- Exemption from the requirement for an automatic suppression system in the Control Building Basement, 903'-6" Elevation.

  This exemption is no longer required for transition because a RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

- Exemption from a fixed fire suppression system in the Auxiliary Relay Room.

  This exemption is no longer required for transition because a RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

- Exemption from a fixed fire suppression system in the Control Room.

  This exemption is no longer required for transition because a RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

- Exemption from the requirement of a three-hour fire barrier, or twenty feet of separation free of intervening combustibles combined with automatic suppression and detection in the 931' Elevation of the Reactor Building.

  This exemption is no longer required for transition because a RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

- Exemption from the requirement of a three-hour fire barrier, or twenty feet of separation free of intervening combustibles combined with automatic suppression and detection, and lack of alternate shutdown capability independent of the area in the 903'-6" Elevation of the Reactor Building (excluding the northeast corner).

  This exemption is no longer required for transition because a RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

- Exemption from the requirement of a three-hour fire barrier, or twenty feet of separation free of intervening combustibles combined with automatic suppression and detection, and lack of alternate shutdown capability independent of the area in the 859' and 881' Elevations of the Reactor Building.

  This exemption is no longer required for transition because a RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

Since the exemptions are no longer necessary, in accordance with the requirements of 10 CFR 50.48(c)(3)(i), NPPD requests that the exemptions listed in Attachment K be rescinded as part of the LAR process. It is NPPDs understanding that implicit in superseding the current license condition, all prior Fire Protection Program Safety Evaluation Reports and commitments will be superseded in their entirety. See Attachment O, Orders and Exemptions.

4.2.4 Fire Area Transition

Overview of Evaluation Process

The Fire Area Transition (NEI 04-02 Table B-3) was performed using the methodology contained in NEI 04-02 and FAQ 08-0054, Revision 1. 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 each fire area.
- Documented evaluation of effects of fire suppression activities. Documented the evaluation of the effects of fire suppression activities on the ability to achieve the nuclear safety performance criteria.
- Performed licensing action reviews. Performed a review of the licensing aspects of the selected fire area and document the results of the review. See Section 4.2.3.
- Performed existing engineering equivalency evaluation reviews. Performed a review of existing engineering equivalency evaluations (or create new evaluations) documenting the basis for acceptability. See Section 4.2.2.
- Pre-transition OMA reviews. Performed a review of pre-transition OMAs to determine those actions taking place outside of the Control Room or outside of the primary control station(s). See Section 4.2.1.3.

Step 3 – VFDR Identification and characterization and resolution considerations. Identified variances from the deterministic requirements of NFPA 805, Section 4.2.3. Documented

variances as either a separation issue or a degraded fire protection system or feature. Developed VFDR problem statements to support resolution.

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

Step 5 – Final Disposition.

- Documented final disposition of the VFDR 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.

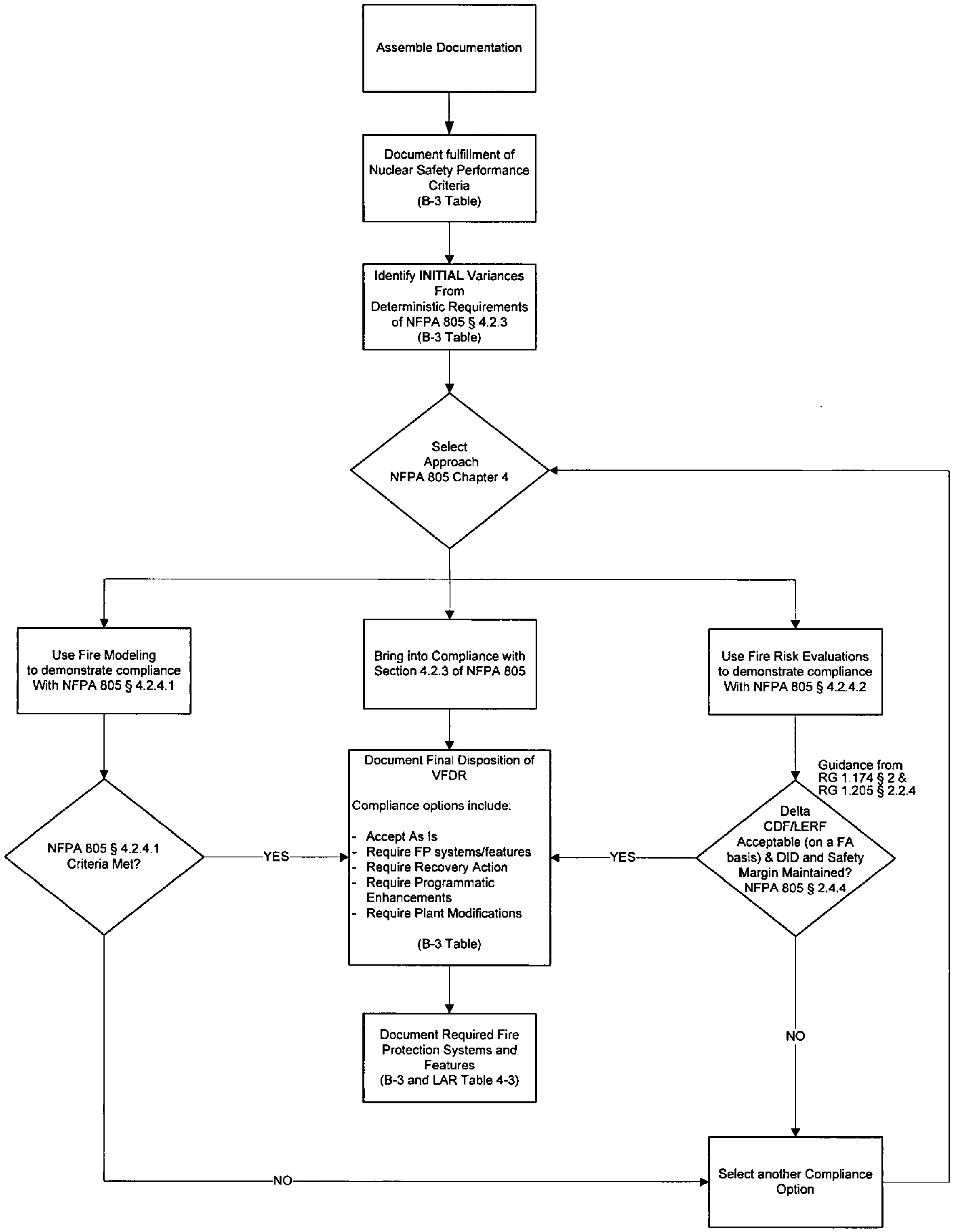

Figure 4-4 – Summary of Fire Area Review
[Based on FAQ 08-0054 Revision 1]

Results of the Evaluation Process

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

Attachment C, Table B-3 includes the following summary level information for each fire area:

- Regulatory Basis – NFPA 805 post-transition regulatory bases are included.
- Performance Goal Summary – An overview of the method of accomplishment of each of the performance criteria in NFPA 805 Section 1.5 is provided.
- Reference Documents – Specific references to Nuclear Safety Capability Assessment Documents are provided.
- Licensing Actions – Specific references to exemption requests that will remain part of the post-transition licensing basis are provided, when applicable. A brief description of the condition and the basis for acceptability of the licensing action is provided.
- EEEE – Specific references to EEEEs that rely on determinations of "adequate for the hazard" that will remain part of the post-transition licensing basis. A brief description of the condition and the basis for acceptability is provided.
- VFDR – Specific variances from the deterministic requirements of NFPA 805 Section 4.2.3 are provided, when applicable. 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

CNS implemented the process outlined in NEI 04-02 and FAQ 07-0040, "Clarification on Non-Power Operations" (Ref. 58). The goal (as depicted in Figure 4-6) is to ensure that contingency plans are established when the plant is in an 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 involves the following steps:

- Review the existing Outage Management Processes.
- Identify Equipment/Cables:
  - Review plant systems to determine success paths that support each of the defense-in-depth Key Safety Functions (KSFs), and
  - Identify cables required for the selected components and determine 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.

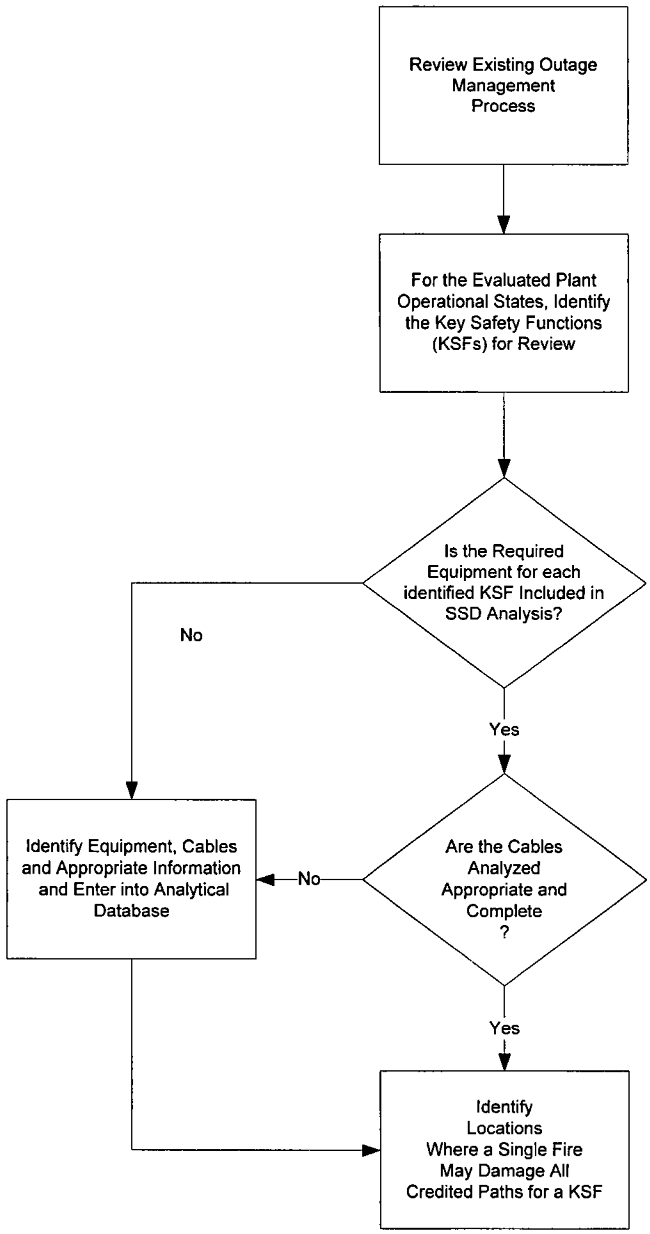

Figure 4-5 Review KSF, Equipment and Cables, and Identify Pinch Points

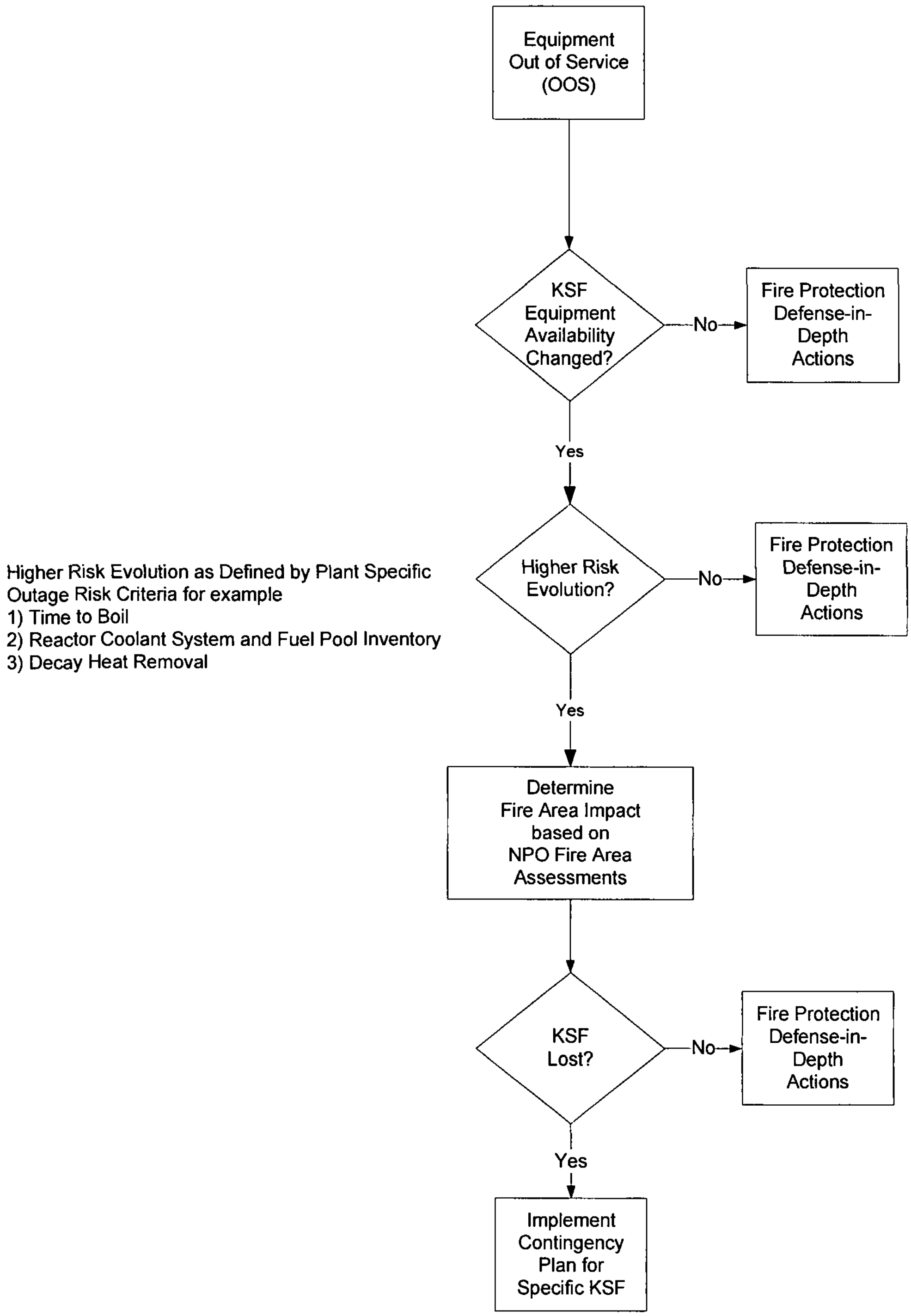

Figure 4-6 Manage Pinch Points

4.3.2 Results of the Evaluation Process

Based on FAQ 07-0040 (Revision 4), the Plant Operating States (POS) considered for equipment and cable selection are defined in Calculation NEDC 11-003 "Non-Power Operation Modes Transition Review". Components were identified to support the KSFs of Inventory Control, Decay Heat Removal Capability, and associated support functions (process cooling and electrical power). A model was developed in the NFPA 805 Analysis Database (Genesis Solution Suite, SAFE Module). Equipment was logically tied to the supported KSF. Power supplies, interlocks, and supporting equipment were logically tied to their parent component.

For those components which had not been previously analyzed in support of the at-power analysis or whose functional requirements may have been different for the non-power analysis, cable selection was performed in accordance with approved project procedures. Cables necessary to support the selected function of a component were selected and analyzed for fire impact.

CNS Calculation NEDC 11-003 contains the fire area assessment, the identified pinch points, and general recommendations for administrative controls to reduce that fire risk as well as a proposed strategy for recovering the KSF should a fire occur. In accordance with FAQ 07-0040, any area experiencing fire damage which eliminates all success paths for a KSF (without recovery actions outside the Control Room) is considered a pinch point. Fire modeling was not used to eliminate any fire area from being a pinch point.

The list of generic recommendations specified in NEDC 11-003 considers the following actions from FAQ 07-0040:

- Prohibition or limitation of hot work in fire areas during periods of increased vulnerability
- Verification of operable detection and/or suppression in the vulnerable areas
- Prohibition or limitation of combustible materials in fire areas during periods of increased vulnerability
- Use of 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)
- Rescheduling of work to a period with lower risk or higher defense-in-depth

Refer to Attachment D for additional details. Based on consideration of the vulnerable areas and incorporation of generic recommendations from FAQ 07-0040 into appropriate plant procedures and practices, prior to implementation of NFPA 805 (See Implementation Item S-3.4 of Attachment S, Table S-3), the performance goals (KSFs) for Non-Power Operations will be fulfilled and the requirements of NFPA 805 will be met.

4.4 Radioactive Release Performance Criteria

4.4.1 Overview of Evaluation Process

The review of the FPP against NFPA 805 requirements for fire suppression related radioactive release was performed using the methodology contained in NEI 04-02 and subsequent

guidance provided in NFPA 805 Task Force FAQ 09-0056 (Ref. 59). The methodology consisted of the following:

- A review to "screen-out" fire zones based on the lack of potential for 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 CNS pre-fire plans. The evaluation focused on radioactive release to any unrestricted area due to fire fighting activities only; radioactive release due to potential fuel cladding damage was not evaluated. The nuclear safety goal, nuclear safety objectives, and nuclear safety performance criteria specified in NFPA 805 require the prevention of fuel cladding damage. As such, radiological release due to fuel damage does not require a separate examination since no such damage is assumed to occur without violating the basic requirements of NFPA 805.
- A review of pre-fire plans and fire brigade training materials to identify FPP elements (e.g., systems / components / procedural control actions / flow paths, etc.) that are being credited to meet the radioactive release goals, objectives, and performance criteria during all plant operating modes, including full power and non-power conditions.
- A review of engineering controls to ensure containment of gaseous and liquid effluents (e.g., smoke and fire fighting agents). This review included all plant operating modes (including full power and non-power conditions). Otherwise, a bounding analysis, quantitative analysis, or other analysis that demonstrates that the limitations for instantaneous release of radioactive effluents specified in the CNS Technical Specifications are met was provided.

4.4.2 Results of the Evaluation Process

CNS Calculation NEDC 10-062, "NFPA 805 Radioactive Release Review," details the results of the screening process and review of pre-fire plans, fire brigade training materials, and engineering controls.

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

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

4.5 Fire PRA and Performance-Based Approaches

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

- A Fire PRA (discussed in Section 4.5.1 and Attachments 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 CNS in compliance with the requirements of Part 4 "Internal Fires at Power Probabilistic Risk Assessment Requirements," of the American Society of Mechanical Engineers (ASME) and American Nuclear Society (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" (Ref. 60), (hereafter referred to as Fire PRA Standard). CNS conducted a

peer review by independent industry analysts in accordance with RG 1.200 (Ref. 61), prior to a risk-informed submittal. The peer review validated the fire risk assessment model used as the analytical tool to perform Fire Risk Evaluations during the transition process.

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

4.5.1.1 Internal Events PRA

The CNS base Internal Events PRA (CNS PRA Model 2007TM-R3) was the basis for the Fire PRA.

The CNS Internal Events PRA has undergone a self-assessment by a team of knowledgeable CNS PRA personnel and consultants. In addition, an independent peer review of the CNS Internal Events PRA was completed. The peer review was done in accordance with ASME Standard Capability Category II requirements. The peer review concluded that CNS PRA fully meets the Capability Category II requirements for 289 of the 301 applicable ASME PRA Standard Supporting Requirements (SRs), as modified by RG 1.200, Revision 1. Each of the remaining 12 open SRs, were addressed after completion of the peer review and none of these 12 are expected to impact the technical adequacy of the PRA for supporting NFPA 805. Attachment U details the 12 open SRs, the actions taken to address them, and the impact on the NFPA 805 application.

4.5.1.2 Fire PRA

A Fire PRA model was developed for CNS using the guidance provided in NUREG/CR-6850/EPRI TR-1011989 (including supplement 1), EPRI TR-1016735 (Ref. 62), and draft NUREG-1921 (Ref. 63). Attachment H provides a listing of the approved FAQs that affect the overall license transition process for CNS. The resulting fire risk assessment model is used to support Fire Risk Evaluations during the transition process and to develop estimates of the potential change in fire related risk.

The Fire PRA was developed using the Internal Events PRA as a starting point. The Internal Events PRA was modified to model the effects of fire, both as an initiator and the subsequent potential failure modes for affected circuits or targets. The Fire PRA has been quantified using the CAFTA PRA software. The CNS Fire PRA is documented in a series of reports and calculations associated with each NUREG/CR-6850 Fire PRA task.

An independent peer review was conducted in April 2010 that included a review of the PRA model, data, and documentation in accordance with ASME Standard Capability Category II requirements as well as an assessment of the impacts on the NFPA 805 program. A follow on focused peer review was conducted in February 2011 to specifically address multi-compartment analyses and final results quantification.

Fire PRA quality and insights are discussed in subsequent sections and in Attachments V and W, respectively.

Fire Model Utilization in the Application

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

The following fire models were used:

- Flame Height (Method of Heskestad)
- Plume Centerline Temperature (Method of Heskestad)
- Radiant Heat Flux (Point Source Method)
- Plume Radius (Method of Heskestad)
- Hot Gas Layer (Method of McCaffrey, Quintiere, and Harkleroad)
- Hot Gas Layer (Method of Beyler)
- Hot Gas Layer (Method of Foote, Pagni, and Alvares)
- Hot Gas Layer (Method of Deal and Beyler)
- Ceiling Jet Temperature (Method of Alpert)
- Hot Gas Layer Calculations using Consolidated Model of Fire Growth and Smoke Transport (CFAST)
- Smoke Detection Actuation Correlation (Method of Heskestad and Delichatsios)
- Heat Detection Actuation Correlation
- Sprinkler Activation Correlation
- Control Room Abandonment Calculation using Fire Dynamics Simulator (FDS)
- Temperature Sensitive Equipment Hot Gas Layer Study using CFAST
- Temperature Sensitive Equipment Zone of Influence Study using FDS
- Plume/Hot Gas Layer Interaction Study using FDS
- Corner and Wall Heat Release Rate
- Correlation for Heat Release Rates of Cables (Method of Lee)
- Correlation for Flame Spread over Horizontal Cable Trays (FLASH-CAT)

The acceptability of the use of these fire models is included in Attachment J. Many of these correlations have been built into a Fire Modeling Workbook utilized for the Fire PRA fire modeling. The Fire Modeling Workbook was verified, by "black box" testing, to ensure that the results were identical to the verified and validated models. "Black box" testing (also called functional testing) is testing that ignores the internal mechanism of a system or component and focuses solely on the outputs generated in response to selected inputs and execution conditions.

The process compared results from the Fire Modeling Workbook to those produced by the NUREG-1805 Fire Dynamic Tools and Fire-Induced Vulnerability Evaluation when identical inputs were entered into both. Since the correlations from NUREG-1805 Fire Dynamic Tools and Fire-Induced Vulnerability Evaluation, Revision 1, were verified and validated in NUREG-1824, and the results match the results produced by the Fire Modeling Workbook, by the transitive property, the Fire Modeling Workbook is verified and validated with respect to NUREG-1824.

The results of this verification are documented in NEDC 10-020, "Verification and Validation of Fire Modeling Tools and Approaches for Use in NFPA 805 and Fire PRA Applications."

4.5.1.3 Results of Fire PRA Peer Review

The CNS Fire PRA was peer reviewed against the requirements of ASME/ANS RA-Sa-2009, Part 4. The BWR Owners Group issued the final report in March 2011 containing the results of the peer review. The identification and resolution of the findings are summarized in Attachment V.

All of the findings have either been formally addressed or evaluated to have no significant impact on the Fire PRA and fire risk evaluations. There were 49 SR identified that did not meet Capability Category II. These are summarized in Attachment V with a CNS resolution for this application.

4.5.1.4 Risk Insights

Risk insights were documented as part of the development of the Fire PRA. The total plant fire core damage frequency (CDF)/large early release frequency (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 scenarios that individually contribute more than 1% of the calculated fire risk is 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

In lieu of the fire modeling approach the fire risk evaluation approach was utilized for the transition.

4.5.2.2 Fire Risk Approach

Overview of Evaluation Process

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

Table 4-1 Fire Risk Evaluation Guidance Summary Table

Document	Section(s)	Topic
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 (VFDR).

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 (see Figure 4-7):

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 included a Safe Shutdown/NSCA Engineer, Fire Protection Engineer, and Fire PRA Engineer. The purpose and objective of this team review was to address the following:
  - Review Fire PRA modeling treatment of VFDR
  - Ensure discrepancies were captured and resolved

Step 2 – Performed the Fire Risk Evaluation

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

Step 3 – Reviewed the Acceptance Criteria

- The acceptance criteria for the Fire Risk Evaluation consist of two parts. One is quantitatively based and the other is qualitatively based. The quantitative figures of merit are ΔCDF and ΔLERF. The qualitative factors are defense-in-depth and safety margin.
  - Risk Acceptance Criteria. The transition risk evaluation was measured quantitatively for acceptability using the ΔCDF and ΔLERF criteria from RG 1.174, as clarified in RG 1.205 Regulatory Position 2.2.4.

- Defense-in-Depth. A review of the impact of the change on defense-in-depth was performed, using the guidance NEI 04-02. NFPA 805 defines defense-in-depth as:

  - Preventing fires from starting.
  - Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting damage.
  - Providing adequate level of fire protection for structures, systems and components important to safety; so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

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

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

  Fire protection features and systems relied upon to ensure defense-in-depth were identified as a result of the assessment of defense-in-depth.

- Safety Margin Assessment. A review of the impact of the change on safety margin was performed. Guidelines for making that assessment are summarized below.

  - Codes and standards or their alternatives accepted for use by the NRC are met, and
  - Safety analysis acceptance criteria in the licensing basis are met, or provides sufficient margin to account for analysis and data uncertainty.

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

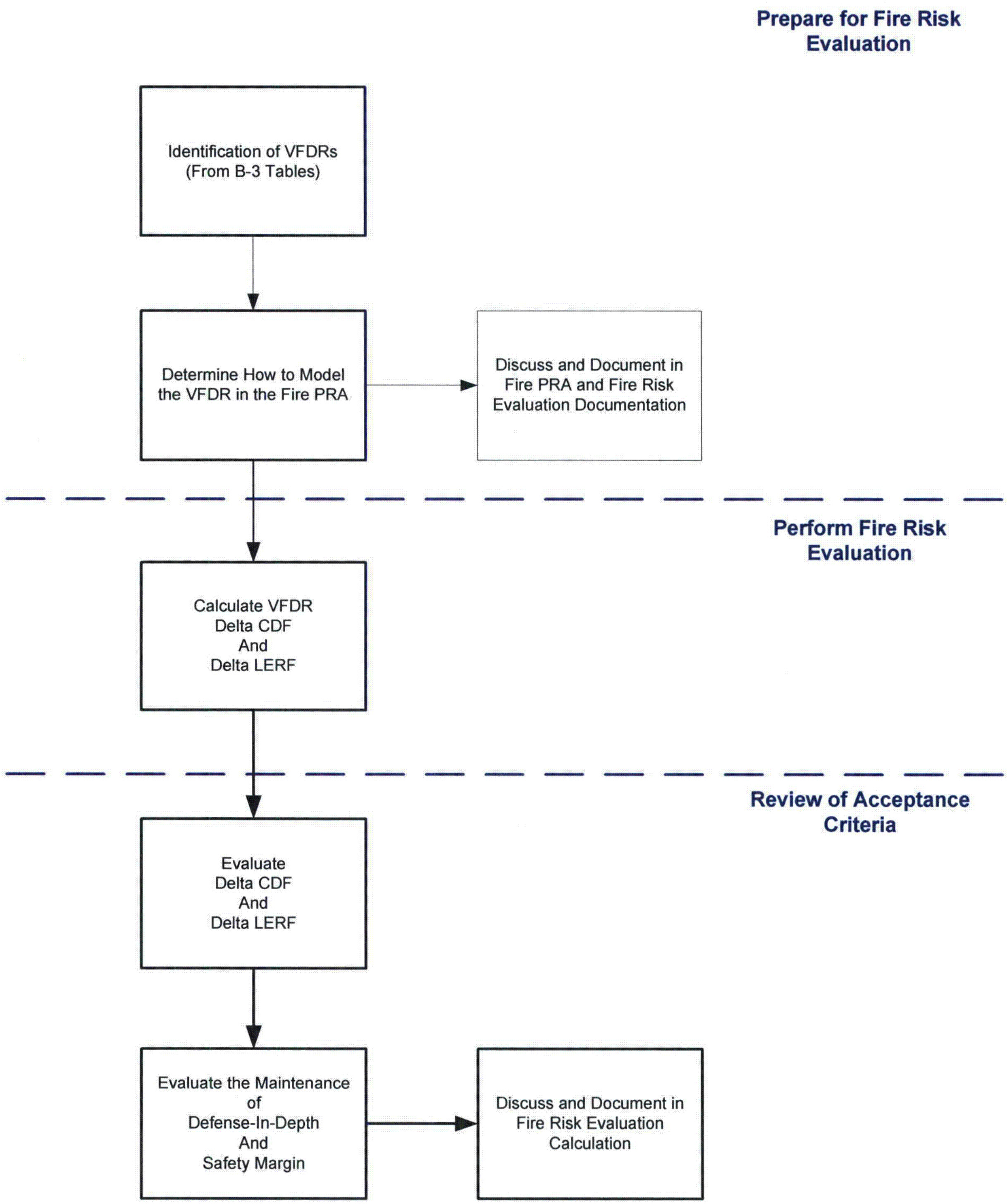

Figure 4-7 – Fire Risk Evaluation Process (NFPA 805 Transition) [Based on FAQ 08-0054 Revision 1]

Results of Evaluation Process

Disposition of VFDR

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

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

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

Risk Change Due to NFPA 805 Transition

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

RG 1.205, Section C.2.2.4.2 states:

> *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

NFPA 805 Section 3.2.3(3) requires that procedures be established for reviews of the fire protection program related performance and trends. NFPA 805, Section 2.6 requires a monitoring program that in part is to establish acceptable performance levels and a method to monitor and assess the performance of the fire protection program. The NFPA 805 requirements for reviews of programs related to performance and trending is provided under the NFPA 805 Monitoring program.

The NFPA 805 Monitoring Program, as described in this Section, will be implemented within six months after issuance of the NFPA 805 Transition License Amendment (see Attachment S, Table S-3, Implementation Item S-3.23). In order to assess the impact of the transition to NFPA 805 on the current monitoring program, the CNS fire protection program documentation such as the maintenance program processes, FPP implementing procedures, and plant change processes will be reviewed. Sections 4.5.3 and 5.2 of the NEI 04-02, as modified by FAQ 10-

0059 (Ref. 64), will be used during the review process and that process is described in the following sections.

The following scope will be documented appropriately:

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

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 structure, systems and components within existing plant programs,
- The performance criteria for the availability and reliability of the required structure, systems and components.
- 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 process will be 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 an overview of the Monitoring Process, while Figure 4-9 provides detail on a process for Phases 1 and 2.

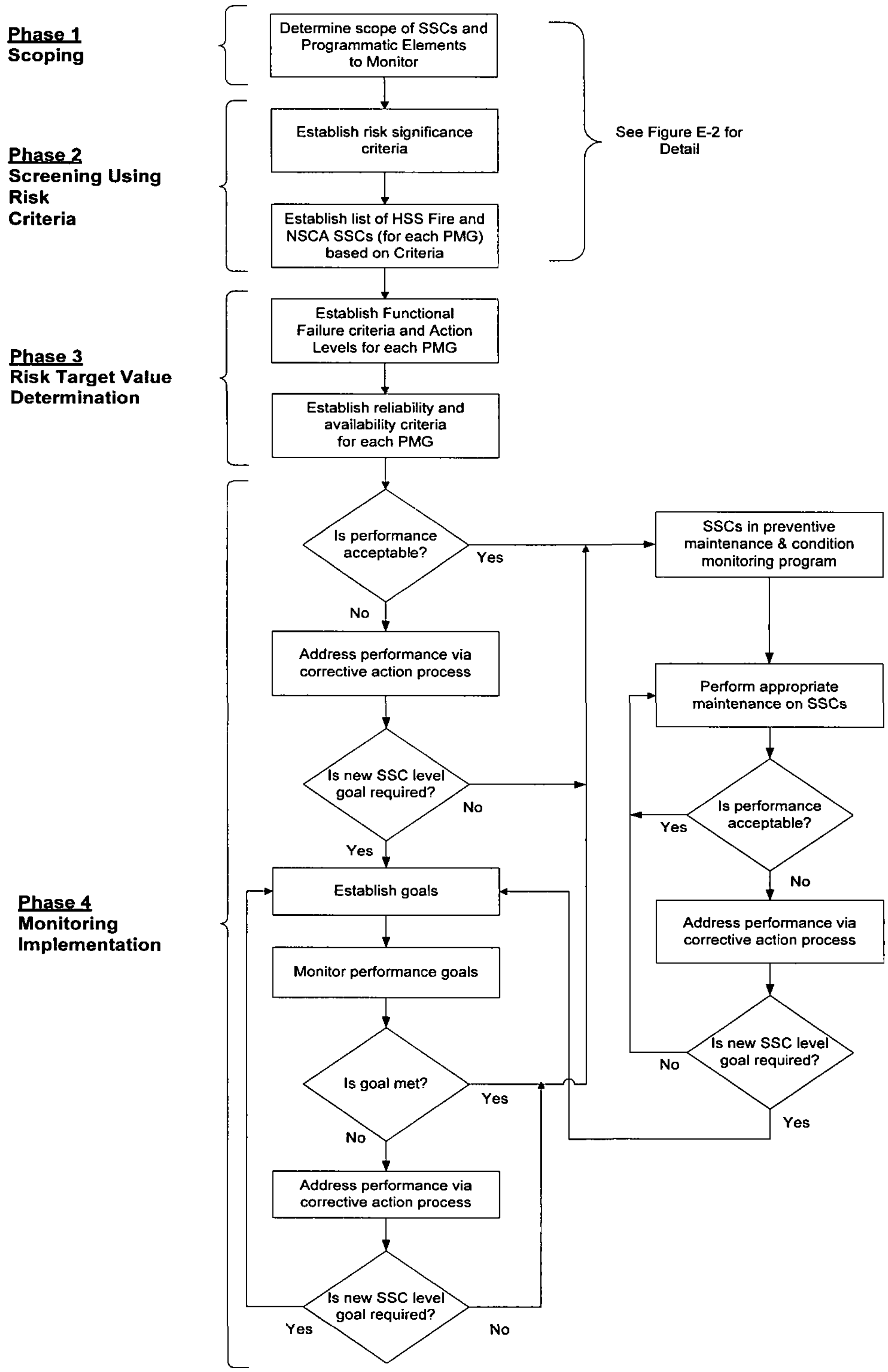

Figure 4-8 – NFPA 805 Monitoring Process [Based on FAQ 10-0059 Rev. 5, Table E-1]

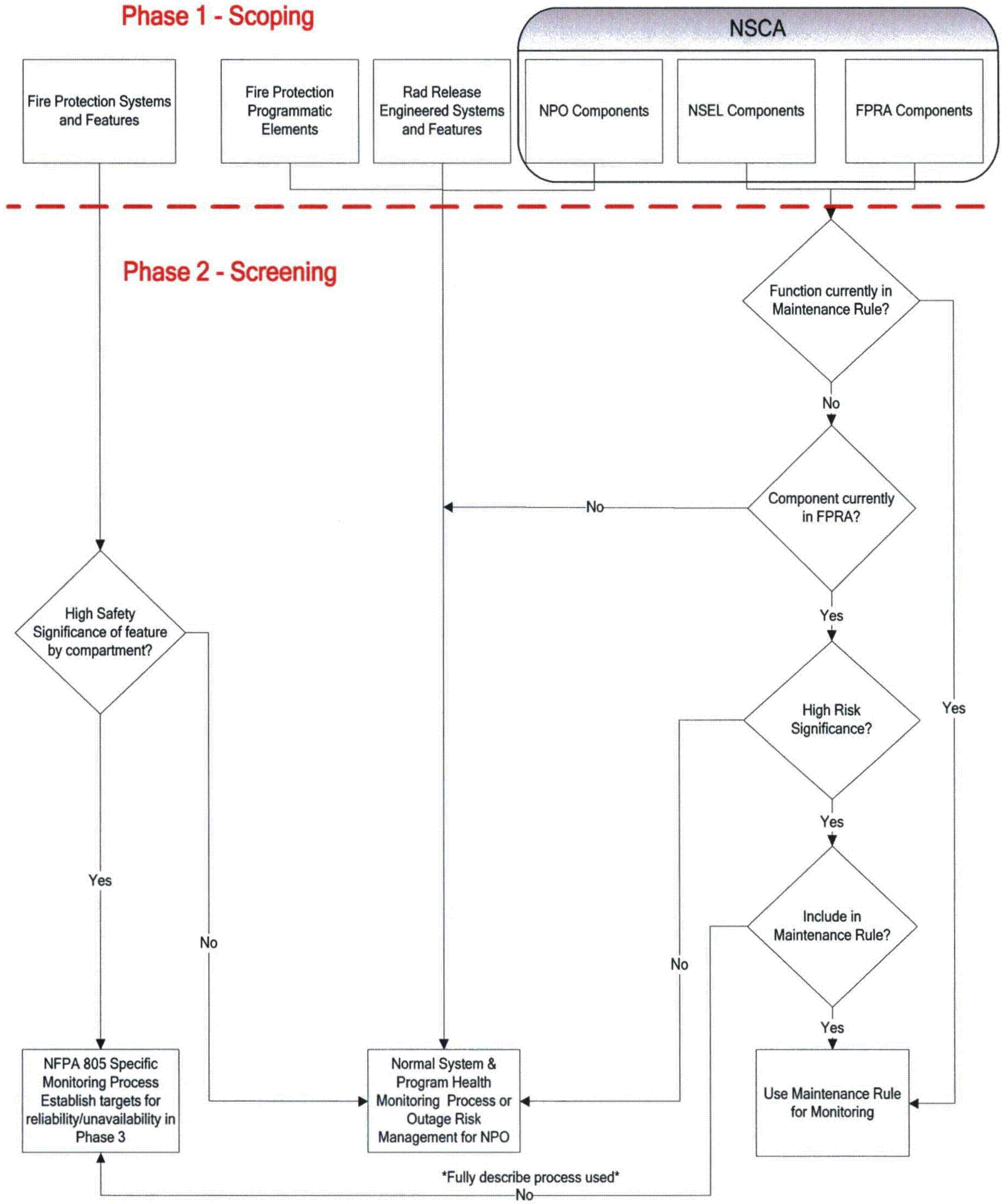

Figure 4-9 – NFPA 805 Monitoring – Scoping and Screening
[Based on FAQ 10-0059 Rev. 5, Table E-2]

Phase 1 – Scoping

In order to meet the NFPA 805 requirements for monitoring, the following categories of SSC and programmatic elements will be included in the NFPA 805 monitoring program:

- Structures, Systems, and Components required to comply with NFPA 805, specifically:
    - Fire protection systems and features
        - Required by the Nuclear Safety Capability Assessment
        - Modeled in the Fire PRA
        - Required by Chapter 3 of NFPA 805
    - Nuclear Safety Capability Assessment equipment[6]
        - 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 SSC 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 SSC 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 surveillance activities.

1. Fire Protection Systems and Features

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

Risk significance will be accomplished at the component, programmatic element, and/or functional level. Since risk will be evaluated at the compartment level or fire area level, criteria must be developed to determine those analysis units for which the fire protection SSCs contained within the area are considered risk significant. Screening compartments and fire areas will also include considerations for design/operation/maintenance limitations. For instance, fire detection will not subdivide systems beyond the system/train/channel level used in normal operation/maintenance.

The Fire PRA is the primary tool used to establish the risk significance criteria and performance bounding guidelines. Screening thresholds used to determine risk significant analysis units will be those that meet the following criteria:

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

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[6] For the purposes of the NFPA 805 Monitoring, "NSCA equipment" is intended to include Nuclear Safety Equipment, Fire PRA equipment, and NPO equipment.

(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 $RAW_{(monitored\ parameter)}$ will be 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). If compartments are used that are smaller than fire areas, sufficient basis will be documented.

The monitoring program will include the appropriate Fire Protection Program SSC based on the criteria above. Additional Fire Protection Program SSC may also be screened in based on plant-specific considerations.

2. Nuclear Safety Capability Assessment Equipment

NSCA equipment may already be appropriately monitored by the Maintenance Rule. A comparison of NSCA equipment to the SSCs that are monitored in the Maintenance Rule program will be performed to determine what equipment may require additional NFPA 805 Monitoring. For NSCA SSC not monitored by the Maintenance Rule, the basis for inclusion or exclusion of the SSC in the NFPA 805 monitoring program will be documented.

The Fire PRA will be used to identify high-safety-significant (HSS) NSCA SSC that require monitoring. The Maintenance Rule guidelines differentiating HSS from low-safety-significant (LSS) SSC will be used. HSS NSCA SSC not currently monitored in Maintenance Rule will be included in the Maintenance Rule program. NSCA SSC that are not HSS will be 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 SSC relied upon to meet the radioactive release performance criteria are not amenable to quantitative risk measurement. Additionally, since 10 CFR Part 20 limits (which are lower than releases due to core damage and containment breach) for radiological effluents are not being exceeded, equipment relied upon to meet the radioactive release performance criteria is considered inherently low risk. Therefore, additional monitoring beyond inspection and test programs and system/program health 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". Programmatic aspects include:

- Transient Combustible Control; Transient Exclusion Zones
- Hot Work Control; Administrative Controls
- Fire Watch Programs; Program compliance and effectiveness

- Fire Brigade Effectiveness

Fire protection health reports, self-assessments, regulator and insurance company reports provide inputs to the monitoring program. The monitoring of programmatic elements and program effectiveness will be performed as part of the management of engineering programs. This monitoring is more qualitative in nature since the programs do not lend themselves to the numerical methods of reliability and availability. These programs form the bases for many of the analytical assumptions used to evaluate compliance with NFPA 805 requirements.

Phase 3 – Risk Target Value Determination

Phase 3 consists of using the Fire PRA, or other processes as appropriate, to determine target values of reliability and availability for the HSS fire protection/NSCA SSC and programmatic elements established in Phase 2 as requiring additional monitoring beyond inspection and test programs and system/program health programs.

Failure criteria will be established by an expert panel or evaluation based on the required fire protection and nuclear safety capability SSC and programmatic elements assumed level of performance in the supporting analyses. Action levels will be established for the SSC at the component level, program level, or functionally through the use of the pseudo system or 'performance monitoring group' concept. An action level will be developed for the NSCA SSC that are included in a monitoring program.

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

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 SSC 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 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 SSC that do not meet performance criteria.

For fire protection and NSCA SSC 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 will be initiated to identify the negative trend. A corrective action plan will then be developed using the appropriate CNS process. Once the plan has been implemented, improved performance will return the SSC back to below the established action level.

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 may 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 nuclear safety capability assessment systems?

- Have the supporting analyses been revised such that the performance criteria are no longer applicable or new fire protection and nuclear safety capability assessment 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?

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, NPPD has documented analyses to support compliance with 10 CFR 50.48(c). The analyses are being performed in accordance with CNS processes for ensuring assumptions are clearly defined, that results are easily understood, that results are clearly and consistently described, and that sufficient detail is provided to allow future review of the entire analyses.

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

The Fire Protection Design Basis Document described in Section 2.7.1.2 of NFPA 805 and necessary supporting documentation described in Section 2.7.1.3 of NFPA 805 will be created as part of transition to 10 CFR 50.48(c) to ensure program implementation following receipt of the safety evaluation. (See Implementation Item S-3.5 of Attachment S, Table S-3). Appropriate cross references will be established to supporting documents as required by CNS processes. Figure 4-10 depicts the planned post-transition documentation and relationships.

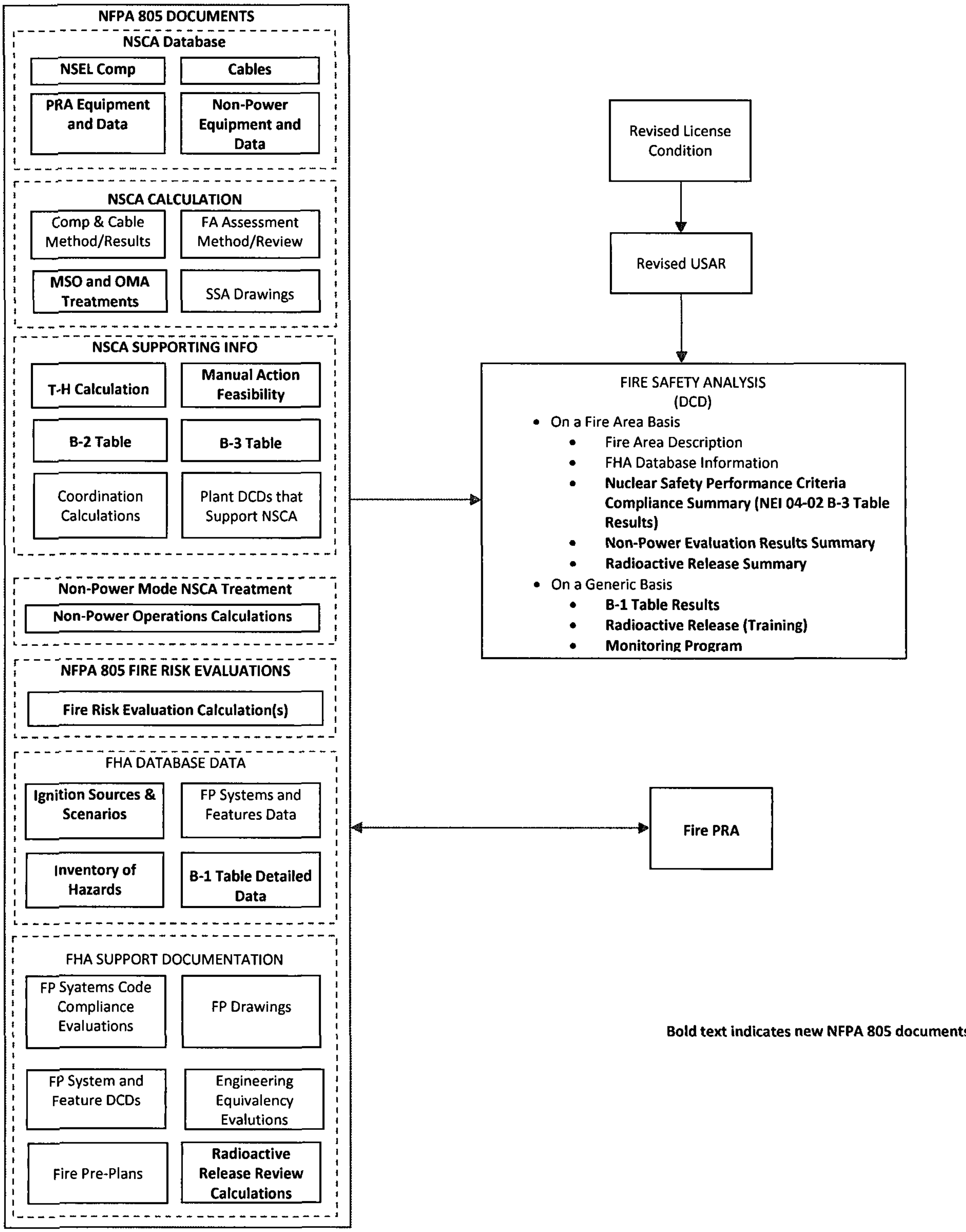

Figure 4-10 – NFPA 805 Planned Post-Transition Documents and Relationships

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

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

Table 4-2 Change Evaluation Guidance Summary Table

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

The Plant Change Evaluation Process required under the revised Fire Protection License Condition will consist of the following 4 steps and is depicted in Figure 4-11:

- 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 will then be performed to identify and resolve minor changes to the fire protection program. This screening will be consistent with fire protection regulatory review processes currently in place at CNS. This screening process will be 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 will be followed by engineering evaluations that may include fire modeling and risk assessment techniques. The results of these evaluations will then be 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 will require that the resultant change in CDF and LERF be consistent with the license condition. The acceptance criteria will also include consideration of defense-in-depth and safety margin, which would typically be qualitative in nature.

The risk evaluation will involve 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 may 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 will be 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 will also be assessed to ensure they are consistent with the defense-in-depth philosophy and that sufficient safety margins are maintained.

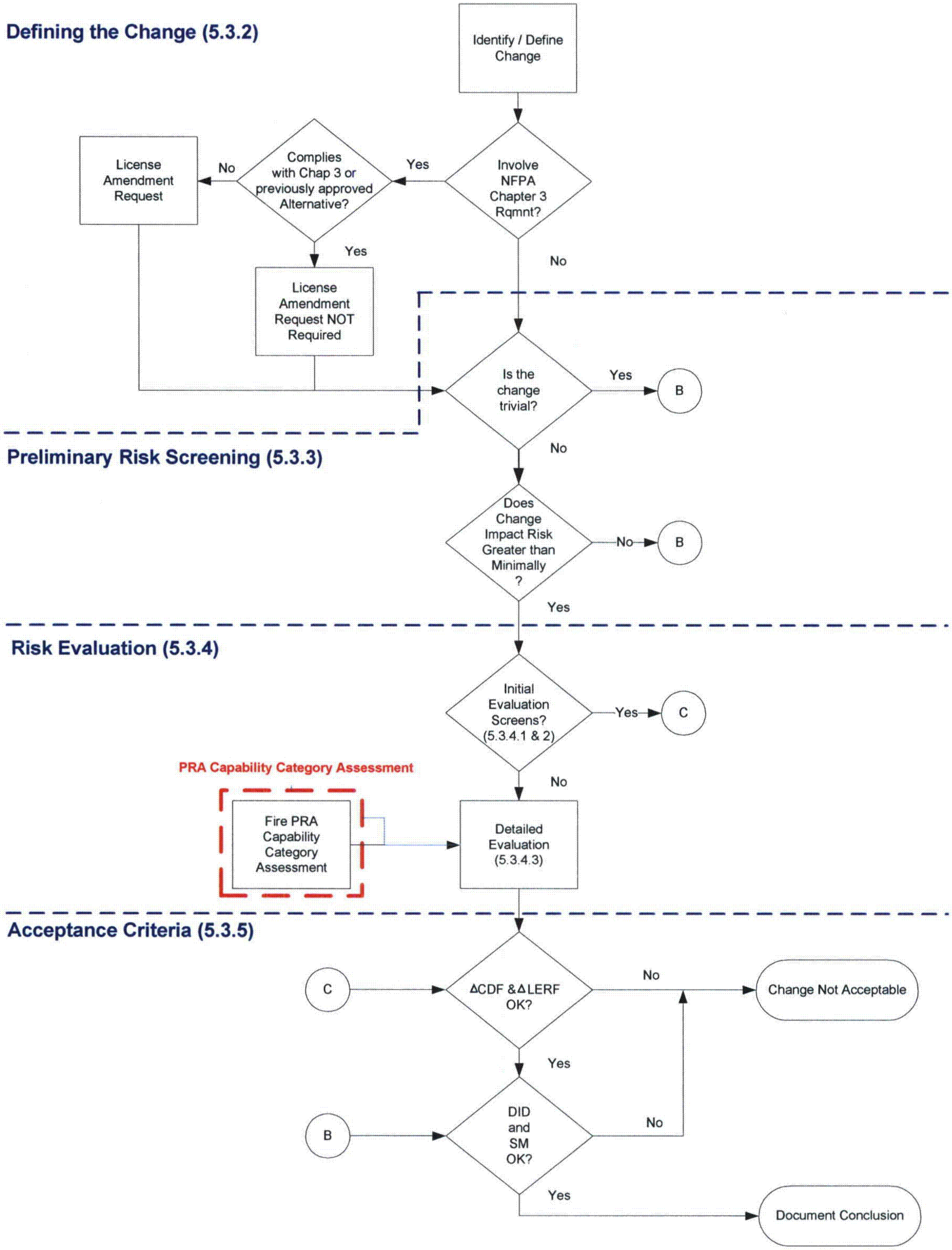

Figure 4-11 Plant Change Evaluation [NEI 04-02 Figure 5-1]
Note references in Figure refer to NEI 04-02 Sections

The CNS 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 CNS documents and databases that currently exist will be revised to reflect the new NFPA 805 licensing bases requirements. See Implementation Item S-3.27 of Attachment S, Table S-3.

Several NFPA 805 document types such as: NSCA supporting information, Non-Power Mode Review, Fire Modeling Calculations, Fire Safety Assessments, risk evaluations, 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. See Implementation Item S-3.28 of Attachment S, Table S-3.

System level design basis documents will be revised to reflect the NFPA 805 role that the system components now play. The new procedures will be developed and existing documentation revised as part of LAR implementation. See Implementation Item S-3.29 of Attachment S, Table S-3.

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 will be 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 program impacts will be sent to qualified individuals (Fire Protection, NSCA, Fire PRA) to ascertain the program impacts, if any. If Fire Protection Program impacts are determined to exist as a result of the proposed change, the issue would be resolved by one of the following:

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

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

4.7.3 Compliance with Quality Requirements in Section 2.7.3 of NFPA 805

Fire Protection Program Quality

NPPD will maintain the existing Fire Protection Quality Assurance program as outlined in the CNS Quality Assurance Program for Operation - Policy Document (QAPD) (Ref. 65), as implemented by CNS Quality Assurance procedures. The FPP procedures will be revised to specify application of the NFPA 805 Section 2.7.3 quality requirements (see Implementation Action S-3.8 of Attachment S, Table S-3).

During the transition to 10 CFR 50.48(c), NPPD performed work in accordance with the quality requirements of Section 2.7.3 of NFPA 805.

Fire PRA Quality

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

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) were performed in accordance with CNS procedures that require independent review.

NFPA 805 Section 2.7.3.2 – Verification and Validation

Calculational models and numerical methods used in support of compliance with 10 CFR 50.48(c) were verified and validated as required by Section 2.7.3.2 of NFPA 805.

NFPA 805 Section 2.7.3.3 – Limitations of Use

Engineering methods and numerical models used in support of compliance with 10 CFR 50.48(c) were 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.

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

For personnel performing fire modeling or Fire PRA development and evaluation, NPPD will develop and maintain qualification requirements for individuals assigned various tasks. Position Specific Guides will be developed to identify and document required training and mentoring to ensure individuals are appropriately qualified per the requirements of NFPA 805 Section 2.7.3.4 to perform assigned work (see Implementation Item S-3.15 of Attachment S, Table S-3).

NFPA 805 Section 2.7.3.5 – Uncertainty Analysis

Uncertainty analyses were performed as required by 2.7.3.5 of NFPA 805 and the results were considered in the context of the application. This is of particular interest in fire modeling and Fire PRA development used to support performance-based approach. However, it is not required for deterministic approach calculations per 10 CFR 50.48(c)(2)(iv).

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 (which is provided in Attachment C):

- 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 (Note: Compliance is determined on a Fire Area basis therefore a compliance basis is not provided for individual fire zones.)
- 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 information is provided on a zone basis. The basis for the requirement of the fire protection system / feature is designated as follows:
  - S – Separation Criteria: Systems required for Chapter 4 Separation Criteria in Section 4.2.3
  - L – Licensing Action Criteria: Systems required for acceptability of NRC approved Licensing Action (i.e., Exemptions/Safety Evaluations) (Section 2.2.7)
  - E – EEEE Criteria: Systems required for acceptability of Existing Engineering Equivalency Evaluations (Section 2.2.7)
  - R – Risk Criteria: Systems required to meet the Risk Criteria for the Performance-Based Approach (Section 4.2.4)
  - D – Defense-In-Depth Criteria: Systems 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, training, programs, procedure changes, and evaluations to comply with NFPA 805 are described in Attachment S.

The Fire PRA model represents the as-built, as-operated and maintained plant as it will be configured at the completion of the transition to NFPA 805. The Fire PRA model includes credit for the implementation of the modifications identified in Attachment S. Following completion of the modifications and implementation items listed in Attachment S, additional refinements may need to be incorporated into the Fire PRA model. However, these changes are not expected to be significant and will likely result in additional risk improvement in areas where limited credit for the proposed modifications were taken. 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 Type of Feature or System		Required for? S	L	E	R	D	Required Fire Protection Feature and System Notes
CB-A	**Control Building Basement, Control Building 903 Corridor and RPS Room 1A**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
CB-A	7A	RHR Service Water Booster Pump and Service Air Compressor Areas	Detection	Ionization	N	N	Y	Y	N	
CB-A	7A	RHR Service Water Booster Pump and Service Air Compressor Areas	Feature	Fire Barrier	N	N	N	Y	N	1-hour concrete enclosure
CB-A	7B	Emergency Condensate Storage Tank Area	Detection	Ionization	N	N	N	N	N	
CB-A	8C	RPS Room 1A	Detection	Ionization	N	N	Y	Y	N	
CB-A	8D	Seal Water Pump Area and Corridor	Suppression	Automatic Wet-Pipe	N	N	N	Y	N	
CB-A	8D	Seal Water Pump Area and Corridor	Detection	Ionization	N	N	Y	Y	N	
CB-A-1	**Control Building 903, DC Switchgear Room 1A and Battery Room 1A**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
CB-A-1	8E	Battery Room 1A	Detection	Ionization	N	N	Y	Y	N	
CB-A-1	8H	DC Switchgear Room 1A	Detection	Ionization	N	N	Y	Y	N	
CB-B	**Control Building 903, DC Switchgear Room 1B and Battery Room 1B**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
CB-B	8F	Battery Room 1B	Detection	Ionization	N	N	Y	Y	N	
CB-B	8G	DC Switchgear Room 1B	Detection	Ionization	N	N	Y	Y	N	
CB-C	**RPS Room 1B**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
CB-C	8B	RPS Room 1B	Detection	Ionization	N	N	Y	Y	N	
CB-D	**Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
CB-D	10A	Computer Room	Suppression	Auto Total Flooding Halon 1301	N	N	Y	N	N	
CB-D	10A	Computer Room	Detection	Ionization	N	N	Y	N	N	Actuates Suppression
CB-D	10B	Control Room and SAS Corridor	Detection	Heat	N	N	N	Y	N	
CB-D	10B	Control Room and SAS Corridor	Detection	Ionization	N	N	N	Y	N	
CB-D	8A	Auxiliary Relay Room	Detection	Ionization	N	N	Y	Y	N	

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 Type of Feature or System		Required for? S	L	E	R	D	Required Fire Protection Feature and System Notes
CB-D	8A	Auxiliary Relay Room	Detection	Incipient (in Panels 9-32 and 9-33)	N	N	N	Y	N	
CB-D	9A	Cable Spreading Room	Detection	Ionization	N	N	Y	Y	N	Actuates Suppression
CB-D	9A	Cable Spreading Room	Detection	Heat Actuated Devices	N	N	Y	Y	N	Actuates Suppression
CB-D	9A	Cable Spreading Room	Suppression	Preaction Sprinkler System	N	N	Y	Y	N	
CB-D	9A	Cable Spreading Room	Feature	Flame Impingement Shield	N	N	N	Y	N	Protects conduit and cable trays above PMIS-MUX-LNK6
CB-D	9A	Cable Spreading Room	Feature	Flame Impingement Shield	N	N	N	Y	N	Promat-H board protects conduit from transient fires
CB-D	9B	Cable Expansion Room	Detection	Ionization	N	N	Y	Y	N	
CB-D	9B	Cable Expansion Room	Suppression	Automatic Wet-Pipe	N	N	Y	Y	N	
CB-D	9B	Cable Expansion Room	Feature	Flame Impingement Shield	N	N	N	Y	N	Promat-H board protects conduit from transient fires
DG-A	**Diesel Generator Room 1A**		**4.2.3.2 - Deterministic Approach**							
DG-A	14A	Diesel Generator Room 1A	Detection	Heat	N	N	Y	Y	N	
DG-A	14A	Diesel Generator Room 1A	Detection	Ionization	N	N	Y	Y*	N	Actuates Suppression
DG-A	14A	Diesel Generator Room 1A	Suppression	Automatic Total Flooding Carbon Dioxide	N	N	Y	Y*	N	*Risk Significance field credits a manual capability of the suppression system only.
DG-A	14C	DG1 Day Tank Room	Suppression	Automatic Total Flooding Carbon Dioxide	N	N	N	N	N	
DG-A	14C	DG1 Day Tank Room	Detection	Heat	N	N	N	N	N	Actuates Suppression
DG-B	**Diesel Generator Room 1B**		**4.2.3.2 - Deterministic Approach**							
DG-B	14B	Diesel Generator Room 1B	Suppression	Automatic Total Flooding Carbon Dioxide	N	N	Y	Y*	N	*Risk Significance field credits a manual capability of the suppression system only.
DG-B	14B	Diesel Generator Room 1B	Detection	Ionization	N	N	Y	Y*	N	Actuates Suppression
DG-B	14B	Diesel Generator Room 1B	Detection	Heat	N	N	Y	Y	N	

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 Type of Feature or System		Required for? S	L	E	R	D	Required Fire Protection Feature and System Notes
DG-B	14D	DG2 Day Tank Room	Suppression	Automatic Total Flooding Carbon Dioxide	N	N	N	N	N	
DG-B	14D	DG2 Day Tank Room	Detection	Heat	N	N	N	N	N	Actuates Suppression
IS-A		**Intake Structure**	**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
IS-A	20A	Service Water Pump Area	Suppression	Total Flooding Halon 1301	N	N	N	Y	N	
IS-A	20A	Service Water Pump Area	Detection	Thermal Heat	N	N	N	Y	N	Actuates Suppression
IS-A	20A	Service Water Pump Area	Detection	Ionization	N	N	N	Y	N	Actuates Suppression
IS-A	20A	Service Water Pump Area	Detection	Flame	N	N	N	N	N	
IS-A	20B	Circulating Water Pump and Traveling Screen Area	None	N/A	-	-	-	-	-	
RB-A		**RCIC and Core Spray Pump Room**	**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-A	1A	RCIC and Core Spray Pump Room	Detection	Heat	N	N	N	Y	N	
RB-B		**Reactor Building South East Quad**	**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-B	1B	Core Spray Pump Room	Detection	Heat	N	N	Y	Y	N	
RB-B	1G	Control Rod Drive Pump Area	Detection	Heat	N	N	Y	Y	N	
RB-CF		**Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A**	**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-CF	1C	RHR Pump Room 1A and 1C	Detection	Heat	N	N	Y	Y	N	
RB-CF	2A-2	CRD Units - North	Detection	Ionization	N	N	N	Y	N	
RB-CF	2A-3	903' South Corridor	None	N/A	-	-	-	-	-	
RB-CF	2B	RHR Heat Exchanger Room A	None	N/A	-	-	-	-	-	
RB-DI		**Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B**	**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							

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 Type of Feature or System		Required for? S	L	E	R	D	Required Fire Protection Feature and System Notes
RB-DI	1D	RHR Pump Room 1B and 1D	Detection	Heat	N	N	N	Y	N	
RB-DI	1E	HPCI Pump Room	Detection	Heat	N	N	N	Y	N	
RB-DI	2A-3	903' South Corridor	None	N/A	-	-	-	-	-	
RB-DI	2C	CRD Units - South	Detection	Heat	N	N	N	N	N	Actuates Suppression
RB-DI	2C	CRD Units - South	Suppression	Preaction Sprinkler System	N	N	N	N	N	
RB-DI	2C	CRD Units - South	Detection	Ionization	N	N	N	Y	N	
RB-DI	2D	RHR Heat Exchanger Room B	None	N/A	-	-	-	-	-	
RB-E	**Suppression Pool Area**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-E	1F	Suppression Pool Area	None	N/A	-	-	-	-	-	
RB-FN	**Reactor Building 903' Northeast Corner**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-FN	2A-1	Reactor Building 903' Northeast Corner	Suppression	Automatic Wet-Pipe	N	N	Y	Y	N	
RB-FN	2A-1	Reactor Building 903' Northeast Corner	Detection	Ionization	N	N	Y	Y	N	
RB-FN	2A-1	Reactor Building 903' Northeast Corner	Detection	Heat	N	N	Y	Y	N	
RB-J	**Critical Switchgear Room 1F**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-J	3A	Critical Switchgear Room 1F	Detection	Ionization	N	N	Y	Y	N	
RB-K	**Critical Switchgear Room 1G**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-K	3B	Critical Switchgear Room 1G	Detection	Ionization	N	N	Y	Y	N	
RB-M	**Reactor Building North / East Side, RHR Heat Exchanger Room A**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-M	2B	RHR Heat Exchanger Room A	None	N/A	-	-	-	-	-	
RB-M	3C	REC Heat Exchanger and Pump Area	Feature	Radiant Energy Shield	N	N	N	Y	N	Protects conduit and cable trays from transient fires
RB-M	3C	REC Heat Exchanger and Pump Area	Detection	Ionization	N	N	N	Y	N	
RB-M	3D	Reactor Recirculation Motor Generator Set Lube Oil Cooler Area	Detection	Ionization	N	N	N	Y	N	

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 Type of Feature or System		Required for? S	L	E	R	D	Required Fire Protection Feature and System Notes
RB-M	3D	Reactor Recirculation Motor Generator Set Lube Oil Cooler Area	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
RB-M	3E-2	RWCU Pump Area and Corridor	None	N/A	-	-	-	-	-	
RB-N	**Reactor Building South West Corner, RHR Heat Exchanger Room B and RWCU Heat Exchanger Room**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-N	2D	RHR Heat Exchanger Room B	None	N/A	-	-	-	-	-	
RB-N	3E-1	RWCU Regenerative Heat Exchanger Areas	Detection	Ionization	N	N	N	N	Y	
RB-N	3E-2	RWCU Pump Area and Corridor	None	N/A	-	-	-	-	-	
RB-P	**Reactor Building 958 Accessible Areas**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-P	4A	Reactor Building Elevator and Accessway Area	Detection	Ionization	N	N	Y	Y	N	
RB-P	4B	Reactor Building HVAC Area	None	N/A	-	-	-	-	-	
RB-P	4C	Fuel Pool Heat Exchanger, CRD Repair Room, and Raw Water Cleanup Areas	Detection	Ionization	N	N	N	Y	N	
RB-P	4D	Reactor Recirculation Motor Generator Set Oil Pump Area	Suppression	Automatic Wet-Pipe	N	N	N	Y	N	
RB-P	4D	Reactor Recirculation Motor Generator Set Oil Pump Area	Detection	Ionization	N	N	N	Y	N	
RB-T	**Reactor Building East Side and Refueling Floor**		**4.2.3.2 - Deterministic Approach**							
RB-T	5A	SLC Pump Tank and Accessway	Detection	Ionization	N	N	Y	N	N	
RB-T	6	Refueling Floor	Detection	Heat	N	N	Y	N	N	
RB-V	**Reactor Recirculation Motor Generator Set Area**		**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
RB-V	5B	Reactor Recirculation Motor Generator Set Area	Suppression	Preaction Sprinkler System	N	N	Y	Y	N	
RB-V	5B	Reactor Recirculation Motor Generator Set Area	Suppression (m	Deluge Water Spray	N	N	N	N	N	
RB-V	5B	Reactor Recirculation Motor Generator Set Area	Detection	Flame	N	N	N	N	N	

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 Type of Feature or System		Required for? S	L	E	R	D	Required Fire Protection Feature and System Notes
RB-V	5B	Reactor Recirculation Motor Generator Set Area	Detection	Heat	N	N	N	N	N	
RB-V	5B	Reactor Recirculation Motor Generator Set Area	Detection	Ionization	N	N	Y	Y	N	
RB-V	5B	Reactor Recirculation Motor Generator Set Area	Detection	Heat Actuated Devices	N	N	Y	Y	N	Actuates Suppression
TB-A		**Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility**	**4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions**							
TB-A	11A	Turbine Lube Oil Storage Tank Room	Detection	Heat Actuated Devices	N	N	N	Y	N	Actuates Suppression
TB-A	11A	Turbine Lube Oil Storage Tank Room	Detection	Ionization	N	N	N	N	N	
TB-A	11A	Turbine Lube Oil Storage Tank Room	Suppression	Automatic Water Spray	N	N	N	Y	N	
TB-A	11B	Turbine Building Basement - South	Suppression	Automatic Wet-Pipe	N	N	N	Y	N	
TB-A	11B	Turbine Building Basement - South	Detection	Ionization	N	N	N	Y	N	
TB-A	11B	Turbine Building Basement - South	Detection	Heat Actuated Devices	N	N	N	Y	N	Actuates Suppression
TB-A	11C	H2 Seal Oil Unit Area	Suppression	Automatic Water Spray	N	N	N	Y	N	
TB-A	11C	H2 Seal Oil Unit Area	Detection	Heat Actuated Devices	N	N	N	Y	N	Actuates Suppression
TB-A	11C	H2 Seal Oil Unit Area	Detection	Ionization	N	N	N	N	N	
TB-A	11D	Condenser Pit Area	Detection	Heat	N	N	N	N	N	
TB-A	11D	Condenser Pit Area	Suppression	Automatic Wet-Pipe	N	N	N	Y	N	
TB-A	11E	Reactor Feed Pumps Area	Detection	Heat	N	N	N	N	N	
TB-A	11E	Reactor Feed Pumps Area	Detection	Heat Activated Devices	N	N	N	Y	N	Actuates Suppression
TB-A	11E	Reactor Feed Pumps Area	Suppression	Automatic Water Spray	N	N	N	Y	N	
TB-A	11F	Turbine Building Controlled Corridor 882' Elevation	Suppression	Automatic Wet-Pipe	N	N	N	Y	N	
TB-A	11F	Turbine Building Controlled Corridor 882' Elevation	Detection	Ionization	N	N	N	Y	N	
TB-A	11G	Steam Jet Air Ejector Room	Detection	Heat	N	N	N	N	N	
TB-A	11G	Steam Jet Air Ejector Room	Suppression	Automatic Wet-Pipe	N	N	N	N	N	

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 Type of Feature or System		Required for? S	L	E	R	D	Required Fire Protection Feature and System Notes
TB-A	11H	Mechanical Vacuum Pumps Room	Detection	Heat	N	N	N	N	N	
TB-A	11H	Mechanical Vacuum Pumps Room	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
TB-A	11J	Condensate, Condensate Booster and TEC Pumps Area	Suppression	Automatic Wet-Pipe	N	N	N	Y	N	
TB-A	11J	Condensate, Condensate Booster and TEC Pumps Area	Detection	Ionization	N	N	N	N	N	
TB-A	11K	Turbine Oil Conditioner Room	Detection	Heat Actuated Devices	N	N	N	Y	N	Actuates Suppression
TB-A	11K	Turbine Oil Conditioner Room	Detection	Ionization	N	N	N	N	N	
TB-A	11K	Turbine Oil Conditioner Room	Suppression	Automatic Water Spray	N	N	N	Y	N	
TB-A	11L	Pipe Chase	None	N/A	-	-	-	-	-	
TB-A	12A	ISO Phase Bus Duct Area	Detection	Heat	N	N	N	N	N	
TB-A	12A	ISO Phase Bus Duct Area	Detection	Heat Activated Devices	N	N	N	N	N	Actuates Suppression
TB-A	12A	ISO Phase Bus Duct Area	Suppression	Automatic Water Spray	N	N	N	N	N	
TB-A	12B	Turbine Building Controlled Corridor 903' Elevation	None	N/A	-	-	-	-	-	
TB-A	12C	Condenser and Heater Bay Areas	Detection	Heat	N	N	N	N	N	
TB-A	12C	Condenser and Heater Bay Areas	Suppression	Automatic Water Spray	N	N	N	Y	N	
TB-A	12C	Condenser and Heater Bay Areas	Detection	Heat Activated Devices	N	N	N	Y	N	Actuates Suppression
TB-A	12D	Turbine Building Floor - North	Detection	Heat Actuated Devices	N	N	Y	Y	N	Actuates Suppression
TB-A	12D	Turbine Building Floor - North	Suppression	Automatic Wet-Pipe	N	N	N	Y	N	
TB-A	12D	Turbine Building Floor - North	Detection	Ionization	N	N	Y	Y	N	
TB-A	12E	Turbine Oil Reservoir Area	Detection	Ionization	N	N	N	N	N	
TB-A	12E	Turbine Oil Reservoir Area	Suppression	Automatic Water Spray	N	N	N	Y	N	
TB-A	12E	Turbine Oil Reservoir Area	Detection	Heat Actuated Devices	N	N	N	Y	N	Actuates Suppression
TB-A	12F	Turbine Building Document Storage Vault	Suppression	Automatic Wet-Pipe	N	N	Y	N	N	
TB-A	12F	Turbine Building Document Storage Vault	Detection	Ionization	N	N	Y	N	N	
TB-A	13A	Turbine Operating Floor	Detection	Flame	N	N	N	Y	N	
TB-A	13A	Turbine Operating Floor	Suppression	Automatic Water Spray	N	N	N	Y	N	
TB-A	13A	Turbine Operating Floor	Detection	Heat	N	N	N	Y	N	Actuates Suppression

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 Type of Feature or System		Required for? S	L	E	R	D	Required Fire Protection Feature and System Notes
TB-A	13A	Turbine Operating Floor	Suppression	Automatic Carbon Dioxide	N	N	N	Y	N	
TB-A	13B	Non-Critical Switchgear Room	Detection	Ionization	N	N	N	Y	N	
TB-A	13C	Electrical Shop	Detection	Ionization	N	N	N	Y	N	
TB-A	13D	Instrument Shop, Instrument Records and Chart Rooms	Detection	Ionization	N	N	Y	N	N	
TB-A	15	Heating Boiler Room	Suppression	Automatic Wet-Pipe	N	N	N	Y	N	
TB-A	15	Heating Boiler Room	Detection	Heat	N	N	N	N	N	
TB-A	16	Turbine Building Exhaust Fan Room	Detection	Ionization	N	N	N	N	N	
TB-A	17	Water Treatment Building	Detection	Ionization	N	N	N	N	N	
TB-A	18A	Machine Shop	Detection	Ionization	N	N	N	N	N	
TB-A	18A	Machine Shop	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
TB-A	18A	Machine Shop	Detection	Heat	N	N	N	N	N	
TB-A	18B	Machine Shop Clean Tool Room	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
TB-A	18C	Machine Shop Oil Storage Room	Detection	Ionization	N	N	N	N	N	
TB-A	18C	Machine Shop Oil Storage Room	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
TB-A	18D	Machine Shop Paint Storage Room	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
TB-A	18D	Machine Shop Paint Storage Room	Detection	Ionization	N	N	N	N	N	
TB-A	18E	Machine Shop Lunch Room and Records Storage Room	Detection	Ionization	N	N	N	N	N	
TB-A	18E	Machine Shop Lunch Room and Records Storage Room	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
TB-A	19A	Office Building Controlled Corridor 903' Elevation	Suppression	Automatic Wet-Pipe	N	N	Y	N	N	
TB-A	19B	Office Building Occupancies and Controlled Corridors	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
TB-A	19B	Office Building Occupancies and Controlled Corridors	Detection	Ionization	N	N	Y	N	N	
TB-A	19C	Office Building Penthouse	Detection	Heat	N	N	N	N	N	
TB-A	19C	Office Building Penthouse	Suppression (m	Water Spray	N	N	N	N	N	
TB-A	19C	Office Building Penthouse	Detection	Ionization	N	N	N	N	N	

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 Type of Feature or System		Required for? S	L	E	R	D	Required Fire Protection Feature and System Notes
TB-A	19C	Office Building Penthouse	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
TB-A	21A	Radwaste Building Basement	Detection	Ionization	N	N	N	N	N	
TB-A	21B	Radwaste Building First Floor	Detection	Heat	N	N	Y	N	N	
TB-A	21B	Radwaste Building First Floor	Detection	Ionization	N	N	Y	N	N	
TB-A	21C	Radwaste Building Second Floor	Suppression	Bottled Halon 1301	N	N	N	N	N	
TB-A	21C	Radwaste Building Second Floor	Detection	Heat	N	N	N	N	N	
TB-A	21C	Radwaste Building Second Floor	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
TB-A	21C	Radwaste Building Second Floor	Detection	Ionization	N	N	N	N	N	
TB-A	21D	Radwaste Building Third Floor	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
TB-A	21D	Radwaste Building Third Floor	Detection	Ionization	N	N	N	N	N	
TB-A	22A	Augmented Radwaste Building Basement	Detection	Heat	N	N	N	N	N	
TB-A	22A	Augmented Radwaste Building Basement	Detection	Ionization	N	N	N	N	N	
TB-A	22B	Augmented Radwaste Building First Floor	Detection	Heat	N	N	Y	N	N	
TB-A	22B	Augmented Radwaste Building First Floor	Detection	Flame	N	N	Y	N	N	
TB-A	22B	Augmented Radwaste Building First Floor	Detection	Ionization	N	N	Y	N	N	
TB-A	22C	Augmented Radwaste Building Second Floor	Detection	Ionization	N	N	N	N	N	
TB-A	22C	Augmented Radwaste Building Second Floor	Detection	Flame	N	N	N	N	N	
TB-A	22C	Augmented Radwaste Building Second Floor	Suppression	Automatic Water Spray	N	N	N	N	N	
TB-A	22C	Augmented Radwaste Building Second Floor	Detection	Heat	N	N	N	N	N	
TB-A	24	Multi-Purpose Facility	Suppression	Preaction Sprinkler System	N	N	Y	N	N	
TB-A	24	Multi-Purpose Facility	Detection	Heat	N	N	Y	N	N	Actuates Suppression
YD	**Yard, Fire Pumphouse, Off-Gas Building, and Optimum Water Chemistry Building**		**4.2.3.2 - Deterministic Approach**							
YD	23A	Electric Motor Driven Fire Pump Room	Detection	Ionization	N	N	N	N	N	
YD	23A	Electric Motor Driven Fire Pump Room	Detection	Heat	N	N	N	N	N	
YD	23B	Diesel Driven Fire Pump Room	Detection	Ionization	N	N	N	N	N	

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 Type of Feature or System		Required for? S	L	E	R	D	Required Fire Protection Feature and System Notes
YD	23B	Diesel Driven Fire Pump Room	Detection	Heat	N	N	N	N	N	
YD	23B	Diesel Driven Fire Pump Room	Detection	Flame	N	N	N	N	N	
YD	23B	Diesel Driven Fire Pump Room	Suppression	Automatic Wet-Pipe	Y	N	N	N	N	Required for Chap. 3 for suppression over the diesel fire pump 3.9.4
YD	23C	Diesel Oil Tank Room	Suppression	Automatic Wet-Pipe	N	N	N	N	N	
YD	23C	Diesel Oil Tank Room	Detection	Heat	N	N	N	N	N	
YD	25	Off-Gas Building	None	N/A	-	-	-	-	-	
YD	26	Optimum Water Chemistry Building	None	N/A	-	-	-	-	-	
YD	Yard	Transformer Yard	Detection	Heat Actuated Devices	N	N	Y	Y	N	Actuates Suppression
YD	Yard	Transformer Yard	Suppression	Deluge Water Spray	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

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 CNS does not eliminate the need to comply with 10 CFR 50.48(a) and 10 CFR 50, Appendix A, General Design Criterion (GDC) 3, Fire Protection. The NRC addressed the overall adequacy of the regulations during the promulgation of 10 CFR 50.48(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.*
>
> (Reference FR Notice 69 FR 33536 dated June 16, 2004, ML041340086)

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

A comparison of the current requirements in Appendix R with the comparable requirements in Section 3 of NFPA 805 shows that the two sets of requirements are consistent in many respects. This was further clarified in FAQ 07-0032, "Clarification of 10 CFR 50.48(c), 10 CFR 50.48(a) and GDC 3 clarification" (Ref. 66). The following tables provide a cross reference of fire protection regulations associated with the post-transition CNS Fire Protection Program and applicable industry and CNS 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 LAR Attachment A 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 LAR Attachment A 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 LAR Attachment A 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 LAR Attachments A and C 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 LAR Attachment A 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 LAR Attachments A and C 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 LAR Attachment C 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. The CNS FPP is maintained as a station procedure. The CNS QAPD and Procedure 1.9. "Control and Retention of Records," provide station direction for the lifetime retention of procedures and procedure changes.

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. CNS 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 LAR Attachments A and C 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 LAR Attachment A 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 LAR Attachments A and C 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 LAR Attachment C 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 CNS LAR.

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

10 CFR 50.48(c) Section(s)	Applicability/Compliance Reference
(ii) *Plant Damage/Business Interruption Goal, Objectives, and Criteria.* The Plant Damage/Business Interruption Goal, Objectives, and Criteria of Chapter 1 are not endorsed.	The Plant Damage/Business Interruption Goal, Objectives, and Criteria of Chapter 1 of NFPA 805 are not part of the CNS 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. CNS is a 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 Attachment A.
(vi) Water supply and distribution. The italicized exception to Section 3.6.4 is not endorsed. Licensees who wish to use the exception to Section 3.6.4 must submit a request for a license amendment in accordance with paragraph (c)(2)(vii) of this section.	See Attachment A.
(vii) Performance-based methods. Notwithstanding the prohibition in Section 3.1 against the use of performance-based methods, the fire protection program elements and minimum design requirements of Chapter 3 may be subject to the performance-based methods permitted elsewhere in the standard. Licensees who wish to use performance-based methods for these fire protection program elements and minimum design requirements shall submit a request in the form of an application for license amendment under § 50.90. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the performance-based approach; (A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (B) Maintains safety margins; and (C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).	The use of performance-based methods for NFPA 805 Chapter 3 is requested by NPPD. See Attachment L.

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

	10 CFR 50.48(c) Section(s)	Applicability/Compliance Reference
(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 CNS LAR was submitted in accordance with 10 CFR 50.90. The CNS 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 CNS 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 by NPPD.

5.2 Regulatory Topics

5.2.1 License Condition Changes

The current CNS Fire Protection Operating License Condition 2.C(4) 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

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

5.2.3 Orders and Exemptions

A review was conducted of the NPPD 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. NPPD 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 proposed amendment 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 the proposed amendment 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 Updated Safety Analysis Report

After the approval of the LAR, in accordance with 10 CFR 50.71(e), the CNS USAR will be revised. The CNS Updated Safety Analysis Report is not formatted per RG 1.70, but will incorporate the applicable subject matter described in FAQ 12-0062, at a level of detail consistent with NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports." See Implementation Item S-26 of Attachment S, Table S-3.

5.5 Transition Implementation Schedule

NPPD proposes a six-month implementation period for the transition of CNS to the new fire protection licensing basis. Pursuant to this, the following activities are planned:

- NPPD commits to the specific actions identified in Table S-3 of Attachment S to Enclosure 1, within six months after issuance of the NFPA 805 License Amendment.

- NPPD will complete implementation of the required modifications identified in Table S-2 of Attachment S to the Transition Report prior to startup from the first refueling outage greater than 12 months following the issuance of the NFPA 805 License Amendment. Appropriate compensatory measures will be maintained until modifications are complete.

6.0 REFERENCES

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

1. NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition.
2. NEI 04-02, Revision 2, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," April 2008.
3. Regulatory Guide 1.205, Revision 1, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," December 2009.
4. Letter from Randall K. Edington (NPPD) to U.S. NRC, dated December 22, 2005, "Letter of Intent to Adopt NFPA 805 - Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (NLS2005109), (Agency-wide Document Access and Management System (ADAMS) Accession Number ML053640280)
5. Letter from Catherine Haney (U.S. NRC) to Randall K. Edington (NPPD), dated March 7, 2006, "NRC Response to Nebraska Public Power District's Letter of Intent to Adopt 10 CFR 50.48(c) (NFPA 805 Rule) for Cooper Nuclear Station (CNS)." (ADAMS Accession Number ML060250510)
6. Letter from Catherine Haney (U.S. NRC) to Randall K. Edington (NPPD), dated October 30, 2006, "Period of Enforcement Discretion During Implementation of National Fire Protection Association Standard 805, Cooper Nuclear Station." (ADAMS Accession Number ML063050556)
7. Letter from Stewart B. Minahan (NPPD) to U.S. NRC, dated September 19, 2008, "Request for Extension of Enforcement Discretion and Revised Submittal Date for 10 CFR 50.48(c) License Amendment Request" (NLS2008079). (ADAMS Accession Number ML082700460)
8. Letter from Joseph G. Giitter (U.S. NRC) to Stewart B. Minahan (NPPD), dated December 19, 2008, "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 (TAC No. MD9823)." (ADAMS Accession Number ML083510309)
9. Letter from John Stang (U.S. NRC) to T. Preston Gillespie (Duke Energy Carolinas, LLC), dated December 29, 2010, "Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program In Accordance With 10 CFR 50.48(c) (TAC Nos. ME3844, ME3845, and ME3846)." (ADAMS Accession Number ML103630612)
10. Letter from Brian J. O'Grady (NPPD) to U.S. NRC, dated June 27, 2011, "Revised Submittal Date for 10 CFR 50.48(c) License Amendment Request and Request for Extension of Enforcement Discretion" (NLS2011057).
11. Letter from Joseph G. Giitter (U.S. NRC) to Brian J. O'Grady (NPPD), dated July 28, 2011, "Cooper Nuclear Station – 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 (TAC No. ME6680)." (ADAMS Accession Number ML112030840)

12. NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," September 2005.
13. Letter from Roger S. Boyd (U.S. NRC), dated May 3, 1976.
14. Letter from Victor Stello (U.S. NRC) to J. M. Pilant (NPPD), dated May 11, 1976, "Cooper Nuclear Station."
15. Letter from Karl R. Goller (U.S. NRC) to J. M. Pilant (NPPD), dated September 30, 1976, "Cooper Nuclear Station."
16. Letter from Jay M. Pilant (NPPD) to D. L. Ziemann (U.S. NRC), dated December 17, 1976, "Response to Appendix A to Branch Technical Position APCB 9.5-1 Guidelines for Fire Protection for Nuclear Power Plants."
17. Letter from Jay M. Pilant (NPPD) to D. L. Ziemann (U.S. NRC), dated March 31, 1977, "Fire Hazard Analysis Response to Appendix A to Branch Technical Position APCSB 9.5-1 Guidelines for Fire Protection for Nuclear Power Plants."
18. Letter from Jay M. Pilant (NPPD) to D. L. Ziemann (U.S. NRC), dated April 6,1977, "Revisions and Additional Information Fire Protection Review."
19. Letter from Jay M. Pilant (NPPD) to D. L. Ziemann (U.S. NRC), dated February 4, 1977, "Proposed Change to the Radiological Technical Specifications."
20. Letter from Jay M. Pilant (NPPD) to Don K. Davis (U.S. NRC), dated July 20, 1977, "Proposed Change to the Radiological Technical Specifications."
21. Letter from Jay M. Pilant (NPPD) to Don K. Davis (U.S. NRC), dated December 19, 1977, "Proposed Changes to the Radiological Technical Specifications."
22. Letter from Jay M. Pilant (NPPD) to George E. Leer (U.S. NRC), dated May 11, 1978, "Fire Protection Program."
23. Letter from Jay M. Pilant (NPPD) to Victor Stello (U.S. NRC), dated June 21, 1978.
24. Letter from Jay M. Pilant (NPPD) to Thomas A. Ippolito (U.S. NRC), dated August 16, 1978, "Fire Protection/Request for Additional Information."
25. Letter from Jay M. Pilant (NPPD) to Thomas A. Ippolito (U.S. NRC), dated December 11, 1978, "Fire Inspection Review Response."
26. Letter from Jay M. Pilant (NPPD) to Thomas A. Ippolito (U.S. NRC), dated April 12, 1979, "Fire Protection Program/Fire Brigade Training."
27. Letter from Karl R. Goller (U.S. NRC) to J. M. Pilant (NPPD), dated November 29, 1977, "Cooper Nuclear Station."
28. Letter from Thomas A. Ippolito (U.S. NRC) to J. M. Pilant (NPPD), dated May 23, 1979.
29. Letter from Jay M. Pilant (NPPD) to Thomas A. Ippolito (U.S. NRC), dated October 22, 1979, "Fire Protection Technical Specification Changes."
30. Letter from Jay M. Pilant (NPPD) to Thomas A. Ippolito (U.S. NRC), dated January 16, 1980, "Fire Protection Modifications."
31. Letter from Thomas A. Ippolito (U.S. NRC) to J. M. Pilant (NPPD), dated November 21, 1980.
32. Letter from Jay M. Pilant (NPPD) to Domenic B. Vassallo (U.S. NRC), dated June 28, 1982, "Fire Protection Rule 10CFR50, Appendix R" (LQA8200158).

33. Letter from Jay M. Pilant (NPPD) to Domenic B. Vassallo (U.S. NRC), dated December 2, 1983, "Response to 10CFR50, Appendix R, 'Fire Protection of Safe Shutdown Capability – Volume III'" (LQA8300256).

34. Letter from Domenic B. Vassallo (U.S. NRC) to J. M. Pilant (NPPD), dated April 16, 1984, "Safety Evaluation for Appendix R to 10 CFR 50, Items II.G.3 and III.L, Alternate or Dedicated Shutdown Capability."

35. Letter from Michael T. Coyle (NPPD) to U.S. NRC, dated July 29, 2002, "Safety Evaluation For Appendix R to 10 CFR Part 50, Items II.G.3 and III.L, Alternate or Dedicated Shutdown Capability Clarification" (NLS2002085).

36. Letter from Jay M. Pilant (NPPD) to Domenic B. Vassallo (U.S. NRC), dated May 9, 1985, "Appendix R Analysis of Cooper Nuclear Station" (NLS8500085).

37. Letter from Jay M. Pilant (NPPD) to Domenic B. Vassallo (U.S. NRC), dated June 7, 1985, "Appendix R – Schedular Exemptions; Request for."

38. Letter from Hugh L. Thompson (U.S. NRC) to J. M. Pilant (NPPD), dated August 21, 1985, "Outstanding Fire Protection Modifications."

39. Letter from Domenic B. Vassallo (U.S. NRC) to J. M. Pilant (NPPD), dated April 29, 1983, "Cooper Nuclear Station."

40. Letter from Byron L. Siegel (U.S. NRC) to J. M. Pilant (NPPD), dated June 1, 1984.

41. Letter from Ernest D. Sylvester (U.S. NRC) to J. M. Pilant (NPPD), dated January 3, 1985.

42. Letter from William O. Long (U.S. NRC) to J.M. Pilant (NPPD), dated April 10, 1986.

43. Letter from William O. Long (U.S. NRC) to J. M. Pilant (NPPD), dated September 9, 1986.

44. Letter from Paul W. O'Connor (U.S. NRC) to George A. Trevors (NPPD), dated November 7, 1988, "Cooper Nuclear Station – Amendment No. 126 to Facility Operating License No. DPR-46 (TAC No. 67835)."

45. Letter from Paul W. O'Connor (U.S. NRC) to George A. Trevors (NPPD), dated February 3, 1989, "Cooper Nuclear Station – Amendment No. 127 to Facility Operating License No. DPR-46 (TAC No. 65623)."

46. Letter from Jack Donohew (U.S. NRC) to G. R. Horn (NPPD), dated July 31,1998, "Conversion to Improved Technical Specifications for the Cooper Nuclear Station – Amendment No. 178 to Facility Operating License No. DPR-46 (TAC No. M98317)."

47. Letter from G. R. Horn (NPPD) to U.S. Nuclear Regulatory Commission, dated December 16, 1994, "Fire Protection Program Commitment Revision" (NLS940085).

48. Letter from James R. Hall (U.S. NRC) to Guy R. Horn (NPPD), dated August 15, 1995, "Revocation of Exemption From 10 CFR Part 50, Appendix R – Cooper Nuclear Station (TAC No. M91269)."

49. Letter from Domenic B. Vassallo (U.S. NRC) to L. G. Kuncl (NPPD), dated September 21, 1983, "Exemption Requests – 10 CFR 50.48 Fire Protection and Appendix R to 10 CFR Part 50."

50. Letter from Jay M. Pilant (NPPD) to Domenic B. Vassallo (U.S. NRC), dated March 18, 1983, "Fire Protection Rule 10 CFR 50, Appendix R, Preliminary Supplemental Response (Revised)" (LQA8300109).

51. Letter from Jay M. Pilant (NPPD) to Domenic B. Vassallo (U.S. NRC), dated June 2, 1983, "Fire Protection Rule 10 CFR 50, Appendix R, Preliminary Supplemental Response (Revision 2)."
52. Letter from Sunil Weerakkody (U.S. NRC) to Alexander Marion (NEI), dated July 12, 2006, "Process for Frequently Asked Questions For Title 10 of the Code of Federal Regulations, Part 50.48(c) Transitions." (ADAMS Accession Number ML061660105)
53. NRC Regulatory Issue Summary 2007-19, "Process For Communicating Clarifications of Staff Positions Provided In Regulatory Guide 1.205 Concerning Issues Identified During the Pilot Application of National Fire Protection Association Standard 805," dated August 20, 2007. (ADAMS Accession Number ML071590227)
54. NEI 00-01, Revision 1, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," January 2005.
55. FAQ 07-0030. Revision 5, dated November 4, 2010, "Establishing Recovery Actions." (ADAMS Accession Number ML103090602)
56. FAQ 07-0038, Revision 3, dated November 4, 2010, "Lessons Learned on Multiple Spurious Operations." (ADAMS Accession Number ML103090608)
57. FAQ 08-0054, Revision 1, dated December 17, 2010, "Demonstrating Compliance with Chapter 4 of NFPA 805." (ADAMS Accession Number ML103510379)
58. FAQ 07-0040, Revision 4, dated July 24, 2008, "Non-Power Operations Clarifications." (ADAMS Accession Number ML082070249)
59. FAQ 09-0056, Revision 2, dated September 23, 2010, "Radioactive Release Transition." (ADAMS Accession Number ML102810600)
60. ASME/ANS RA-Sa 2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers, New York, NY, and American Nuclear Society, La Grange Park, IL, 2009.
61. Regulatory Guide 1.200, Revision 2, "An Approach For Determining the Technical Adequacy of Probabilistic Risk Assessment Results For Risk-Informed Activities," March 2009.
62. EPRI 1016735, "Fire PRA Methods Enhancements – Additions, Clarifications, and Refinements to EPRI 1019189," Interim Report, December 2008.
63. NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," Draft Report for Comment, November 2009.
64. FAQ 10-0059, Revision 5, dated February 10, 2012, "NFPA 805 Monitoring." (ADAMS Accession Number ML120410589)
65. Cooper Nuclear Station Quality Assurance Program for Operation – Policy Document, Revision 21, dated April 16, 2010.
66. FAQ 07-0032, Revision 2, dated May 9, 2008, "Clarification of 10 CFR 50.48(c), 50.48(a), and GDC 3." (ADAMS Accession Number ML081300697)

ATTACHMENT A

NEI 04-02 Table B-1 – Transition of Fundamental Fire Protection Program and Design Elements

80 Pages

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
3.1 General	This chapter contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. These fire protection program elements and minimum design requirements shall not be subject to the performance-based methods permitted elsewhere in this standard. Previously approved alternatives from the fundamental protection program attributes of this chapter by the AHJ take precedence over the requirements contained herein.	N/A	CNS compliance with the Fire Protection Program elements and minimum design requirements presented in NFPA 805 Chapter 3 is not based on the performance-based methods permitted elsewhere in the standard. The following discussion of Chapter 3 requirements documents CNS compliance or intended compliance with the individual code sections. In addition, documentation is provided where previous NRC approval has been granted for a deviation to the code.	None
3.2 Fire Protection Plan	N/A	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
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	Procedure 0.23 establishes a Fire Protection Program for the plant. This procedure describes the functional responsibilities and administrative controls for implementing the CNS Fire Protection Program.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
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	Procedure 0.23 establishes the general policy for the site Fire Protection Program, Attachment 6, Section 2, defines the administrative controls and departmental responsibilities.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.2.2.1 [Management Policy on Senior	The policy document shall designate the senior management	Complies	Procedure 0.23 establishes that the Vice President-Nuclear and	Procedure 0.23, Rev. 64, CNS Fire Protection Plan

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
Management]	position with immediate authority and responsibility for the fire protection program.		Chief Nuclear Officer is ultimately responsible for the FP Program including the implementation of functional elements of the program.	
3.2.2.2 [Management Policy on Daily Administration]	The policy document shall designate a position responsible for the daily administration and coordination of the fire protection program and its implementation.	Complies	Procedure 0.23, Attachment 6, Paragraph 2.9 states that the FP Manager is responsible for the daily administration and coordination of the program. Procedure 0.23, Attachment 6, Section 2.9.1, states that the FP Manager shall be, or have within his organization, a qualified FP Engineer. The FP Program Engineer is able to meet the requirements for TQD-0986 Fire Protection Program Engineer.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.2.2.3 [Management Policy on Interfaces]	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	Procedure 0.23, Attachment 6, Section 2 outlines the Fire Protection Program organization, which includes the Vice President-Nuclear and Chief Nuclear Officer, General Manager of Plant Operations, and Director of Engineering. These positions have the authority for implementing the various areas of the Fire Protection Program. Procedure 0.23 also defines the fire protection interfaces with other organizations and the responsibilities for the coordination of activities.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.2.2.4 [Management Policy on AHJ]	The policy document shall identify the appropriate AHJ for the various areas of the fire protection program.	Complies with Required Action	The Authority Having Jurisdiction (AHJ) (e.g. NRC, NEIL, etc.) is not identified in Procedure 0.23, CNS Fire Protection Plan.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Implementation Item S-3.9 – Revise Procedure 0.23 to identify the AHJ for the various areas of the Fire Protection Program. See Attachment S, Table S-3.	
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	CNS Technical Specification (TS) 5.4.1d requires that written procedures be established, implemented, and maintained for fire protection program implementation. Note - This License Amendment Request deletes TS 5.4.1d. See Attachment N.	CNS Technical Specifications, through License Amendment 241
3.2.3 Procedures (1)	Inspection, testing, and maintenance for fire protection systems and features credited by the fire protection program	Complies with Required Action	Inspection, testing, and maintenance requirements for credited fire protection systems and features are described in the Technical Requirements Manual (TRM), a document incorporated by reference into the CNS Updated Safety Analysis Report. This includes surveillances for the fire main, automatic suppression systems, high pressure carbon dioxide extinguishing system, Halon suppression system, fire detection systems, standpipe and hose systems, fire pumps, fire barriers, and penetration seals. The fire protection systems and features are inspected, tested, and maintained in accordance with the 6, 7, and 15 series of CNS Procedures. Implementation Item S-3.1 –	CNS Technical Requirements Manual, Rev. 10/28/11

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			During the implementation of the NFPA 805 licensing basis, performance-based surveillance frequencies will be established as described in Electric Power Research Institute (EPRI) Technical Report (TR) 1006756, "Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features". The performance-based surveillance frequencies will be evaluated in the monitoring program in accordance with NFPA 805 FAQ 10-0059. See Attachment S, Table S-3.	
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	The TRM establishes the limits on impairment duration of the credited fire protection features. CNS Procedure 0.23 implements these TRM limits on impairment duration and establishes the required compensatory actions. Procedure 0.23, Section 6.1, requires that impairments to fire protection features be minimized to maintain the greatest level of readiness possible. When features are impaired, compensatory measures are commonly required to provide an acceptable level of protection. CNS license requirements dictate specific compensatory measures for specific impairments. The Impairments and Compensatory Measures Matrix, Attachment 1,	CNS Technical Requirements Manual, Rev. 10/28/11 Procedure 0.23, Rev. 64, CNS Fire Protection Plan Procedure 0.39.1, Rev. 7, Fire Watches and Fire Impairments

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			describes conditions when fire impairments may be needed and the associated compensatory action. Section 6.2 requires that when a fire protection discrepancy is noted that causes an impairment, corrective actions are initiated to restore the impaired fire protection feature as soon as possible. Procedure 0.39.1 provides criteria for issuing Fire Protection Impairment Permits for reporting, tracking, providing compensatory measures, and ensuring restoration of impaired features.	
3.2.3 Procedures (3)	Reviews of fire protection program — related performance and trends	Complies	Per Procedure 0.23, the Fire Protection System Engineer is responsible for equipment trends associated with the FP Systems. Procedure 0-CNS-64 identifies the objective of the System Health Teams to produce timely and effective solutions to current and future reliability performance problems. Procedure EN-DC-329 provides direction for the use of Program Notebooks, Program Health Reports, self assessments, benchmarks, and similar Management oversight tools.	EN-DC-329, Rev. 4C0, Engineering Programs Control and Oversight Procedure 0.23, Rev. 64, CNS Fire Protection Plan Procedure 0-CNS-64, Rev. 3, System Health Teams
3.2.3 Procedures (4)	Reviews of physical plant modifications and procedure changes for impact on the fire	Complies	Procedure 3.4 provides the requirements for Configuration Changes at CNS, including	EDP-06, Rev. 45, Supporting Requirements for Configuration Change Control

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	protection program		identifying applicable interdepartment/interdiscipline reviews per Procedure 3.4.8. Procedure EDP-06 provides criteria for when a fire protection review is required to support configuration changes. Procedure 0.4 provides administrative controls over the procedure change process at CNS. Procedures 0.8 and 3.3SAFE identify when a 0.29.4 fire protection review is required.	Procedure 0.4, Rev. 55, Procedure Change Process Procedure 0.8, Rev. 24, 10CFR50.59 and 10CFR72.48 Reviews Procedure 0.29.4, Rev. 17, Other Regulatory Reviews Procedure 3.3SAFE, Rev. 16, Safety Assessment Procedure 3.4, Rev. 54, Configuration Change Control Procedure 3.4.8, Rev. 19, Design Verification
3.2.3 Procedures (5)	Long-term maintenance and configuration of the fire protection program	Complies	Procedure 0.23, Section 1.1 describes the purpose of the procedure to establish functional responsibilities and administrative controls for implementing the CNS Fire Protection Program as part of its Long Term Compliance Program to 10CFR50.48 (Appendix R). This includes meeting the requirements of: Technical Specifications Administrative Controls, the TRM, the Updated Safety Analysis Report (USAR), Appendix A to Branch Technical Position 9.5-1, NFPA codes and standards, and Nuclear Electric Insurance Limited (NEIL) requirements. Procedure 0.23, Section 4.1 invokes the Configuration	Procedure 0.23, Rev. 64, CNS Fire Protection Plan

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Control process which contains design input considerations that are utilized to ensure configuration changes do not adversely affect the Fire Protection Design Basis, and ensure updates are performed to maintain the Fire Protection Long Term Compliance Program.	
3.2.3 Procedures (6)	Emergency response procedures for the plant industrial fire brigade	Complies	Procedure 5.1INCIDENT contains emergency response guidelines for the industrial fire brigade.	Procedure 5.1INCIDENT, Rev. 21, Site Emergency Incident
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	Procedure 0.23, Section 3 establishes specific requirements for a fire prevention program.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.3 Prevention (1)	Prevention of fires and fire spread by controls on operational activities	Complies	Procedure 0.23, Step 3.1 provides requirements for control of ignition sources. Procedures 0.39 and 7.7.1 establish controls on operational activities in order to prevent fires.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan Procedure 0.39, Rev. 45, Hot Work Procedure 7.7.1, Rev. 15, Special Process Control Maintenance Procedure
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	Procedure 0.23, Step 3.2 provides requirements for control of combustibles. Procedure 3.4 provides design controls on the use of combustible materials.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan Procedure 3.4, Rev. 54, Configuration Change Control
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	Complies	Procedure 0.7.1 provides requirements and controls for use and staging of estimated amounts of transient	Procedure 0.39, Rev. 45, Hot Work Procedure 0.39.1, Rev. 7, Fire Watches and Fire Impairments

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	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.		combustible materials within the power block and other plant areas. Procedure 0.39 establishes requirements for hot work procedures, including training and supervision. Procedure 0.39.1 establishes requirements for fire watches during hot work activities. Procedure 0.7.1.1 provides requirements for liquids not in use to be stored in approved lockers. Procedure 0.7 provides guidance to control storage, handling, and use of chemical materials at CNS. Procedure 7.3.61 requires that temporary power installations be controlled to ensure the installation is per approved standards, and that any temporary power installation that is in place > 90 days shall be inspected to ensure the installation has not degraded.	Procedure 0.7, Rev. 32, Chemical Material Control Procedure 0.7.1, Rev. 30, Control of Combustibles Procedure 0.7.1.1, Rev. 1, Control of Flammable Materials Lockers Procedure 7.3.61, Rev. 6, Temporary Power
3.3.1.1 General Fire Prevention Activities	The fire prevention activities shall include but not be limited to the following program elements:	Complies	Plant procedures for general fire prevention activities have been developed and implemented. The procedures address, at a minimum, the fire protection program elements identified in the sections below, but are not limited to these elements. Upon review of these procedures, NPPD concludes that the NFPA 805 code requirements in the following subsections are satisfied.	None
3.3.1.1 General Fire Prevention Activities (1)	Training on fire safety information for all employees and contractors	Complies with Clarification	Site-Specific Plant Access Training GEN001-02-04	GEN001-02-04, Rev. 29, Site-Specific Plant Access Training

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	including, as a minimum, familiarization with plant fire prevention procedures, fire reporting, and plant emergency alarms		provides training information on fire reporting, plant fire alarms, and general plant fire prevention. In accordance with the guidance in Section K.3 of NEI 04-02, this training complies with the interpreted meaning of "familiarization with plant fire prevention procedures, fire reporting, and plant emergency alarms."	NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," Rev. 2
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	Procedure 0.23, Attachment 6, Paragraph 2.12 describes that the Station Fire Marshal implements, maintains, and updates the Fire Protection administrative portion of the Fire Protection Program. The Fire Marshal performs these duties through implementation of the Hot Work and transient combustible procedures, plant fire tours, and for assuring compliance with the Fire Protection Program. Procedure 0.23, Section 8 requires that records which demonstrate conformance with Fire Protection requirements be maintained and retrievable through the Records Management System. The record types include inspections and tests, non-conformances, and deficiencies. Procedure 0.23 Step 6.6.1.4 requires that deficiencies requiring Alternative Compensatory Measures are documented within the Corrective Action Program.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.3.1.1 General Fire	Administrative controls addressing	Complies	Procedure 0.23, Attachment 6,	EDP-06, Rev. 45, Supporting

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
Prevention Activities (3)	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		Paragraph 2.10 describes the Fire Protection Program Engineer as being responsible for reviews of Change Evaluation Documents and ensures the appropriate Fire Protection Program documentation is updated as a result. Attachment 6, Paragraph 2.12 requires that the Station Fire Marshal implements, maintains, and updates the Fire Protection administrative portion of the Fire Protection Program. As part of this responsibility, the Fire Marshal reviews implementation of the Hot Work and transient combustible procedures, performs plant fire tours assuring compliance with the Fire Protection Program, and performs the pre- and post-maintenance reviews of fire protection-related Work Orders. EDP-06 provides controls for fire protection review of plant modifications. Procedure 0.40, Attachment 5, Section 4.18 describes the responsibilities of the Fire Protection Engineer to review the Fire Protection impact of maintenance activities.	Requirements for Configuration Change Control Procedure 0.23, Rev. 64, CNS Fire Protection Plan Procedure 0.40, Rev. 82, Work Control Program
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	Procedure 0.23, Step 3.4 requires that housekeeping inspections be periodically performed by Plant Management per Procedure 0-CNS-07-MGMT. Procedure 0.23, Step 3.2 provides requirements for control of	Procedure 0.23, Rev. 64, CNS Fire Protection Plan Procedure 0.7.1, Rev. 30, Control of Combustibles Procedure 0-CNS-07-MGMT, Rev. 6, Management Observation Program

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			combustibles. Combustibles are controlled by Procedure 0.7.1. Note: Plant procedures for the control of general housekeeping practices and the control of transient combustibles have been developed and implemented. The procedures address, at a minimum, the fire protection program elements identified in the sections below, but are not limited to these elements. Upon review of these procedures, NPPD concludes that the NFPA 805 code requirements in the following subsections are satisfied.	
3.3.1.2 Control of Combustible Materials (1)	Wood used within the power block shall be listed pressure-impregnated or coated with a listed fire-retardant application. *Exception: Cribbing timbers 6 in. by 6 in. (15.2 cm by 15.2 cm) or larger shall not be required to be fire-retardant treated.*	Submit for NRC Approval	Procedure 0.7.1, Section 4.3 provides guidelines for control of Class A combustibles within the Power Block. The requirement for all wood to be treated with fire retardant is intended to apply to raw materials such as plywood and lumber products and doesn't apply to commercially available products which utilize small quantities of wood as an integral part of a finished product (e.g., tools, janitorial supplies, special fixtures, M&TE, and office type furniture). Per Section 4.3.4, bulk lumber at CNS is purchased through the procurement process, which requires lumber to be fire retardant pressure treated. If pressure impregnated wood is	Procedure 0.7.1, Rev. 30, Control of Combustibles

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			not available, Fire Protection personnel approval is required for using wood treated with surface applied chemicals. Per Section 4.3.6, heavy wood members with a cross-sectional area greater than 6" x 6" are not required to be treated with a fire retardant. Refer to Attachment L for further details on the request for NRC approval of the use of commercially available products which utilize small quantities of non-treated wood as an integral part of a finished product (e.g., tools, janitorial supplies, special fixtures, M&TE, and office type furniture).	
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	Procedure 0.7.1, Section 4.3.7 requires plastic film and fabrics used as sheeting material for protective floor coatings, contamination control, temporary enclosure, etc., shall be approved self-extinguishing fire retardant plastic sheeting (NFPA 701, UL Standard 214, or equivalent standard). Attachment 5 of Procedure 0.7.1 details a list of acceptable plastics for use within the Power Block.	Procedure 0.7.1, Rev. 30, Control of Combustibles
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	Procedure 0.7.1, Section 4.2.7 requires waste, debris, scrap, oil spills, or other combustibles resulting from the work activity, should be removed promptly following completion of the work or at the end of each shift,	Procedure 0.7.1, Rev. 30, Control of Combustibles

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			whichever comes first. Per Section 4.4, when practical, equipment shipped in untreated combustible containers should be unpacked prior to plant entry. When untreated combustible packing materials and shipping containers are unpacked in the plant. A Transient Combustible Evaluation is obtained for quantities of packing materials exceeding the threshold values for the associated plant area.	
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	Procedure 0.7.1, Section 4.2.5 requires that transient combustible materials should be located in designated storage areas or arranged so as to minimize the fire hazard to cable trays and plant equipment. Attachment 3 of Procedure 0.7.1 establishes the designated storage areas for combustibles. Procedure 0.7.1.1, Attachment 2 establishes the maximum allowable size of flammable or combustible liquids in storage containers and portable tanks, as well as the maximum storage quantities for cabinets.	Procedure 0.7.1, Rev. 30, Control of Combustibles Procedure 0.7.1.1, Rev. 1, Control of Flammable Materials Lockers
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 with use of Existing Engineering Equivalency Evaluations (EEEEs) Complies with Clarification	<u>Complies with use of EEEEs</u> Plant procedures for the controls on the use and storage of flammable and combustible liquids are in compliance with the requirements of NFPA 30, as documented in the NFPA 30-1973 code review checklist in Engineering Evaluation EE 10-071.	EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," Rev. 2

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Complies with Clarification No other NFPA Standards were determined to be applicable based on guidance in Nuclear Energy Institute (NEI) 04-02, Rev. 2, Section K.1 (FAQ 06-0020).	
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 use of Existing Engineering Equivalency Evaluations (EEEEs) Complies with Clarification	Complies with use of EEEEs Plant procedures for the controls on the use and storage of hydrogen are in compliance with the requirements of NFPA 50A, as documented in the NFPA 50A-1973 code review checklist in Engineering Evaluation EE 10-071. Complies with Clarification No other NFPA Standards were determined to be applicable based on guidance in Nuclear Energy Institute (NEI) 04-02, Rev. 2, Section K.1 (FAQ 06-0020).	EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," Rev. 2
3.3.1.3 Control of Ignition Sources	N/A	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
3.3.1.3.1 [Control of Ignition Sources - Code Requirements]	A hot work safety procedure shall be developed, implemented, and periodically updated as necessary in accordance with NFPA 51B, "Standard for Fire Prevention During Welding, Cutting, and Other Hot Work", and NFPA 241, "Standard for Safeguarding Construction, Alteration, and Demolition Operations."	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	Hot work safety procedures are in compliance with the requirements of NFPA 51B and NFPA 241, as documented in the NFPA 51B–1999 and NFPA 241–2000 code review checklists in Engineering Evaluation EE 10-071.	EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review
3.3.1.3.2 [Control of Ignition Sources -	Smoking and other possible sources of ignition shall be	Complies	Site-Specific Plant Access Training GEN001-02-04	GEN001-02-04, Rev. 29, Site-Specific Plant Access Training

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
Smoking Limitations]	restricted to properly designated and supervised safe areas of the plant.		requires that smoking is prohibited in all NPPD buildings, facilities, vehicles, and aircraft. Smoking is permitted on NPPD property out-of-doors. Local management has the authority to designate areas for smoking out-of-doors. Procedure 0.23, Step 3.1 requires that ignition sources are controlled through Procedure 0.39, Hot Work.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.3.1.3.3 [Control of Ignition Sources - Leak Testing]	Open flames or combustion-generated smoke shall not be permitted for leak or air flow testing.	Complies	Procedure 0.23, Step 3.1 and Procedure 0.39, Step 3.2 state that the use of ignition sources or combustion generated smoke for leak testing is prohibited. A commercially approved aerosol technique shall be used for this purpose.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan Procedure 0.39, Rev. 45, Hot Work
3.3.1.3.4 [Control of Ignition Sources - Portable Heaters]	Plant administrative procedure shall control the use of portable electrical heaters in the plant. Portable fuel-fired heaters shall not be permitted in plant areas containing equipment important to nuclear safety or where there is a potential for radiological releases resulting from a fire.	Complies with Required Action	Administrative procedures do not establish controls on the use of portable electrical heaters in the plant. Portable fuel-fired heaters are currently allowed inside buildings provided written approval and guidance from FP/Designee is attained prior to commencing work per Procedure 0.39. Implementation Item S-3.10 – Administrative procedures will be revised to control the use of portable electric heaters, and revised to document that portable fuel-fired heaters are not permitted in plant areas containing equipment important to nuclear safety, or where there is a potential for radiological release resulting from a fire. See	Procedure 0.39, Rev. 45, Hot Work

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Attachment S, Table S-3.	
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	USAR Section XII-2.0 (Structural Design) provides the construction of the walls, floors, and components required for structural integrity. These material include concrete and steel which are noncombustible.	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2
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 Complies by Previous NRC Approval Submit for NRC Approval	<u>Complies</u> CNS complies with NFPA 805, Section 3.3.3, except for NRC previous approval of plastic laminated particle board installed within the Control Room and Computer Room. The Fire Hazards Analysis (FHA) requires all interior finishes to have a structural base of non-combustible material, with a surface not exceeding a thickness of 1/8" which has a flame spread rating of not >50 according to ASTM E-84 testing procedures. This definition meets the requirements of the NRC in NUREG-0800, CMEB 9.5-1, Rev. 2, July 1981. Class A finishes as defined in NFPA 101 are those that have a flame spread index of less than or equal to 25 and a smoke developed index of less than or equal to 450. Per Section 4.3.1 of Procedure 0.7.1, interior wall and structural components should be non-combustible or listed by a nationally recognized testing	CNS Fire Hazards Analysis, Rev. 9/26/2011 EE 12-009, Rev. 0, Acceptance of Unqualified Coatings within CNS Letter dated 10/19/79, Fire Protection Modifications, from Pilant (NPPD) to Ippolito (NRC) Letter dated 11/21/80, from Ippolito (NRC) to Pilant (NPPD) Plant Modification MDC 87-024 Procedure 0.7.1, Rev. 30, Control of Combustibles Procedure 7.0.15, Rev. 14, Station Painting Procedure

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Laboratory, such as Factory Mutual or Underwriters Laboratory, Inc., for flame spread of 25 or less and a smoke development of 50 or less. <u>Complies by Previous NRC Approval</u> CNS was originally constructed and licensed to the 1967 Draft General Design Criterion 3, and Branch Technical Position APCSB 9.5-1, Appendix A. The NRC approved this design based on the following: The NPPD response to Branch Technical Position APCSB 9.5-1, Appendix A, dated 12/17/76, and a revised submittal dated 4/6/77 identified the use of a plastic laminate particle board used for the south, east, and north walls of the Control Room (Fire Zone 10B) and Computer Room (Fire Zone 10A). Per letter dated 4/6/77, "A plastic laminate is adhered to the particle board wall panels provided on the south, east, and north wall of the Control Room. This particle board is approximately 5/8" thick and is fastened to wood furring which is fastened to the concrete fire walls. As both rooms are provided with adequate manual back-up firefighting equipment, Automatic Fire Detection Systems (Products-of-	

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Combustion detectors), the Computer Room does not hose any safety related equipment, and the Control Room is constantly attended, no further action will be taken nor additional fire protection provided." Section 3.15.2 of the CNS Fire Protection SER dated 5/23/79 stated "Certification will be obtained or tests performed to show that the particle board used in the control room has acceptable flame spread and smoke spread ratings." Supplement 1 to the CNS Fire Protection SER dated 11/21/80 documented that this item was closed based on the flame spread rating test results per ANSI/ASTM E 84-77a provided by NPPD in a letter dated 10/19/79. The plastic laminate particle board, as closed per Supplement 1 to the SER, is still installed at CNS. There have been no plant modifications or other changes that would invalidate the basis for approval. The plastic laminate particle board features remain unchanged. <u>Submit for NRC Approval</u> Attachment 3 of Procedure 7.0.15, Station Painting Procedure lists the	

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			paints/coatings that can be used. There are paints and coatings which do not have the necessary documentation to demonstrate testing under ASTM E-136 or an equivalent test method. EE 12-009 justifies the acceptability of these paints and coatings. Refer to Attachment L for further details on the request for NRC approval for the continued use of these unqualified paints and coatings.	
3.3.4 Insulation Materials	Thermal insulation materials, radiation shielding materials, ventilation duct materials, and soundproofing materials shall be noncombustible or limited combustible.	Complies with Required Action	Procedure 0.7.1, Section 4.3.1 requires that interior wall and structural components, thermal insulation materials, radiation shielding materials, and soundproofing should be non-combustible or listed by a nationally recognized testing Laboratory, such as Factory Mutual or Underwriters Laboratory, Inc., for flame spread index of 25 or less and a smoke development index of 50 or less. Procedure 7.0.13 states that insulation should be fabricated and installed per Topical Reference Information Manual DCD-40. DCD-40 states that all insulation installed should meet the minimum fire and smoke spread ratings. Implementation Item S-3.21 - Revise Procedure 0.7.1 to include the requirement for	DCD-40, Topical Reference Information Manual, Thermal Insulation, 2/2/09 Procedure 0.7.1, Rev. 30, Control of Combustibles Procedure 7.0.13, Rev. 15, Control of Insulation Removal and Installation

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			ventilation duct materials to be non-combustible or listed by a nationally recognized testing Laboratory, such as Factory Mutual or Underwriters Laboratory, Inc., for flame spread index of 25 or less and a smoke development index of 50 or less. See Attachment S, Table S-3.	
3.3.5 Electrical.	Electrical	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
3.3.5.1 [Electrical - Wiring Above Suspended Ceiling Limitations]	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.	Submit for NRC Approval	Procedure 0.7.1, Section 4.3.2 requires that concealed spaces should be devoid of combustibles. Section 4.3.2.1 states minor amounts of approved communications cabling may be used within these concealed spaces. There are no significant amounts of wiring above suspended or dropped ceilings, and most of the wiring and cabling that is installed above the suspended or dropped ceiling is in conduit and/or meet one of the acceptable cable qualifications listed within FAQ 06-0022 Rev. 3. EDP-06, Attachment 1, Section F1 includes the requirements for electrical wiring above suspended ceilings and specifically references NFPA 805, Section 3.3.5.1 as part of the fire protection design considerations.	EDP-06, Rev. 45, Supporting Requirements for Configuration Change Control Procedure 0.7.1, Rev. 30, Control of Combustibles

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Some of the wiring installed above the suspended ceilings in the Power Block does not comply with the requirements of Section 3.3.5.1. The wiring in these locations that is not approved for plenum use and not installed in conduit includes lighting/power receptacle circuits, Gai-tronics cables, fire detection circuits and/or communication cables associated with computers, telephones, televisions, or projectors. Refer to Attachment L for further details on the request for NRC approval of the minimal amount of wiring located above suspended ceilings in the Power Block that are not approved for plenum use and not installed in conduit.	
3.3.5.2 [Electrical - Raceway Construction Limits]	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 Submit for NRC Approval	<u>Complies</u> Procedure 7.3.55, Section 8.4 requires that flexible conduit runs shall be designed as short as possible. Section VIII-4 (Auxiliary Power Distribution System) of the USAR requires the 4160 volt cables be installed in conduits and the 480 volt power cables and control cables be installed in conduit and metal trays. Section VIII-6 (125-250 Volt DC Power Systems) of the USAR requires the 125 and 250 Volt DC system cables be installed in conduits and metal trays.	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2 Procedure 7.3.55, Rev. 8, Raceway Installation

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Submit for NRC Approval Refer to Attachment L for further details on the request for NRC approval of the use of plastic conduit for below grade and embedded conduit.	
3.3.5.3 [Electrical - Cable Flame Propagation Limits]	Electric cable construction shall comply with a flame propagation test as acceptable to the AHJ. Note: the exception to this section is not endorsed by 10 CFR 50.48(c)(2)(v) and has been removed.	Complies by Previous NRC Approval Complies with Required Action	Complies by Previous NRC Approval The NPPD response to Branch Technical Position APCSB 9.5-1, Appendix A dated 12/17/76, as revised on 4/6/77 states, "Refer to Appendices A, B, C & D for the Cable Design Specifications and Cable Fire Tests performed on cable at CNS. Refer to Appendix 'E' for an item by item comparison of Cable Fire Tests performed on CNS cable vs. the IEEE No. 383 Flame Test. Based on the results of the above attachments, the District feels that the fire-retardant characteristics of the cables at CNS far exceed the type of "fire-retardant" cable used in other plants of its vintage. Therefore, no additional action will be taken." Section 4.8 of the CNS Fire Protection SER dated 5/23/79 states: "Flame tests conducted on the electrical cables used at Cooper Plant were comparable to the combustibility tests set forth in IEEE 383. The results show that, in the configurations and with the ignition sources used in the tests the cable	Letter dated 12/17/76, Response to Appendix A to Branch Technical Position APCB 9.5-1 Guidelines for Fire Protection for Nuclear Power Plants, from Pilant (NPPD) to Ziemann (NRC). Letter dated 4/6/77, Revisions and Additional Information Fire Protection Review, from Pilant (NPPD) to Ziemann (NRC). Letter dated 5/23/79, from Ippolito (NRC) to Pilant (NPPD).

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			insulation burns slowly. Nevertheless, we consider all cable insulation made of organic material as combustible and, therefore, we find that retest to the IEEE 383 procedures and criteria would not provide information that would alter our conclusions. Accordingly, we find the electrical cables used at the Cooper Plant acceptable." The electrical cables, as approved by the SER, are still installed at CNS. There have been no plant modifications or other changes that would invalidate the basis for approval. These cables remain unchanged. <u>Complies with Required Action</u> Implementation Item S-3.12 – Procedures will be revised to require new cable installations to meet the requirements of IEEE-383 or similar. See Attachment S, Table S-3.	
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	Roofs comply with Factor Mutual Class I requirements per the associated references. A Factory Mutual Class I roof is equivalent to a NFPA 256 Class A roof classification. EDP-06, Attachment 1, Section F1 includes the requirements for roof construction, and specifically references NFPA 805, Section 3.3.6, as part of the fire protection design	CED 6013340, Reactor Building Roof Repair and Replacement Contract No. 87-30, Re-roof Various Plant Structures Drawing 4506, Rev. N09, Roof Plan and Details Drawing CNS-BLDG-94, Rev. N05, Re-Roofing Plan and Details Drawing CNS-BLDG-353, Rev. N02,

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			considerations.	CNS Re-Roofing Plan and Details Reactor Building EDP-06, Rev. 45, Supporting Requirements for Configuration Change Control MP 96-122, Steam Tunnel Roof Replacement MWR 97-1891, Maintenance Work Request - Turbine Building Roof MWR 98-4046, Maintenance Work Request Reactor Building Roof PO 351766, Purchase Order - 4160V Switchgear Roof Requisition No. 96-2231, Steam Tunnel Roof
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 Complies with Required Action	<u>Complies</u> Bulk storage of hydrogen gas, in D.O.T.-approved high pressure cylinders, is located in a totally separate building approximately 80 feet east of the Water Treatment Plant. There is no other bulk gas storage other than the hydrogen gas located outside. <u>Complies with Required Action</u> Implementation Item S-3.13 – Procedure 0.7.1 will be revised to include a requirement that bulk gas storage not be allowed inside structures housing systems, equipment, or components important to	Procedure 0.7.1, Rev. 30, Control of Combustibles

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			nuclear safety. See Attachment S, Table S-3.	
3.3.7.1 [Bulk Flammable Gas Storage - Location Requirements]	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 Complies with the use of Existing Engineering Equivalency Evaluations (EEEEs)	<u>Complies</u> Bulk storage of hydrogen gas, in D.O.T.-approved high pressure cylinders, is located in a totally separate building approximately 80 feet east of the Water Treatment Building. The long axis of the hydrogen storage containers is pointed towards the Intake Structure to the north. However, the building is located approximately 100 feet south of the Intake Structure. In addition, the walls of the hydrogen storage structure are constructed of 1 foot reinforced poured concrete, and each hydrogen container is provided with two mounting frames that provide the necessary restraint in the event of failure. <u>Complies with use of EEEEs</u> Hydrogen storage is in compliance with the requirements of NFPA 50A, as documented in the NFPA 50A-1973 code review checklist in Engineering Evaluation EE 10-071.	EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review Drawing 4044, Rev. 1, Gas Bottle Storage Building Drawing 4519, Rev. N01, Gas Bottle Storage Building
3.3.7.2 [Bulk Flammable Gas Storage - Container Restrictions]	Outdoor high-pressure flammable gas storage containers shall be located so that the long axis is not pointed at buildings.	Submit for NRC Approval	Bulk storage of hydrogen gas, in D.O.T.-approved high pressure cylinders, is located in a totally separate building approximately 80 feet east of the Water Treatment Building. The long axis of the hydrogen storage containers is pointed towards the Intake Structure to the north.	Drawing 4003, Rev N39, Overall Site & Vicinity Plan Drawing 4044, Rev. 1, Gas Bottle Storage Building Drawing 4519, Rev. N01, Gas Bottle Storage Building

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			However, the building is located approximately 100 feet south of the Intake Structure. In addition, the walls of the hydrogen storage structure are constructed of 1 foot reinforced poured concrete and each hydrogen container is provided with two mounting frames that provide the necessary restraint in the event of failure. Refer to Attachment L for further details on the request for NRC approval of the current configuration of the bulk hydrogen storage containers inside the hydrogen storage structure with long axis pointed towards the Intake Structure.	
3.3.7.3 [Bulk Flammable Gas Storage - Cylinder Limitations]	Flammable gas storage cylinders not required for normal operation shall be isolated from the system.	Complies	Procedure 0.7.1, Section 4.8 provides for the segregation, storing, and transporting of flammable gases in accordance with site-specific procedures and when flammable gases are used or staged for use in Level 1 plant areas. In these cases, a Transient Combustible Evaluation is processed. The site-specific procedures which control the use and storage of gases include Procedures 0.36 and 0.36.6.	Procedure 0.7.1, Rev. 30, Control of Combustibles Procedure 0.36, Rev. 36, Industrial Safety Procedure Procedure 0.36.6, Rev. 10, Monitoring for Industrial Gases
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	Complies by Previous NRC Approval Complies Complies with use of Existing Engineering Equivalency Evaluations	<u>Complies by Previous NRC Approval</u> The storage areas inside structures containing systems, equipment, or components important to nuclear safety have been previously approved by the NRC. The NPPD response to	EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review Letter dated 12/17/76, Response to Appendix A to Branch Technical Position APCB 9.5-1 Guidelines for Fire Protection for Nuclear Power

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	Liquids Code."	(EEEEs)	Branch Technical Position APCSB 9.5-1, Appendix A, dated 12/17/76 states: "As shown in Fig. 4, both of the 2,500 gallon Diesel Generator Fuel Oil Day tanks are cut off from the remainder of the Diesel Generator Room by a minimum 3 hour rated Fire Barrier, with a Class A fire door. Each doorway to the Day Tank Room has a 2' high curb to contain a total tank spill or rupture. As indicated in the paragraph above, the Day Tank Rooms are protected by CO_2, with detection and actuation by a thermal detector." The NPPD response to Branch Technical Position APCSB 9.5-1, Appendix A, dated 12/17/76 also states: "As shown in Fig. 3 & 4, rooms housing equipment and/or storage containers are cut off from the remainder of the Turbine Generator Building by minimum 3 hour fire-rated masonry Fire Barriers, and are provided with Class A fire doors and HVAC duct penetrations without fire-rated dampers. The Turbine Lube Oil Concentric Piping runs below the Turbine Operating Floor, at approximately the 920' elevation. Fig. 18, 19 & 20 show the Fire Zones which indicate Automatic Water Spray Fire Protection Systems, with Rate-	Plants, from Pilant (NPPD) to Ziemann (NRC). Letter dated 5/23/79, from Ippolito (NRC) to Pilant (NPPD). Procedure 0.7.1, Rev. 30, Control of Combustibles Procedure 0.7.1.1, Rev. 1, Control of Flammable Materials Lockers

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			of-Rise detection and actuation systems, are provided to protect these areas. It should be noted that none of these areas expose safety related systems and equipment. As indicated in Response to D.2(a), the E.H. Governor Control Fluid is of a fire-resistant Quality." Section 5.2.6 of the CNS Fire Protection SER dated 5/23/79 states: "The licensee has proposed to electrically supervise the doors between diesel generator rooms. "A curb will be provided at the entrance to the boiler room to prevent oil from entering the diesel generator room. "The curb between the two diesel rooms will be modified to prevent an oil spill from affecting the adjacent room. "The abort switch for the carbon dioxide systems will be electrically supervised to annunciate in the control room. "We find that, subject to the implementation of the above described modifications, the fire protection of the diesel generator room satisfies the objectives identified in Section 2.2 of this report and is, therefore, acceptable."	

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Section 5.9.6 of the CNS Fire Protection SER dated 5/23/79 states: "The licensee will provide a drainage system for the lube oil reservoir room at elevation 903'. "Information will be provided by the licensee to assure satisfactory fire resistance capability of the penetration barriers. "An automatic sprinkler system will be installed to provide coverage of the vertical electrical cable chase and penetrations in the north wall of the turbine building. "We find that, subject to the implementation of the above modifications, the fire protection for the turbine building satisfies the objectives identified in Section 2.2 of this report and is, therefore, acceptable." These configurations, as approved by the SER, are still used at CNS. There have been no other plant modifications or other changes that would invalidate the basis for approval. This feature remains unchanged. <u>Complies</u> There is no bulk storage of flammable or combustible liquids	

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			inside structures containing systems, equipment, or components important to nuclear safety, with the exception of those previously approved. Procedure 0.7.1, Section 4.5 provides the guidelines for the staging, storage, and use of combustible and flammable liquids. Procedure 0.7.1.1 provides administrative controls for flammable material storage at CNS. <u>Complies with use of EEEEs</u> Bulk storage of flammable and combustible liquids is in accordance with the requirements of NFPA 30, as documented in the NFPA 30-1973 code review checklist in Engineering Evaluation EE 10-071.	
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 Required Action	CNS does not utilize transformer oil drain paths. Spill containment that could accumulate debris are not inspected to prohibit the proper collection of oil. Implementation Item S-3.14 – Procedures will be revised to include the requirement for the inspection of the transformer spill containment area. See Attachment S, Table S-3.	None
3.3.10 Hot Pipes and Surfaces	Combustible liquids, including high flashpoint lubricating oils, shall be kept from coming in contact with hot	Complies	Procedure 0.7.1, Section 4.5.3.1 requires that flammable liquids should not be used in close	Procedure 0.7.1, Rev. 30, Control of Combustibles

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	pipes and surfaces, including insulated pipes and surfaces. Administrative controls shall require the prompt cleanup of oil on insulation.		proximity to sources of ignition, or in such a way that flammable vapors could come in contact with a source of ignition. Section 4.2.8 requires that combustible liquids, or Class A materials impregnated with combustible liquids, should be removed promptly and prevented from contacting hot pipes or other hot surfaces, including those which are insulated. Section 4.2.8.1 provides the requirement for prompt cleanup of oil on insulation. Procedure 7.0.13 provides the method and controls for removing piping, equipment, and HVAC Systems insulation.	Procedure 7.0.13, Rev. 15, Control of Insulation Removal and Installation
3.3.11 Electrical Equipment	Adequate clearance, free of combustible material, shall be maintained around energized electrical equipment.	Complies	Procedure 0.7.1, Section 4.2.3, requires that transient combustible materials should not be stored within 3 feet of energized electrical equipment.	Procedure 0.7.1, Rev. 30, Control of Combustibles
3.3.12 Reactor Coolant Pumps	For facilities with non-inerted containments, reactor coolant pumps with an external lubrication system shall be provided with an oil collection system. The oil collection system shall be designed and installed such that leakage from the oil system is safely contained for off normal conditions such as accident conditions or earthquakes. All of the following shall apply.	N/A	Not Applicable - CNS has an inerted containment.	None
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	N/A	Not Applicable - CNS has an inerted containment.	None

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	system.			
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	Not Applicable - CNS has an inerted containment.	None
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	Not Applicable - CNS has an inerted containment.	None
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	Not Applicable - CNS has an inerted containment.	None
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	Not Applicable - CNS has an inerted containment.	None
3.4 Industrial Fire Brigade	N/A	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
3.4.1 On-Site Fire-Fighting Capability	On-Site Fire-Fighting Capability. All of the following requirements shall apply.	N/A	Introductory Statement - See compliance bases below for compliance statements for specific subsections.	None
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	Complies by Previous NRC Approval	Procedure 0.23, Section 1.1 and USAR Section XIII-10.5.1 describe that a fire brigade of at least five members is maintained at all times. The fire brigade composition may be less than the minimum	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2 Letter dated 7/20/77, Proposed Change to the Radiological Technical Specifications Cooper Nuclear Station, from Pilant (NPPD) to Davis

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	NFPA standards as applicable:		requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of fire brigade members, provided immediate action is taken to restore the fire brigade to within the minimum requirements. This fire brigade staffing configuration, including this reduced composition allowance for a 2-hour time period, was previously approved by the NRC. Letter dated 7/20/77 included the following statement in the proposed Fire Protection System Technical Specifications: "In the event that any member of a minimum shift crew is absent or incapacitated due to illness or injury a qualified replacement shall be designated to report on-site within two hours." Safety Evaluation Report, dated November 29, 1977, approved the Fire Protection System Technical Specifications which includes this reduced composition allowance for a 2-hour time period. The fire brigade staffing, as approved by the NRC Safety Evaluation Report, is still intact at CNS. There have been no changes that would invalidate	(NRC). Letter dated 11/29/77, Cooper Nuclear Station, from Goller (NRC) to Pilant (NPPD). Procedure 0.23, Rev. 64, CNS Fire Protection Plan

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			the basis for approval.	
3.4.1 On-Site Fire-Fighting Capability (a)(1)	NFPA 600, "Standard on Industrial Fire Brigades" (interior structural fire fighting)	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	The fire brigade is in compliance with the requirements of NFPA 600, as documented in the NFPA 600-2000 code review checklist in Engineering Evaluation EE 10-071.	EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review
3.4.1 On-Site Fire-Fighting Capability (a)(2)	NFPA 1500, "Standard on Fire Department Occupational Safety and Health Program"	N/A	Not Applicable - NFPA 1500 (2007) Chapter 1 "Administration" Section 1.3.2 states that this standard does not apply to Industrial Fire Brigades that might also be known as Emergency Brigades, Emergency Response Teams, Response Teams, Fire Teams, Plant Emergency Organizations, or mine emergency response teams. Since CNS has an "Industrial Fire Brigade," this NFPA 805 Element is not applicable.	NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," Rev. 2 NFPA 1500, Standard on Fire Department Occupational Safety and Health Program, 2007 Edition
3.4.1 On-Site Fire-Fighting Capability (a)(3)	NFPA 1582, "Standard on Medical Requirements for Fire Fighters and Information for Fire Department Physicians"	N/A	Not Applicable - NFPA 1582 is not applicable to CNS per the scope statement below, and is consistent with NEI 04-02, Rev. 2, Section K.6 (FAQ 06-0007). NFPA 1582 (2007) Chapter 1 "Administration" Section 1.1.4 states that this standard does not apply to Industrial Fire Brigades that might also be known as Emergency Brigades, Emergency Response Teams, Response Teams, Fire Teams, Plant Emergency Organizations, or mine emergency response teams. Since CNS has an "Industrial Fire Brigade," this NFPA 805 Element is not	NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," Rev. 2

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			applicable.	
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	Per USAR Section XIII-10.5, a fire brigade of at least five members is maintained at all times. This excludes the four members of the minimum shift crew necessary for safe shutdown, and other personnel required for other essential functions during a fire emergency.	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2
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	Procedure 0.23, Attachment 6, Section 1.2.12 requires that the fire brigade shall be a minimum of five (5) individuals. A minimum of three (3) from Operations and two (2) from another department trained in fire fighting techniques to control and extinguish fires. Qualified station operators have a basic knowledge of nuclear safety systems. Procedure 0.23, Attachment 6 defines the training and drills requirements for all fire brigade members to be considered qualified, which includes but is not limited to fire behavior, fire fighting tactics and strategies, radiological and electrical hazards, and damage control and salvage.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.4.1 On-Site Fire-Fighting Capability (d)	The industrial fire brigade shall be notified immediately upon verification of a fire.	Complies	Site-Specific Plant Access Training GEN001-02-04 provides general training to plant employees, which includes notifying the Control Room via the telephone using 911 or gaitronics of a discovered fire. Procedure 5.1INCIDENT, Attachment 1, Section 1.2.1	Procedure 5.1INCIDENT, Rev. 21, Site Emergency Incident GEN001-02-04, Site-Specific Plant Access Training

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			requires that the Control Room is responsible to notify the fire brigade of any fire incident via a Gaitronics announcement.	
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 firefighting operations. The physical examination shall determine the ability of each member to use respiratory protection equipment.	Complies	Procedure 0.23, Section 5.4.2 states Fire Brigade physical examinations are provided annually to Fire Brigade members and Section 5.4.3 states respiratory protection training re-qualification is completed annually.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.4.2 Pre-Fire Plans	Current and detailed pre-fire plans shall be available to the industrial fire brigade for all areas in which a fire could jeopardize the ability to meet the performance criteria described in Section 1.5.	Complies	CNS Pre-Fire Plans are established as detailed drawings under the CNS Drawing Control Program. Procedure 0.23, Section 2.5 requires that copies of these pre-fire plan drawings are retained at strategic locations throughout the plant for use by the fire brigade during a fire event. Procedure 15.FP.650 identifies the inventory of the fire brigade lockers, which includes the pre-fire plans. Procedure 6.FP.608 documents the location of the fire brigade lockers.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan Procedure 6.FP.608, Rev. 23, License Required Fire Fighting Equipment Monthly Examination Procedure 15.FP.650, Rev. 16, Fire Locker and Hazardous Material Response Inventory
3.4.2.1 [Pre-Fire Plans - Contents]	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	Per Section 2.5 of Procedure 0.23, Pre-Fire Plans detail hazardous conditions and firefighting recommendations in certain areas. The Pre-Fire Plans detail an area drawing, potential hazards, cautions, fixed suppression, specialized fire fighting equipment, and the major shutdown equipment	Procedure 0.23, Rev. 64, CNS Fire Protection Plan

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			within various areas of the plant.	
3.4.2.2 [Pre-Fire Plans - Updates]	Pre-fire plans shall be reviewed and updated as necessary.	Complies with Required Action	Procedure 0.23, Section 2.5 requires that Pre-Fire Plan drawings are controlled under the Configuration Change and Drawing Control processes. Changes to these Pre-Fire Plans are distributed by the Fire Protection Staff, the Fire Brigade Training Instructor, or Operations personnel. However, there is no procedure for updating the pre-fire plans in each of the inventoried locations. Implementation Item S-3.16 – Revise procedures to inventory which pre-fire plans are in the fire lockers, and ensure that updates of the pre-fire plans include replacing the updated pages in each of the inventoried locations throughout the plant. See Attachment S, Table S-3.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.4.2.3 [Pre-Fire Plans - Locations]	Pre-fire plans shall be available in the control room and made available to the plant industrial fire brigade.	Complies with Required Action	A copy of the Pre-Fire Plans is contained within the Control Room. In addition, Procedure 0.23, Section 2.5 requires that Pre-Fire Plans are retained at strategic locations throughout the plant for use by the Fire Brigade during a fire event. Implementation Item S-3.17 – Revise procedures to ensure that Pre-Fire Plan drawings, are maintained in the Control Room and to ensure that the latest revisions are available. See Attachment S, Table S-3.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
3.4.2.4 [Pre-Fire Plans - Coordination Needs]	Pre-fire plans shall address coordination with other plant groups during fire emergencies.	Complies with Clarification	Fire brigade coordination with other plant groups during fire emergencies is not addressed in individual pre-fire plans. However, the coordination between the fire brigade and other plant groups during a fire emergency is defined in Procedure 5.1INCIDENT.	Procedure 5.1INCIDENT, Rev. 21, Site Emergency Incident
3.4.3 Training and Drills	Training and Drills. Industrial fire brigade members and other plant personnel who would respond to a fire in conjunction with the brigade shall be provided with training commensurate with their emergency responsibilities.	Complies	Training Program Procedure TPP 206 establishes the fire brigade training program such that training is commensurate with their emergency responsibilities. Training Program Procedure TPP 101 establishes the emergency response organization training program.	TPP 101, Rev. 14, Emergency Response Organization TPP 206, Rev. 19, Fire Brigade
3.4.3 Training and Drills (a)	Plant Industrial Fire Brigade Training. All of the following requirements shall apply.	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
3.4.3 Training and Drills (a)(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 Existing Engineering Equivalency Evaluations (EEEEs)	The fire brigade training program is in compliance with the requirements of NFPA 600, as documented in the NFPA 600-2000 code review checklist in Engineering Evaluation EE 10-071. (NFPA 1500 does not apply, as the plant operates a fire brigade, not a fire department).	EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review
3.4.3 Training and Drills (a)(2)	Industrial fire brigade members shall be given quarterly training and practice in fire fighting, including radioactivity and health physics considerations, to ensure that each member is thoroughly familiar with the steps to be taken in the event of a fire.	Complies	Per Section 5.4 of Procedure 0.23, brigade members are required to attend requalification training on a quarterly basis.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan TPP 206, Rev. 19, Fire Brigade

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
3.4.3 Training and Drills (a)(3)	A written program shall detail the industrial fire brigade training program.	Complies	Procedure 0.23, Section 5 details the fire brigade training program and requirements. Procedure TPP 206, Section 1.1.2 states that this guide covers suggested training aids, strategies, and methods used to conduct the program. Section 2.2.1 details the training documents presented to all fire brigade members, which includes, but is not limited to, training on personal protective equipment, fires and extinguishing agents, pre-fire plans, and hazard identification.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan TPP 206, Rev. 19, Fire Brigade
3.4.3 Training and Drills (a)(4)	Written records that include but are not limited to initial industrial fire brigade classroom and hands-on training, refresher training, special training schools attended, drill attendance records, and leadership training for industrial fire brigades shall be maintained for each industrial fire brigade member.	Complies	Per Section 2.2 of Training Program Procedure TPP 206, initial classroom training sessions, ongoing classroom training sessions, and fire drills are recorded and maintained in the Training Records System. Per Section 2.6 of Attachment 6 of Procedure 0.23, the Training Manager provides overall management of the Training Programs which support the Fire Protection Program. This includes maintenance of the fire brigade training records. The Operations Training Supervisor is responsible for ensuring all drills are scheduled and performed as required.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan TPP 206, Rev. 19, Fire Brigade
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	Complies	Training Program Procedure TPP 101 establishes the emergency response organization training program.	TPP 101, Rev. 14, Emergency Response Organization

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	responsibilities, potential hazards to be encountered, and interfacing with the industrial fire brigade.		Plant personnel who respond with the fire brigade are included in the fire brigade fire drills.	
3.4.3 Training and Drills (c)	Drills. All of the following requirements shall apply.	N/A	Introductory Statement - See compliance bases below for compliance statements for specific subsections.	None
3.4.3 Training and Drills (c)(1)	Drills shall be conducted quarterly for each shift to test the response capability of the industrial fire brigade.	Complies	Procedure 0.23, Section 5.5 states fire drills shall be performed at least once per calendar quarter per Operating Crew.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
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	Procedure 0.23, Section 5.5.4 requires that the CNS Training Department shall develop drill scenarios with Fire Protection staff assistance to determine the effectiveness of: Pre-Fire Plan strategies, Adequacy of equipment, Personal understanding of responsibilities, and Team Work and Communication. Per Attachment 3, a fire drill critique is performed after the drill to determine its effectiveness.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
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 with Required Action	Plant documents do not provide a description of fire drills that indicate the quarterly drills are conducted in various areas of the plant. Implementation Item S-3.18 - Station Fire Brigade Training Program will be updated to include guidance to ensure fire drills are conducted in various plant areas, especially in those areas identified to be essential to plant operation and to contain significant fire hazards. See	Procedure 0.23, Rev. 64, CNS Fire Protection Plan

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Attachment S, Table S-3.	
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	Procedure 0.23, Section 5.5.4.4 requires that fire drills be tracked using the Fire Drill Spreadsheet administered by Operations Training. The spreadsheet is entered into records documenting the completion of all required fire drills. Attachment 2 of this procedure is utilized to authorize and approve drills. Procedure 0.23, Section 5.5 requires that a post-fire drill critique is recorded and maintained. This critique includes an assessment of individual and team performance of the fire brigade. Section 9 establishes plant records requirements. Attachment 3 provides the post-fire drill critique form.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.4.3 Training and Drills (c)(5)	A critique shall be held and documented after each drill.	Complies	Procedure 0.23, Section 5.5 requires that a post-fire drill critique is recorded and maintained. Attachment 3 is utilized for post-fire drill critiques.	Procedure 0.23, Rev. 64, CNS Fire Protection Plan
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	Fire brigade equipment is purchased through qualified vendors and required to meet the latest NFPA Code. Procedure 15.FP.650, Attachment 1 provides the quantity and type of protective equipment located in the fire lockers and hazardous material carts.	Emergency Plan for Cooper Nuclear Station, Rev. 59 Procedure 6.FP.608, Rev. 23, License Required Fire Fighting Equipment Monthly Examination Procedure 15.FP.650, Rev. 16, Fire Locker and Hazardous Material Response Inventory

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Per Section 6.6.1 of the Emergency Plan for Cooper Nuclear Station, emergency workers will wear dosimetry, as required by Radiological Protection personnel. Emergency worker dosimetry will be provided on a 24-hour basis by Radiological Protection personnel. Procedure 6.FP.608 ensures the operability and availability of Fire Brigade equipment.	
3.4.5 Off-Site Fire Department Interface	N/A	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
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	Per Section 5.5.9 of Procedure 0.23, local off-site responders are offered site familiarization training on a recurring basis to share information about the site layout and B.5.b event-related mitigation strategies and measures. Per Section 8.1.3 of the CNS Emergency Plan, training for participating agencies is programmed by the individual agencies with aid from the State Governments in Nebraska, Missouri, Kansas, and Iowa. NPPD personnel are available to describe the special conditions and constraints involved in dealing with the station emergencies and any radiological release situations.	Emergency Plan for Cooper Nuclear Station, Rev. 59 Procedure 0.23, Rev. 64, CNS Fire Protection Plan

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
3.4.5.2 Site-Specific Training	Fire fighters from the off-site fire authorities who are expected to respond to a fire at the plant shall be offered site-specific training and shall be invited to participate in a drill at least annually.	Complies	Procedure 0.23, Section 5.5.8 requires that on an annual basis, a station fire drill is conducted where local fire departments are invited to participate. Per Section 5.5.9, local off-site responders are offered site familiarization training on a recurring basis to share information about the site layout and B.5.b event-related mitigation strategies and measures. Per Section 8.1.3 of the CNS Emergency Plan, NPPD offers training annually for members of the Volunteer Fire Departments of Brownville, Nemaha, Peru, and Auburn. GEN005-10-12 provides training for familiarity with the expected process for and expectations/procedures for responding to CNS in the event of an emergency as a member of an offsite emergency response organization. USAR Section XIII-10.5.1 describes that frequent visits to CNS have been arranged for the outside fire fighting companies from the nearby communities, and that on an annual basis, a Station fire drill is conducted where local Fire Departments are afforded an opportunity to participate.	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2 Emergency Plan for Cooper Nuclear Station, Rev. 59 GEN005-10-12, Rev. 02, Offsite Fire Department Training Procedure 0.23, Rev. 64, CNS Fire Protection Plan
3.4.5.3 Security and	Plant security and radiation	Complies	Security Procedure 3.14	Security Procedure 3.14, Rev. 15,

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
Radiation Protection	protection plans shall address off-site fire authority response.		provides instructions for preparing for off-site emergency responders. Security Procedure 3.15 provides instructions for expediting site access to off-site emergency responders. This includes the proper security and radiological protection controls to allow them to assist plant personnel.	Non-Security Emergencies Security Procedure 3.15, Rev. 10, Emergency Entry/Exit
3.4.6 Communications	An effective emergency communications capability shall be provided for the industrial fire brigade.	Complies	Per Section XIII-10.6.3 of the USAR, the communication system consists of two-way radios and sound power telephones. System Operating Procedures define the operation of the Sound Power System. Procedure 2.2.4, Attachment 2, Section 1.2.5 requires that the site 450 MHz (UHF) radio system uses three repeaters, Base 1, Base 2, and Base 4 with an 800 MHz as Base 3. Base 2 is the primary frequency used by CNS fire brigade.	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2 Procedure 2.2.4, Rev. 43, Communications Systems
3.5 Water Supply	N/A	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
3.5.1 [Water Supply - Flow Code Requirements]	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 (1,135,500-L) supplies. (b) Calculate the fire flow rate for 2 hours. This fire flow rate shall be	Complies	CNS utilizes the method allowed in subsection (a) to comply with Section 3.5.1. Per Section X-9.3.2.1 of the USAR, the fire water supply is stored in two 500,000 gallon capacity fire water tanks. These tanks are vented to atmosphere and provide clean fire water to the electric motor driven pump	Alarm Procedure 2.3_FP-PNL-4, Rev. 3, Pump House Local Control Panel Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	based on 500 gpm (1892.5 L/min) for manual hose streams plus the largest design demand of any sprinkler or fixed water spray system(s) in the power block as determined in accordance with NFPA 13, Standard for the Installation of Sprinkler Systems, or NFPA 15, "Standard for Water Spray Fixed Systems for Fire Protection." The fire water supply shall be capable of delivering this design demand with the hydraulically least demanding portion of fire main loop out of service.		(FP-P-E) and a diesel driven pump (FP-P-D). Makeup water to the fire water storage tanks is normally supplied by the Fresh Well Water Pumping System or the Makeup Water Treatment System. Fire water tank low level annunciation in the Fire Pump House assures each tank will have a minimum level of greater than 26 feet per Procedure 2.3_FP-PNL-4. This correlates to a minimum capacity of 366,000 gallons per tank.	
3.5.2 [Water Supply - Tank Code Requirements]	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 Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	Complies Drawing 2016, Sht. 2 documents that fire pumps can take suction from either or both storage tanks and that a failure in one tank or piping will not allow both tanks to drain. Complies with use of EEEEs The design and installation of the water supply tanks is in accordance with the requirements of NFPA 22, as documented in Engineering Evaluation EE 01-006.	Drawing 2016, Sht. 2, Rev. N31, Fire Protection System Flow Diagram for Pumphouse and Storage Tanks EE 01-006, Rev. 3, Disposition of NFPA Code Compliance Deviations (Systems in Fire Zones Requiring an IPEEE Phase 2 Screening)

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
3.5.3 [Water Supply - Pump Code Requirements]	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 Existing Engineering Equivalency Evaluations (EEEEs) Complies Submit for NRC Approval	Complies with use of EEEEs The design and installation of the fire pumps is in accordance with the requirements of NFPA 20, as documented in Engineering Evaluation EE 01-006. A detailed review of Fire Pump "E" Controller has been performed against the requirements of NFPA 20, as detailed in the NFPA 20-1999 code review checklist in Engineering Evaluation EE 10-071. Complies Per Section X-9.3.2.1 of the USAR, fire pumps FP-P-E and FP-P-D are each rated for 3,000 gpm. They are sized to provide water for the largest fire suppression system demand plus the simultaneous flow of 1,000 gpm from manual hose stations. Per MDC 81-114, the diesel driven fire pump and the electric motor driven fire pump are rated for 3,000 gpm at a total head of 321 feet. Submit for NRC Approval Refer to Attachment L for further details on the request for NRC approval for the remote stop of fire pump FP-P-E from the Control Room.	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2 EE 01-006, Rev. 3, Disposition of NFPA Code Compliance Deviations (Systems in Fire Zones Requiring an IPEEE Phase 2 Screening) EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review MDC 81-114, Fire Protection Clean Water Supply
3.5.4 [Water Supply - Pump Diversity and Redundancy]	At least one diesel engine-driven fire pump or two more seismic Category I Class IE electric motor-	Complies	Per USAR Section X-9.3.2.1, the fire water supply is stored in two 500,000 gallon capacity fire	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	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.		water tanks. These tanks are vented to atmosphere and provide clean fire water to the electric motor driven pump (FP-P-E) and a diesel driven pump (FP-P-D). Makeup water to the fire water storage tanks is normally supplied by the Fresh Well Water Pumping System or the Makeup Water Treatment System. Fire pumps FP-P-E and FP-P-D are each rated for 3,000 gpm. They are sized to provide water for the largest fire suppression system demand plus the simultaneous flow of 1,000 gpm from manual hose stations for two hours. Per MDC 81-114, the diesel driven fire pump and the electric motor driven fire pump are rated for 3,000 gpm at a total head of 321 feet.	MDC 81-114, Fire Protection Clean Water Supply
3.5.5 [Water Supply - Pump Separation Requirements]	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 Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	<u>Complies</u> Per Drawings 4571 and 4572 the Electric Fire Pump Room (Fire Zone 23A) is separated from the Diesel Fire Pump Room (Fire Zone 23B) by a 3-hour rated wall. Per Drawings CNS-FP-182 and CNS-FP-285 Sheet 1 the penetrations within the wall separating the Electric Fire Pump Room from the Diesel Fire Pump Room are provided with 3-hour rated seals. Site Plan Drawing 4003 documents that the Fire	Drawing 4003, Rev. N35, Overall Site & Vicinity Plan Drawing 4571, Rev. N01, Fire Protection Pumphouse Plans and Elevations Drawing 4572, Rev. 1, Fire Protection Pumphouse Sections, Details & Schedule Drawing CNS-FP-285, Rev N04, Sht. 1, CNS Fire Barrier Penetration Seal Details EE 94-8, Rev. 2, Block Wall Gap Separation

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Protection Pumphouse containing the fire pumps is spatially separated from the remainder of the plant. Complies with use of EEEEs Engineering Evaluation EE 94-8 justifies the adequacy of the hairline cracks in the Fire Protection Pumphouse barrier separating the diesel fire pump from the electric fire pump.	Fire Protection Plan CNS-FP-182, Rev N05, Fire Area Boundary Drawing Diesel Driven Fire Pump Room West Wall 903'-6" Fire Pump House
3.5.6 [Water Supply - Pump Start/Stop Requirements]	Fire pumps shall be provided with automatic start and manual stop only.	Complies Submit for NRC Approval	Complies Per Drawings A10-308468 Sht. 1 and Sht. 2, the electric motor driven fire pump FP-P-E is provided with automatic start and manual stop. Per Drawing A10-308583, the engine driven fire pump FP-P-D is provided with automatic start and manual stop. Submit for NRC Approval Refer to Attachment L for further details on the request for NRC approval for the remote stop of fire pump FP-P-E from the Control Room.	Drawing A10-308468 Sht. 1, Rev. N03, Fire Pump Controller 1C Drawing A10-308468 Sht. 2, Rev. N06, Fire Pump Controller 1C Drawing A10-308583, Rev. 3, Engine Driven Fire Pump Controller
3.5.7 [Water Supply - Pump Connection Requirements]	Individual fire pump connections to the yard fire main loop shall be provided and separated with sectionalizing valves between connections.	Complies	Drawing 2016, Sht. 2 documents the sectionalizing valves adequately separating the individual fire pump connections to the yard fire main loop.	Drawing 2016, Sht. 2, Rev. N31, Fire Protection System Flow Diagram for Pumphouse and Storage Tanks
3.5.8 [Water Supply - Pressure Maintenance Limitations]	A method of automatic pressure maintenance of the fire protection water system shall be provided independent of the fire pumps.	Complies	Per USAR Section X-9.3.2.1, a 30 gpm jockey pump maintains system header pressure by automatically starting on low system pressure and stopping when system pressure is	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			restored. If pressure continues to drop (as from a system actuation) FP-P-E will automatically start. A continued drop in system pressure will automatically start FP-P-D.	
3.5.9 [Water Supply - Pump Operation Notification]	Means shall be provided to immediately notify the control room, or other suitable constantly attended location, of operation of fire pumps.	Complies	Drawing SK300, Sht. 2 documents that pump supervisory signals such as pump power and pump running are annunciated in the Control Room.	Drawing SK300, Rev. N11, Sht. 2, Fire Alarm Panel Annunciator Elementary
3.5.10 [Water Supply - Yard Main Code Requirements]	An underground yard fire main loop, designed and installed in accordance with NFPA 24, "Standard for the Installation of Private Fire Service Mains and Their Appurtenances," shall be installed to furnish anticipated water requirements.	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	The design and installation of the underground fire loop is in accordance with the requirements of NFPA 24-1973, as documented in Engineering Evaluation EE 01-009.	EE 01-009, Rev. 3, Review of NFPA 10 & 14 & 24 Code Conformance for Power Block Buildings
3.5.11 [Water Supply - Yard Main Maintenance Issues]	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	The Fire Protection System Flow Drawings 2016 Sht. 1, 2, 3, 6, 7, 1A, 1B, and 1C document that the system has sufficient sectionalizing valves for maintenance or repair without simultaneously shutting off supply to both fixed suppression systems and fire hose stations. Sprinkler systems and standpipe systems are connected such that the systems can be isolated to prevent major impairments to the systems.	Drawing 2016, Sht. 1, Rev. N63, Flow Diagram Fire Protection Turbine Generator Bldg. Drawing 2016, Sht. 1A, Rev. N08, Flow Diagram Fire Protection Service Bldg's &Yard Drawing 2016, Sht. 1B, Rev. N02, Flow Diagram Fire Protection Cont. RDW & ARDW Bldg.'s Drawing 2016, Sht. 1C, Rev. N03, Flow Diagram Fire Protection Reactor Building Drawing 2016, Sht. 2, Rev. N31, Fire Protection System Flow Diagram for Pumphouse and Storage Tanks Drawing 2016, Sht. 3, Rev. N25, Fire

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
				Protection System Flow Diagram Drawing 2016, Sht. 6, Rev. N01, Flow Diagram Fire Protection System Multi Purpose Facility Drawing 2016, Sht. 7, Rev. N07, Fire Protection System Site Plan Flow Diagram
3.5.12 [Water Supply - Compatible Thread Connections]	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	Per the original Response to Appendix A to BTP 9.5-1 submittal dated December 17, 1976, the threads on hydrants, hose couplings, and standpipe risers are compatible with those used by the nearby local community fire departments.	Letter dated 12/17/76, Response to Appendix A to Branch Technical Position APCB 9.5-1 Guidelines for Fire Protection for Nuclear Power Plants, from Pilant (NPPD) to Ziemann (NRC)
3.5.13 [Water Supply - Header Options]	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	E69-4 documents steel piping and fittings meet the requirements ANSI B31.1. Per USAR Section X-9.5, the safety-related Class I systems and equipment necessary for safe shutdown are located only in the Reactor Building, the Control Building, the Diesel Generator Building, and in the Intake Structure. The Fire Protection System piping is Seismic Class IIS. However, the fire water piping and sprinklers in the Reactor Building and Control Building and the high pressure CO_2 system in the Diesel Generator Building have been supported and restrained	Contract E69-4, Mechanical Piping Equipment & Erection Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2 Drawing 2016, Sht. 1, Rev. N63, Flow Diagram Fire Protection Turbine Generator Bldg. Drawing 2016, Sht. 1A, Rev. N08, Flow Diagram Fire Protection Service Bldg's &Yard Drawing 2016, Sht. 1B, Rev. N02, Flow Diagram Fire Protection Cont. RDW & ARDW Bldg.'s Drawing 2016, Sht. 1C, Rev. N03, Flow Diagram Fire Protection Reactor

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			to Seismic Class IS criteria. The Intake Structure fire water piping in the SW Pump Room is supported to the barge impact criteria. In this manner, when Fire Protection System piping passes over or near the Seismic Class IS piping or Class I equipment in the above buildings, it is supported and restrained to withstand a Class I Seismic occurrence and maintain structural and pressure integrity. The Fire Protection System Flow Drawings 2016 Sht. 1, 2, 3, 6, 7, 1A, 1B, and 1C document that each sprinkler and standpipe system is equipped with an outside screw and yoke (OS&Y) gate valve or other approved shutoff valve.	Building Drawing 2016, Sht. 2, Rev. N31, Fire Protection System Flow Diagram for Pumphouse and Storage Tanks Drawing 2016, Sht. 3, Rev. N25, Fire Protection System Flow Diagram Drawing 2016, Sht. 6, Rev. N01, Flow Diagram Fire Protection System Multi Purpose Facility Drawing 2016, Sht. 7, Rev. N07, Fire Protection System Site Plan Flow Diagram
3.5.14 [Water Supply - Control Valve Supervision]	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	Complies	A program at the station requires fire protection valves to be sealed or locked in the normal open position. A periodic recorded surveillance is conducted to ensure that the Fire Protection system valve positions are correct. Procedures 6.FP.201, 6.FP.301, and 6.FP.302 provide verification of the status (locked or sealed) of each fire main and suppression system valve. The Fire Protection System Flow Drawing 2016 Sht. 7 documents that each valve is located within the CNS Protected Area.	Drawing 2016, Sht. 7, Rev. N07, Fire Protection System Site Plan Flow Diagram Procedure 6.FP.201, Rev. 15, Operations Cycling of Fire Main Valves Procedure 6.FP.301, Rev. 17, Operations Power Block Sprinkler System Testing Procedure 6.FP.302, Rev. 22, Automatic Deluge and Pre-Action Systems Testing

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	positions. This option shall be utilized only where valves are located within fenced areas or under the direct control of the owner/operator.			
3.5.15 [Water Supply - Hydrant Code Requirements]	Hydrants shall be installed approximately every 250 ft (76 m) apart on the yard main system. A hose house equipped with hose and combination nozzle and other auxiliary equipment specified in NFPA 24, "Standard for the Installation of Private Fire Service Mains and Their Appurtenances," shall be provided at intervals of not more than 1000 ft (305 m) along the yard main system. *Exception: Mobile means of providing hose and associated equipment, such as hose carts or trucks, shall be permitted in lieu of hose houses. Where provided, such mobile equipment shall be equivalent to the equipment supplied by three hose houses.*	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs) Complies	<u>Complies with use of EEEEs</u> Hose house equipment is in accordance with the requirements of NFPA 24-1973, as documented in Engineering Evaluation EE 01-009. <u>Complies</u> Per USAR Section X-9.3.2.1, an outside, 12-inch, underground yard loop surrounds the Station and provides water to hydrants, wet standpipes, hose stations, deluge spray systems, and sprinkler systems. Hydrants with two gated discharge ports are provided on the yard main at approximately 250-foot intervals. Fire hydrants are provided with an isolation valve in order to isolate the hydrant in the event of physical damage or mechanical malfunction. Procedure 6.FP.608, Attachment 2 documents the location and equipment provided in each outside hose cabinet. Per Section 4.3.1.3 of the CNS Fire Protection SER dated 5/23/79, hose houses have been provided at, as a minimum, every other hydrant.	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2 EE 01-009, Rev. 3, Review of NFPA 10 & 14 & 24 Code Conformance for Power Block Buildings Letter dated 5/23/79, from Ippolito (NRC) to Pilant (NPPD). Procedure 6.FP.608, Rev. 23, License Required Fire Fighting Equipment Monthly Examination
3.5.16 [Water Supply - Dedicated Limits]	The fire protection water supply system shall be dedicated for fire protection use only.	Complies Complies with Required	<u>Complies</u> The Fire Protection System Flow Drawings 2016 Sht. 1, 2, 3, 6, 7,	Drawing 2016, Sht. 1, Rev. N63, Flow Diagram Fire Protection Turbine Generator Bldg.

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	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.*	Action	1A, 1B, and 1C document that the water tanks, fire pumps, and piping network are dedicated for fire protection use only. Complies with Required Action Implementation Item S-3.22 – Procedures will be revised to ensure that the fire protection system is not to be used for non-emergency usage. See Attachment S, Table S-3.	Drawing 2016, Sht. 2, Rev. N31, Fire Protection System Flow Diagram for Pumphouse and Storage Tanks Drawing 2016, Sht. 3, Rev. N25, Fire Protection System Flow Diagram Drawing 2016, Sht. 6, Rev. N01, Flow Diagram Fire Protection System Multi Purpose Facility Drawing 2016, Sht. 7, Rev. N07, Fire Protection System Site Plan Flow Diagram Drawing 2016, Sht. 1A, Rev. N08, Flow Diagram Fire Protection Service Bldg's &Yard Drawing 2016, Sht. 1B, Rev. N02, Flow Diagram Fire Protection Cont. RDW & ARDW Bldg.'s Drawing 2016, Sht. 1C, Rev. N03, Flow Diagram Fire Protection Reactor Building
3.6 Standpipe and Hose Stations	N/A	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
3.6.1 [Standpipe and Hose Stations - Code Requirements]	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 by Previous NRC Approval Complies with use of Existing Engineering Equivalency Evaluations (EEEEs) Submit for NRC Approval	Complies by Previous NRC Approval CNS utilizes a Class II standpipe service, not Class III as required by this section. The system was previously approved by the NRC. The NPPD response to Branch Technical Position APCSB 9.5-1, Appendix A, dated 12/17/76, states:	EE 01-009, Rev. 3, Review of NFPA 10 & 14 & 24 Code Conformance for Power Block Buildings Letter dated 12/17/76, Response to Appendix A to Branch Technical Position APCB 9.5-1 Guidelines for Fire Protection for Nuclear Power Plants, from Pilant (NPPD) to Ziemann (NRC).

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			"As a result of the October 1976 site inspection, it was found that a minimum of five additional 1 1/2" fire hose connections are needed to provide application of at least one effective hose stream to all areas of the Main Plant. At present, a survey is underway to determine which areas of the Main Plant are not provided with manual back-up fire hose coverage. As a result of this survey, consideration will be given to providing an additional 25' length of 1 1/2" woven jacket lined fire hose to the existing 75' length of fire hose and replacement of the existing hose racks with hose reels... "...All existing inside 1 1/2" fire hose connections are provided with combination spray nozzles suitable for Class A & B fires. These nozzles will be replaced with nozzles suitable for use on Class A, B & C type fires..." Section 4.3.14 of the CNS Fire Protection SER dated 5/23/79 states: "Fifty-four interior hose stations are strategically located throughout the plant. Each hose station is equipped with 75 feet of 1-1/2 inch woven jacketed rubber lined hose with adjustable fog nozzles...	Letter dated 5/23/79, from Ippolito (NRC) to Pilant (NPPD). MDC 77-015, Fire Protection Modifications

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			"Four additional fire hose stations will be provided in the reactor building to adequately protect the area. "Additional hose (25 feet) will be provided for the computer room and 903' elevation of the control building. With the addition of these lengths of hose, all safety related areas can be reached by at least one hose stream. Some interior nozzles will be replaced by spray only types. "We find, that subject to the above described modifications, the hose stations satisfy the objectives identified in Section 2.2 of this report and are, therefore, acceptable." Modification MDC 77-015 installed the additional four hose stations in the Reactor Building and installed the additional 25 feet of hose. The hose station and standpipe configuration, as approved by the SER, is still used at CNS. There have been no additional plant modifications or other changes that would invalidate the basis for approval. This feature remains unchanged. <u>Complies with use of EEEEs</u> Installation of the Class II standpipe and hose systems is in accordance with the requirements of NFPA 14-1974, as documented in Engineering	

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			Evaluation EE 01-009. Submit for NRC Approval Refer to Attachment L for further details on the request for NRC approval for the deviation against Sections 322, 442, 625, and 671 of NFPA 14-1974 for the standpipe and hose station installations.	
3.6.2 [Standpipe and Hose Stations - Capability Limitations]	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	Hydraulic calculation 9.41-001 demonstrates adequate water flow rate and nozzle pressure for all hose stations.	Calculation 9.41-001, Fire Flow, Pressure Required, and Storage Requirements, Rev. 0
3.6.3 [Standpipe and Hose Stations - Nozzle Restrictions]	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	Per USAR Section X-9.3.2.2, wet standpipe hose stations are located throughout the plant in strategic locations to assure hose stream coverage and to serve as backup for fixed suppression systems. Hose stations have either 75 or 100 feet of 1 1/2-inch lined hose, as deemed necessary. Where appropriate, hoses are supplied with a nozzle suitable for use on Class A, B, and C type fires.	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2
3.6.4 [Standpipe and Hose Stations - Earthquake Provisions]	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	Complies by Previous NRC Approval	The licensing of CNS preceded the issuance of NUREG-75/087, Section 9.5-1, Rev. 1, which required the fire suppression system to be capable of delivering water to manual hose	Letter dated 12/4/1972, Amendment No. 15 to License Application Filed on July 27, 1967, from Reder (NPPD) to Muntzing (U.S. Atomic Energy Commission)

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	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.		stations located within hose reach of areas containing equipment required for safe plant shutdown following the safe shutdown earthquake (SSE). However, in response to Final Safety Analysis Report (FSAR) Question and Answer 10.15, NPPD committed in FSAR Amendment 15 that whenever fire protection piping passes over or near the Class IS piping or equipment in the Reactor Building, Control Building, or Intake Structure (the building containing Essential Class IS equipment necessary for safe shutdown serviced by the Fire Water System), it will be supported and restrained to withstand a Class IS seismic occurrence and maintain structural and pressure integrity. This was partially acknowledged in Section 9.3.4, "The applicant agreed to modify Fire Protection System piping to Class I (seismic) standards in all areas where it passed over or near the Class I (seismic) systems." Furthermore, USAR Section X-9.5 describes that the entire Fire Protection System piping and sprinklers in the Reactor Building and Control Building as being supported and restrained to Seismic Class IS criteria. The Fire Protection System piping in the SW Pump Room is supported to the barge impact criteria, which is bounding over SSE.	NUREG-75/087, Standard Review Plan, Section 9.5-1, Fire Protection Program, Rev. 1, dated May 1976 Report dated 2/14/1973, Safety Evaluation of the Cooper Nuclear Station, U.S. Atomic Energy Commission.

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
3.6.5 [Standpipe and Hose Stations - Seismic Connection Limitations]	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	Not Applicable. The fire protection system seismically-restrained piping is not cross-connected to any essential non-fire systems.	None
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 Existing Engineering Equivalency Evaluations (EEEEs) Submit for NRC Approval	<u>Complies with use of EEEEs</u> Portable fire extinguishers are in accordance with the requirements of NFPA 10-1975, as documented in Engineering Evaluation EE 01-009. <u>Submit for NRC Approval</u> Refer to Attachment L for further details on the request for NRC approval for the deviation against Section 3-3 of NFPA 10-1975 for fire extinguisher size and placement for Class B fires.	Engineering Evaluation EE 01-009, Rev. 3, Review of NFPA 10 & 14 & 24 Code Conformance for Power Block Buildings
3.8 Fire Alarm and Detection Systems	N/A	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
3.8.1 Fire Alarm	Alarm initiating devices shall be installed in accordance with NFPA 72, "National Fire Alarm Code®." Alarm annunciation shall allow the proprietary alarm system to transmit fire-related alarms, supervisory signals, and trouble signals to the control room or other constantly attended location from which required notifications and response can be initiated. Personnel assigned to the proprietary alarm station shall be permitted to have other duties. The following fire-related signals shall be transmitted:	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs) Complies	<u>Complies with use of EEEEs</u> The fire alarm system is in accordance with the requirements of NFPA 72A-1975 and NFPA 72D-1975, as documented in Engineering Evaluations EE 01-006, EE 01-013, and EE 01-014. <u>Complies</u> Schematic drawings SK300, Sht. 1, 2, and 3 document alarm annunciation of transmit fire-related alarms, supervisory signals, and trouble signals in the Control Room.	Drawing SK300, Sht. 1, Rev. N10, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 2, Rev. N11, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 3, Rev. N05, Fire Panel Alarm Window #5 Annunciator Elementary EE 01-006, Rev. 3, Disposition of NFPA Code Compliance Deviations (Systems in Fire Zones Requiring an IPEEE Phase 2 Screening)

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
				EE 01-013, Rev. 2, Disposition of NFPA Code Deviations (Non Phase 2 IPEEE Power Block Systems) EE 01-014, Rev. 0, Disposition of NFPA Code Compliance Deviations (Non-Power Block Systems)
3.8.1 Fire Alarm (1)	Actuation of any fire detection device	Complies	Schematic drawings SK300, Sht. 1, 2, and 3 document the Control Room fire panel alarm annunciation for fire detection devices.	Drawing SK300, Sht. 1, Rev. N10, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 2, Rev. N11, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 3, Rev. N05, Fire Panel Alarm Window #5 Annunciator Elementary
3.8.1 Fire Alarm (2)	Actuation of any fixed fire suppression system	Complies	Schematic drawings SK300, Sht. 1, 2, and 3 document the Control Room fire panel alarm annunciation for fixed fire suppression systems.	Drawing SK300, Sht. 1, Rev. N10, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 2, Rev. N11, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 3, Rev. N05, Fire Panel Alarm Window #5 Annunciator Elementary
3.8.1 Fire Alarm (3)	Actuation of any manual fire alarm station	Complies	Schematic drawing SK300, Sht. 1 documents the Control Room fire panel alarm annunciation for manual pull stations.	Drawing SK300, Sht. 1, Rev. N10, Fire Alarm Panel Annunciator Elementary
3.8.1 Fire Alarm (4)	Starting of any fire pump	Complies	Schematic drawing SK300, Sht. 2 documents the Control Room fire panel alarm annunciation for the starting of the fire pumps.	Drawing SK300, Sht. 2, Rev. N11, Fire Alarm Panel Annunciator Elementary
3.8.1 Fire Alarm (5)	Actuation of any fire protection supervisory device	Complies	Schematic drawings SK300, Sht. 1, 2, and 3 document the Control Room fire panel alarm	Drawing SK300, Sht. 1, Rev. N10, Fire Alarm Panel Annunciator Elementary

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			annunciation for fire protection supervisory devices.	Drawing SK300, Sht. 2, Rev. N11, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 3, Rev. N05, Fire Panel Alarm Window #5 Annunciator Elementary
3.8.1 Fire Alarm (6)	Indication of alarm system trouble condition	Complies	Schematic drawings SK300, Sht. 1, 2, and 3 document the Control Room fire panel alarm annunciation for alarm system trouble conditions.	Drawing SK300, Sht. 1, Rev. N10, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 2, Rev. N11, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 3, Rev. N05, Fire Panel Alarm Window #5 Annunciator Elementary
3.8.1.1 [Fire Alarm - Communications Requirements]	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	USAR Section X-16 documents the communication systems are available to allow a person observing a fire to effectively communicate to the Control Room. Procedure 5.7COMMUN, Attachment 1 details the communications systems which are installed at the plant and basic instructions for their operation, which includes portable radios, plant phones, and gaitronics.	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2 Procedure 5.7COMMUN, Rev. 16, Communications
3.8.1.2 [Fire Alarm - Prompt Notification Limits]	Means shall be provided to promptly notify the following of any fire emergency in such a way as to allow them to determine an appropriate course of action:	N/A	Introductory Statement - See compliance bases below for compliance statements for specific subsections.	None
3.8.1.2 [Fire Alarm - Prompt Notification	General site population in all occupied areas	Complies	USAR Section X-16 describes that emergency signals (fire,	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
Limits] (1)			evacuation, all clear) are generated by a tone generator located in the Control Room. The emergency signals are carried over the paging channel and all of the speakers.	
3.8.1.2 [Fire Alarm - Prompt Notification Limits] (2)	Members of the industrial fire brigade and other groups supporting fire emergency response	Complies	USAR Section X-16 describes that emergency signals (fire, evacuation, all clear) are generated by a tone generator located in the Control Room. The emergency signals are carried over the paging channel and all of the speakers.	Cooper Nuclear Station Updated Safety Analysis Report, loep xxv2
3.8.1.2 [Fire Alarm - Prompt Notification Limits] (3)	Off-site fire emergency response agencies. Two independent means shall be available (e.g., telephone and radio) for notification of off-site emergency services.	Complies	Per Procedure 5.1INCIDENT, Attachment 2 requires that the Shift Manager is responsible for contacting the Nebraska Emergency Management Agency NEMA for additional fire response assistance. The Nebraska Emergency Management Agency NEMA is available by two different telephone numbers. Procedure 5.7COMMUN, Section 12.1 requires that a cross-band, two-way radio communications system exists between CNS and the Nemaha County Sheriff's Office as an additional source of communication between the plant and the off-site emergency response teams.	Procedure 5.1INCIDENT, Rev. 21, Site Emergency Incident Procedure 5.7COMMUN, Rev. 16, Communications
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	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	The fire detection devices are installed in accordance with the requirements of NFPA 72E-1974, as documented in Engineering Evaluations EE	EE 01-007, Rev. 2, Fire Detector Location and Spacing EE 01-013, Rev. 2, Disposition of NFPA Code Deviations (Non Phase 2

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	NFPA 72, "National Fire Alarm Code," and its applicable appendixes.		01-007, EE 01-013, and EE 01-014. The systems credited to meet the requirements of Chapter 4 are identified in Table 4-3 of the Transition Report.	IPEEE Power Block Systems) EE 01-014, Rev. 0, Disposition of NFPA Code Compliance Deviations (Non-Power Block Systems)
3.9 Automatic and Manual Water-Based Fire Suppression Systems	N/A	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
3.9.1 [Automatic and Manual Water-Based Fire Suppression Systems - Code Requirements]	If an automatic or manual water-based fire suppression system is required to meet the performance or deterministic requirements of Chapter 4, then the system shall be installed in accordance with the appropriate NFPA standards including the following:	N/A	Introductory Statement - See compliance bases below for compliance statements for specific subsections.	None
3.9.1 [Automatic and Manual Water-Based Fire Suppression Systems - Code Requirements] (1)	NFPA 13, "Standard for the Installation of Sprinkler Systems"	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	The automatic sprinkler systems are installed in accordance with the requirements of the applicable code of record as documented in Engineering Evaluations EE 01-006, EE 01-013, and EE 01-014. The systems credited to meet the requirements of Chapter 4 are identified in Table 4-3 of the Transition Report.	EE 01-006, Rev. 3, Disposition of NFPA Code Compliance Deviations (Systems in Fire Zones Requiring an IPEEE Phase 2 Screening) EE 01-013, Rev. 2, Disposition of NFPA Code Deviations (Non Phase 2 IPEEE Power Block Systems) EE 01-014, Rev. 0, Disposition of NFPA Code Compliance Deviations (Non-Power Block Systems)
3.9.1 [Automatic and Manual Water-Based Fire Suppression Systems - Code Requirements] (2)	NFPA 15, "Standard for Water Spray Fixed Systems for Fire Protection"	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	The automatic water spray systems are installed in accordance with the requirements of the applicable code of record as documented in Engineering Evaluations EE 01-006 and EE 01-013. The systems credited to meet the requirements of Chapter 4 are identified in Table 4-3 of the Transition Report.	EE 01-006, Rev. 3, Disposition of NFPA Code Compliance Deviations (Systems in Fire Zones Requiring an IPEEE Phase 2 Screening) EE 01-013, Rev. 2, Disposition of NFPA Code Deviations (Non Phase 2 IPEEE Power Block Systems)

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
3.9.1 [Automatic and Manual Water-Based Fire Suppression Systems - Code Requirements] (3)	NFPA 750, "Standard on Water Mist Fire Protection Systems"	N/A	Not Applicable. There are no water mist systems at CNS.	None
3.9.1 [Automatic and Manual Water-Based Fire Suppression Systems - Code Requirements] (4)	NFPA 16, "Standard for the Installation of Foam-Water Sprinkler and Foam-Water Spray Systems"	N/A	Not Applicable. There are no foam-water sprinkler or foam-water spray systems at CNS.	None
3.9.2 [Automatic and Manual Water-Based Fire Suppression Systems - Flow Alarm]	Each system shall be equipped with a water flow alarm.	Complies	Water flow alarms are installed on all systems per review of associated references.	Drawing 2016, Sht. 1, Rev. N62, Flow Diagram Fire Protection Turbine Generator Bldg. Drawing 2016, Sht. 1A, Rev. N08, Flow Diagram Fire Protection Service Bldg's & Yard Drawing 2016, Sht. 1B, Rev. N02, Flow Diagram Fire Protection Cont. RDW & ARDW Bldg.'s Drawing 2016, Sht. 1C, Rev. N03, Flow Diagram Fire Protection Reactor Building Procedure 6.FP.301, Rev. 17, Operations Power Block Sprinkler System Testing Procedure 6.FP.302, Rev. 22, Automatic Deluge and Pre-Action Systems Testing Procedure 6.FP.307, Rev. 19, Operations Out-Building Sprinkler System Testing Procedure 15.FP.648, Rev. 5, Outside Transformer Deluge System Flow Test

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
3.9.3 [Automatic and Manual Water-Based Fire Suppression Systems - Alarm Locations]	All alarms from fire suppression systems shall annunciate in the control room or other suitable constantly attended location.	Complies	Schematic drawings SK300, Sht. 1, 2, and 3 document the Control Room fire panel alarm annunciation for fire suppression system alarms.	Drawing SK300, Sht. 1, Rev. N10, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 2, Rev. N11, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 3, Rev. N05, Fire Panel Alarm Window #5 Annunciator Elementary
3.9.4 [Automatic and Manual Water-Based Fire Suppression Systems - Diesel Pump Sprinkler Protection]	Diesel-driven fire pumps shall be protected by automatic sprinklers.	Complies	Drawing 2016, Sht. 2 documents that the Diesel Driven Fire Pump Room is provided with an automatic wet pipe sprinkler system.	Drawing 2016, Sht. 2, Rev. N31, Fire Protection System Flow Diagram for Pumphouse and Storage Tanks
3.9.5 [Automatic and Manual Water-Based Fire Suppression Systems - Shutoff Controls]	Each system shall be equipped with an OS&Y gate valve or other approved shutoff valve.	Complies	The referenced flow diagrams show valves for each suppression system.	Drawing 2016, Sht. 1, Rev. N63, Flow Diagram Fire Protection Turbine Generator Bldg. Drawing 2016, Sht. 2, Rev. N31, Fire Protection System Flow Diagram for Pumphouse and Storage Tanks Drawing 2016, Sht. 3, Rev. N25, Fire Protection System Flow Diagram Drawing 2016, Sht. 6, Rev. N01, Flow Diagram Fire Protection System Multi Purpose Facility Drawing 2016, Sht. 7, Rev. N07, Fire Protection System Site Plan Flow Diagram Drawing 2016, Sht. 1A, Rev. N08, Flow Diagram Fire Protection Service Bldg's & Yard Drawing 2016, Sht. 1B, Rev. N02, Flow Diagram Fire Protection Cont.

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
				RDW & ARDW Bldg.'s Drawing 2016, Sht. 1C, Rev. N03, Flow Diagram Fire Protection Reactor Building
3.9.6 [Automatic and Manual Water-Based Fire Suppression Systems - Valve Supervision]	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	A program at the station requires fire protection valves to be sealed or locked in the normal open position. A periodic recorded surveillance is conducted to ensure that the Fire Protection system valve positions are correct. Procedures 6.FP.201, 6.FP.301, and 6.FP.302 provide verification of the status (locked or sealed) of each fire main and suppression system valve. The Fire Protection System Flow Drawing 2016 Sht. 7 documents that each valve is located within the CNS Protected Area.	Drawing 2016, Sht. 7, Rev. N07, Fire Protection System Site Plan Flow Diagram Procedure 6.FP.201, Rev. 15, Operations Cycling of Fire Main Valves Procedure 6.FP.301, Rev. 17, Operations Power Block Sprinkler System Testing Procedure 6.FP.302, Rev. 22, Automatic Deluge and Pre-Action Systems Testing
3.10 Gaseous Fire Suppression Systems	N/A	N/A	Section Heading - See compliance bases below for compliance statements for specific subsections.	None
3.10.1 [Gaseous Fire Suppression Systems - Code Requirements]	If an automatic total flooding and local application gaseous fire suppression system is required to meet the performance or deterministic requirements of Chapter 4, then the system shall be designed and installed in accordance with the following applicable NFPA codes:	N/A	Introductory Statement - See compliance bases below for compliance statements for specific subsections.	None
3.10.1 [Gaseous Fire Suppression Systems - Code Requirements] (1)	NFPA 12, "Standard on Carbon Dioxide Extinguishing Systems"	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	The carbon dioxide systems are installed in accordance with the requirements of NFPA 12-1972 as documented in Engineering Evaluation EE 01-006. The systems credited to meet the	EE 01-006, Rev. 3, Disposition of NFPA Code Compliance Deviations (Systems in Fire Zones Requiring an IPEEE Phase 2 Screening)

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			requirements of Chapter 4 are identified in Table 4-3 of the Transition Report.	
3.10.1 [Gaseous Fire Suppression Systems - Code Requirements] (2)	NFPA 12A, "Standard on Halon 1301 Fire Extinguishing Systems"	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	The Halon systems in the Computer Room and SW Pump Room are installed in accordance with the requirements of NFPA 12A-1980 as documented in Engineering Evaluations EE 01-006 and EE 01-013. The systems credited to meet the requirements of Chapter 4 are identified in Table 4-3 of the Transition Report.	EE 01-006, Rev. 3, Disposition of NFPA Code Compliance Deviations (Systems in Fire Zones Requiring an IPEEE Phase 2 Screening) EE 01-013, Rev. 2, Disposition of NFPA Code Deviations (Non Phase 2 IPEEE Power Block Systems)
3.10.1 [Gaseous Fire Suppression Systems - Code Requirements] (3)	NFPA 2001, "Standard on Clean Agent Fire Extinguishing Systems"	N/A	Not Applicable. There are no clean agent fire extinguishing systems installed.	None
3.10.2 [Gaseous Fire Suppression Systems - Alarm Location]	Operation of gaseous fire suppression systems shall annunciate and alarm in the control room or other constantly attended location identified.	Complies	Carbon dioxide system connection diagrams document annunciation and alarm of the operation of the CO_2 systems in the Control Room. Schematic drawings SK300, Sht. 2 and 3 document the Control Room fire panel alarm annunciation for CO_2 and Halon suppression system alarms.	Drawing FH-16282, Sht. 3, Rev. N07, Cardox High Press Fire Extinguishing System Elementary Line Connection Diagram Drawing FL-16551, Sht. 6, Rev. N05, Cardox Fire Extinguishing System Elementary Line & Connection Diagram Drawing SK300, Sht. 2, Rev. N11, Fire Alarm Panel Annunciator Elementary Drawing SK300, Sht. 3, Rev. N05, Fire Panel Alarm Window #5 Annunciator Elementary
3.10.3 [Gaseous Fire Suppression Systems - Ventilation Limitations]	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 Complies with Clarification	Complies Drawing 2221 indicates that pressure relief dampers are provided for the DG Room CO_2 systems. System Pre-Operational Test BR-67 documents that concentration	Drawing 2221, Rev. N03, H.V.A.C-Plan & Sections Diesel Generator Bldg. Heating Boiler Room System Pre-Operational Test BR-67, dated 04/04/1973

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			and hold time requirements were met showing that adequate sealing is provided. <u>Complies with Clarification</u> Halon activation does result in an over-pressurization in Halon concentrations of 10% or greater per NFPA 12A-2009 Annex B. Halon systems installed at CNS are designed to maintain concentrations below 10% during system activation. The SW Pump Room concentration is 8% and the Computer Room is 6%. These systems are designed with safety margins of 20% and 40%, respectively. There are no radioactive contaminants in the areas provided with Halon or CO_2 system protection.	
3.10.4 [Gaseous Fire Suppression Systems - Single Failure Limits]	In any area required to be protected by both primary and backup gaseous fire suppression systems, a single active failure or a crack in any pipe in the fire suppression system shall not impair both the primary and backup fire suppression capability.	N/A	Not Applicable. No areas are protected by both primary and backup gaseous suppression systems.	None
3.10.5 [Gaseous Fire Suppression Systems - Disarming Controls]	Provisions for locally disarming automatic gaseous suppression systems shall be secured and under strict administrative control.	Complies Submit for NRC Approval	<u>Complies</u> The SW Pump Room and Computer Room Halon systems are provided with keyed abort switches. Operation of the abort switch will annunciate in the Control Room. <u>Submit for NRC Approval</u>	DC 85-01, Halon 1301 Fire Suppression System for Service Water Pump Room Drawing FH-16282, Sht. 3, Rev. N07, Cardox High Press Fire Extinguishing System Elementary Line Connection Diagram

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			The Diesel Generator High Pressure Cardox System, although not provided with keyed abort switches, is provided with abort switches that will annunciate in the Control Room upon operation. Refer to Attachment L for further details on the request for NRC approval of the lack of local keyed abort switches.	Procedure 6.FP.305, Rev. 11, Halon 1301 Service Water Pump Room Fire Suppression Surveillance Checks Procedure 15.FP.307, Rev. 6, Halon 1301 Computer Room Fire Suppression Surveillance Checks
3.10.6 [Gaseous Fire Suppression Systems - CO_2 Limitations]	Total flooding carbon dioxide systems shall not be used in normally occupied areas.	Complies	The Diesel Generator Rooms are not normally occupied areas.	CNS Fire Hazards Analysis, Rev. 9/26/2011
3.10.7 [Gaseous Fire Suppression Systems - CO_2 Warnings]	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 Submit for NRC Approval	Complies Carbon dioxide system connection diagrams document an audible alarm of the operation of the CO_2 systems in the Control Room. Submit for NRC Approval The DG High Pressure CO_2 System is not provided with an odorizer. Refer to Attachment L for further details on the request for NRC approval of the lack of an odorizer on the DG High Pressure CO_2 System.	Drawing FH-16282, Sht. 3, Rev. N07, Cardox High Press Fire Extinguishing System Elementary Line Connection Diagram Drawing FL-16551, Sht. 6, Rev. N05, Cardox Fire Extinguishing System Elementary Line & Connection Diagram
3.10.8 [Gaseous Fire Suppression Systems - CO_2 Required Disarming]	Positive mechanical means shall be provided to lock out total flooding carbon dioxide systems during work in the protected space.	Complies	The referenced drawings and procedures document the mechanical lock out capability via transfer selector switch for the DG Room High Pressure CO_2 System.	Drawing FH-16282, Sht. 3, Rev. N07, Cardox High Press Fire Extinguishing System Elementary Line Connection Diagram Drawing FL-16551, Sht. 6, Rev. N05, Cardox Fire Extinguishing System Elementary Line & Connection Diagram Procedure 2.2.2, Rev. 35, Carbon

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
				Dioxide Systems
3.10.9 [Gaseous Fire Suppression Systems - Cooling Considerations]	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	Halon does not have a potential for thermal shock damage. Appendix B of the Fire Hazards Analysis (FHA) assesses the possibility of secondary thermal shock (cooling) damage due to discharge of the turbine bearing and DG Room CO_2 systems.	CNS Fire Hazards Analysis, Rev. 9/26/2011
3.10.10 [Gaseous Fire Suppression Systems - Decomposition Issues]	Particular attention shall be given to corrosive characteristics of agent decomposition products on safety systems.	Complies	CNS has Halon and CO_2 systems which do not have any corrosive characteristics of agent decomposition products on safety systems.	NFPA Fire Protection Handbook, 20th edition
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	Introductory Statement - See compliance bases below for compliance statements for specific subsections.	None
3.11.1 Building Separation	Each major building within the power block shall be separated from the others by barriers having a designated fire resistance rating of 3 hours or by open space of at least 50 ft (15.2 m) or space that meets the requirements of NFPA 80A, "Recommended Practice for Protection of Buildings from Exterior Fire Exposures." *Exception: Where a performance-*	Complies Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	<u>Complies</u> Table 3 of the Fire Hazards Analysis (FHA) provides a listing of required fire barriers, with their plant orientation, adjacent zones and associated barrier type (Appendix A, Appendix R or Common to both Appendix A and Appendix R). In addition, the individual FHA Fire Zone Matrix Tables, which are located in Section 9 of the FHA, provide	CNS Fire Hazards Analysis, Rev. 9/26/2011 Drawing 4003, Rev. N39, Overall Site & Vicinity Plan Drawing 4058, Rev. N02, Structural Turbine Generator Building Foundation Walls and Sections Drawing 4059, Rev. 13, Structural Turbine Generator Building Concrete

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	based analysis determines the adequacy of building separation, the requirements of 3.11.1 shall not apply.		a detailed fire barrier listing for the applicable Fire Zone. This matrix identifies fire barrier orientation, adjacent Fire Zone, applicable CNS-FP drawing, construction features, fire rating, and associated barrier type (Appendix A, Appendix R or Common). Fire barrier ratings are assigned based on the construction features associated with the barrier. Construction features such as barrier material, barrier thickness, fire damper design, fire door design, and penetration seal assemblies were reviewed as part of assigning a rating to an individual fire barrier. The Reactor Building, Turbine Building, Control Building, Diesel Generator Building, Radwaste Building, Augmented Radwaste Building, Multi-Purpose Facility, and other buildings of the Power Block are separated from the Intake Structure, Hydrogen Storage area, Off-Gas Building, ISFSI, flammable liquid storage trailer, warehouses, and Security Building by open space greater than 50 feet where facing such buildings. All outdoor oil-filled transformers are more than 50 feet away from any opening in the exterior wall of the buildings containing safety related equipment, with the exception of the spare Main Transformer.	Walls Sht. No.1 Drawing 4123, Rev. N03, Structural Diesel Generator Building Plan at Elev. 903'-6" & 917'-6" Drawing 4173, Rev. N01, Structural Control Building Cable Room - Plans & Sections Drawing 4174, Rev. N05, Structural Control Building Basement Plan & Sections Drawing 4175, Rev. N02, Structural Control Building Operating Floor - Plan and Sections Drawing 4180, Rev. N05, Structural Control Building Corridor Sh 1 Drawing 4219, Rev. N05, Structural Reactor Bldg. EL. 903'-6" Plan, Sects., & Dets. SH #1 Drawing 4221, Rev. N03, Structural Reactor Building EL 931'-6" Plan & Schedules Drawing 4222, Rev. N02, Structural Reactor Building EL. 958'-3" Plan and Schedule Drawing 4223, Rev. N02, Structural Reactor Building EL. 976'-0" Plan & Schedule Drawing 4226, Rev. N01, Structural Miscellaneous Concrete Details Drawing 4306, Rev. N03, Structural Radwaste Building Plan at Elev.

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NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
				903'-6"
			Complies with use of EEEEs Structural drawings for the Turbine Building, Control Building, Reactor Building, Diesel Generator Building, Multi-Purpose Facility, Office Building, Radwaste Building and Augmented Radwaste Building indicate that the walls separating adjacent buildings are constructed of poured concrete with 3-hour fire rating. Doors within these barriers that do not meet the 3-hour fire rating have been evaluated as adequate in Engineering Evaluation EE 09-047. Engineering Evaluation EE 09-035 evaluates the adequacy of the metal plate covered door located less than 50 feet from the spare Main Transformer. The Fire Pump House is located less than 50 feet from the Multi-Purpose Facility. The exposed faces of the Fire Pump House and Multi-Purpose Facility are solid poured concrete construction, with only door opening in each exposed building wall. The distance between the two buildings is 45 feet, and based on the reductions in separation distance allowed for in Chapter 4 for automatic suppression installed in the Fire Pump House, this configuration meets the requirements of NFPA 80A-1996.	Drawing 4307, Rev. N04, Structural Radwaste Building Plan at Elev. 918'-0" Drawing 4308, Rev. N03, Structural Radwaste Building Plan at Elev. 934'-0" Drawing 4342, Rev. N03, Structural Office Building Plan - Elev. 918'-0" Drawing 4403, Rev. 11, Structural Augmented Radwaste Building Basement Plan Drawing 4406, Rev. N02, Structural Augmented Radwaste Building Plans at Elev. 903'-6" Drawing 4408, Rev. N01, Structural Augmented Radwaste Building Floor El. 918'-6" Plan, Sect. & Dets. EE 09-035, Rev. 1, Evaluation of Fire Doors EE 09-047, Rev. 0, Doors Required for NFPA 805 Building Separation NFPA 80A-1996, Recommended Practice for Protection of Buildings from Exterior Fire Exposures

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
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 Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	Complies Per Section 5.1.7(6) of the Fire Hazards Analysis (FHA), fire barrier ratings are assigned based on the construction features associated with the barrier. In addition, the individual FHA Fire Zone Matrix Tables, which are located in the FHA, provide a detailed fire barrier listing for the applicable fire zone. This matrix identifies fire barrier orientation, adjacent zone, applicable CNS-FP drawing, construction features, fire rating and associated barrier type. The referenced structural drawings for required fire barriers indicate that the barriers are typically constructed of poured concrete with 3-hour fire rating. Complies with use of EEEEs Fire barrier features within these barriers that do not meet the 3-hour fire rating have been determined to be equivalent in the referenced evaluations.	CNS Fire Hazards Analysis, Rev. 9/26/2011 Drawing 4058, Rev. N02, Structural Turbine Generator Building Foundation Walls and Sections Drawing 4059, Rev. 13, Structural Turbine Generator Building Concrete Walls Sht. No.1 Drawing 4123, Rev. N03, Structural Diesel Generator Building Plan at Elev. 903'-6" & 917'-6" Drawing 4173, Rev. N01, Structural Control Building Cable Room - Plans & Sections Drawing 4174, Rev. N05, Structural Control Building Basement Plan & Sections Drawing 4175, Rev. N02, Structural Control Building Operating Floor - Plan and Sections Drawing 4180, Rev. N05, Structural Control Building Corridor Sh 1 Drawing 4219, Rev. N05, Structural Reactor Bldg. EL. 903'-6" Plan, Sects., & Dets. SH #1 Drawing 4221, Rev. N03, Structural Reactor Building EL 931'-6" Plan & Schedules Drawing 4222, Rev. N02, Structural Reactor Building EL. 958'-3" Plan and Schedule

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
				Drawing 4223, Rev. N02, Structural Reactor Building EL. 976'-0" Plan & Schedule Drawing 4226, Rev. N01, Structural Miscellaneous Concrete Details Drawing 4306, Rev. N03, Structural Radwaste Building Plan at Elev. 903'-6" Drawing 4307, Rev. N04, Structural Radwaste Building Plan at Elev. 918'-0" Drawing 4308, Rev. N03, Structural Radwaste Building Plan at Elev. 934'-0" Drawing 4342, Rev. N03, Structural Office Building Plan - Elev. 918'-0" Drawing 4403, Rev. 11, Structural Augmented Radwaste Building Basement Plan Drawing 4406, Rev. N02, Structural Augmented Radwaste Building Plans at Elev. 903'-6" Drawing 4408, Rev. N01, Structural Augmented Radwaste Building Floor El. 918'-6" Plan, Sect. & Dets. EE 05-034, Rev. 0, Evaluation of the Reclassification of Door R104 Under the FHA EE 09-031, Rev. 0, Evaluation of Critical Switchgear Rooms 1F and 1G Fire Barrier Separation

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
				EE 09-036, Rev. 0, Evaluation of Cable Expansion Room Penetration Seals EE 09-040, Rev. 0, Evaluation of Auxiliary Relay Room and RPS Room 1B Appendix R Fire Barriers EE 09-042, Rev. 0, Evaluation of 1-Hour Marinite Wall in Battery Room 1B EE 12-013, Rev. 0, Evaluation of the SLC Pump Tank and Accessway (Fire Zone 5A) and Refueling Floor (Fire Zone 6) Fire Barrier Separation EE 86-2, Rev. 1, Evaluation of a Ventilation Opening Through the Cable Spreading Room Floor Appendix R Fire Barrier EE 86-5, Rev. 1, Evaluation of HVAC Ducts and Fire Door Between the Control Room and Controlled Corridor EE 97-121, Rev. 1, Appendix R Fire Protection Evaluation of Control Building 882' Underground Cable Manholes EE 97-124, Rev. 1, Evaluation of Steam Tunnel East Wall Fire Barrier LBDCR 2004-023, Rev. 0, Evaluation of a FHA Revision to Relocate the Fire Barrier Between Fire Area IV/Fire Zone 8D and Fire Area VII/Fire Zone 24
3.11.3 Fire Barrier Penetrations	Penetrations in fire barriers shall be provided with listed fire-rated door	Complies with use of Existing Engineering	<u>Complies with use of EEEEs</u> In some cases the doors and	CNS Fire Hazards Analysis, Rev. 9/26/2011

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	assemblies or listed rated fire dampers having a fire resistance rating consistent with the designated fire resistance rating of the barrier as determined by the performance requirements established by Chapter 4. (See 3.11.3.4 for penetration seals for through penetration fire stops.) *Exception: Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, a performance-based analysis shall be required to assess the adequacy of fire barrier forming the fire boundary to determine if the barrier will withstand the fire effects of the hazards in the area. Openings in fire barriers shall be permitted to be protected by other means as acceptable to the AHJ.*	Equivalency Evaluations (EEEEs) Complies	dampers do not have the same rating. If the fire barrier could not be assigned a rating or could not achieve a rating of approximately 50% greater than the equivalent fire severity to which it could be exposed, then engineering evaluations were developed to justify the adequacy of the barrier configuration. Fire barriers which required these evaluations are annotated in the FHA Fire Zone Matrix Table fire barrier listings. Doors and fire dampers within required fire barriers that do not meet the 3-hour fire rating have been evaluated as equivalent in the engineering evaluations referenced in Sections 3.11.3(1) and 3.11.3(2). Complies Fire barrier openings (doors and dampers) typically have the same fire resistance as the fire barrier in which they are installed.	
3.11.3 [Fire Barrier Penetrations - NFPA 80] (1)	Passive fire protection devices such as doors and dampers shall conform with the following NFPA standards, as applicable: NFPA 80, "Standard for Fire Doors and Fire Windows"	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	A detailed review of CNS fire doors has been performed against the requirements of NFPA 80, as detailed in the NFPA 80-1975 code review checklist in Engineering Evaluation EE 10-071. Doors within required fire barriers that do not meet the 3-hour fire rating have been	EE 05-034, Rev. 0, Evaluation of the Reclassification of Door R104 Under the FHA EE 09-031, Rev. 0, Evaluation of Critical Switchgear Rooms 1F and 1G Fire Barrier Separation EE 09-032, Rev. 0, Evaluation of DC SWGR Rooms 1A and 1B and Battery Rooms 1A and 1B Fire Barrier

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
			evaluated as equivalent in the referenced engineering evaluations.	Separation EE 09-035, Rev. 1, Evaluation of Fire Doors EE 09-040, Rev. 0, Evaluation of Auxiliary Relay Room and RPS Room 1B Appendix R Fire Barriers EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review EE 86-5, Rev. 1, Evaluation of HVAC Ducts and Fire Door Between the Control Room and Controlled Corridor
3.11.3 [Fire Barrier Penetrations - NFPA 90A] (2)	Passive fire protection devices such as doors and dampers shall conform with the following NFPA standards, as applicable: NFPA 90A, "Standard for the Installation of Air-Conditioning and Ventilating Systems"	Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	A detailed review of CNS fire dampers has been performed against the requirements of NFPA 90A, as detailed in the NFPA 90A-1985 code review checklist in Engineering Evaluation EE 10-071. Fire dampers within required fire barriers that do not meet the 3-hour fire rating have been evaluated as equivalent in Engineering Evaluation EE 09-031 and EE 09-032.	EE 09-031, Rev. 0, Evaluation of Critical Switchgear Rooms 1F and 1G Fire Barrier Separation EE 09-032, Rev. 0, Evaluation of DC SWGR Rooms 1A and 1B and Battery Rooms 1A and 1B Fire Barrier Separation EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review
3.11.3 [Fire Barrier Penetrations - NFPA 101] (3)	Passive fire protection devices such as doors and dampers shall conform with the following NFPA standards, as applicable: NFPA 101, "Life Safety Code"	Complies	As described in Engineering Evaluation EE 10-071, the requirements of NFPA 101 applicable to fire doors and fire dampers are bound by NFPA 80 and NFPA 90A. NFPA 101 Section 8.2.3.2.1 refers to NFPA 80, and NFPA 101 Section 9.2.1 refers to NFPA 90A.	EE 10-071, Rev. 0, CNS Acceptance of EPM Report No: R1906-002-002, NFPA Code Conformance Review
3.11.4 Through Penetration Fire Stops	Through penetration fire stops for penetrations such as pipes,	N/A	Introductory Statement - See compliance bases below for	None

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	conduits, bus ducts, cables, wires, pneumatic tubes and ducts, and similar building service equipment that pass through fire barriers shall be protected as follows.		compliance statements for specific subsections.	
3.11.4 [Through Penetration Fire Stops] (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.	Complies Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	<u>Complies</u> Per Section 2.1 of Procedure 3.6.1, openings in or penetrations of required fire barriers shall be sealed with an approved seal design with a fire rating commensurate with the rating of the fire barrier. The CNS-FP-285 Fire Protection Plan drawings depict approved seal design details. Per Section 2.3 of Procedure 3.6.1, all objects penetrating fire barriers (penetrants) shall be externally sealed using an approved fire rated seal design. Various designs of external seals are provided on the CNS-FP-285 Fire Protection Plan drawings. Calculation NEDC 91-3, Qualification of Fire Barrier Penetration Seal Details, provides technical justification for the penetration seal details utilized at CNS. <u>Complies with use of EEEEs</u> Penetrations within required fire barriers that do not meet the 3-hour fire rating have been evaluated as equivalent in the referenced engineering evaluations.	EE 09-031, Evaluation of Critical Switchgear Rooms 1F and 1G Fire Barrier Separation, Rev. 0 EE 09-032, Evaluation of DC SWGR Rooms 1A and 1B and Battery Rooms 1A and 1B Fire Barrier Separation, Rev. 0 EE 09-036, Evaluation of Cable Expansion Room Penetration Seals, Rev. 0 EE 97-124, Evaluation of Steam Tunnel East Wall Fire Barrier, Rev. 1 Fire Protection Plan CNS-FP-285, Sht. 1, Rev. N06, CNS Fire Barrier Penetration Seal Details Fire Protection Plan CNS-FP-285, Sht. 2, Rev. N04, CNS Fire Barrier Penetration Seal Details Fire Protection Plan CNS-FP-285, Sht. 3, Rev. N06, CNS Fire Barrier Penetration Details NEDC 91-3, Qualification of Fire Barrier Penetration Seal Details Procedure 3.6.1, Rev. 20, Fire Barrier Control

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
3.11.4 [Through Penetration Fire Stops] (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 conduit 4 in. (10.2 cm) or less in diameter shall be sealed at the fire barrier with a fire-rated internal seal unless the conduit extends greater than 5 ft (1.5 m) on each side of the fire barrier. In this case the conduit opening shall be provided with noncombustible material to prevent the passage of smoke and hot gases. The fill depth of the material packed to a depth of 2 in. (5.1 cm) shall constitute an acceptable smoke and hot gas seal in this application.*	Complies Complies with use of Existing Engineering Equivalency Evaluations (EEEEs)	<u>Complies</u> Section 2.4.1.3 of Procedure 3.6.1 provides the requirements for fire rated, smoke, and hot gas internal seals. <u>Complies with use of EEEEs</u> EE 10-055, Fire Protection Evaluation of Internal Conduit Seals, provides a technical evaluation of the internal conduit seal program, specifies the specific criteria for providing internal seals for conduits penetrating fire barriers, and evaluates the impact of these criteria on existing internal conduit seals.	EE 10-055, Rev. 0, Fire Protection Evaluation of Internal Conduit Seals Procedure 3.6.1, Rev. 20, Fire Barrier Control
3.11.5 Electrical Raceway Fire Barrier Systems (ERFBS)	ERFBS required by Chapter 4 shall be capable of resisting the fire effects of the hazards in the area. ERFBS shall be tested in accordance with and shall meet the acceptance criteria of NRC Generic Letter 86-10, Supplement 1, "Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Safe Shutdown Trains Within the Same Fire Area." The ERFBS needs to adequately address the design requirements and limitations of supports and intervening items and their impact on the fire barrier system rating. The fire barrier system's ability to	N/A	There are no ERFBS installed at CNS.	None

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

NFPA 805 Element	NFPA 805 Requirement	Compliance Statement	Compliance Basis	Reference Document
	maintain the required nuclear safety circuits free of fire damage for a specific thermal exposure, barrier design, raceway size and type, cable size, fill, and type shall be demonstrated. *Exception No. 1: When the temperatures inside the fire barrier system exceed the maximum temperature allowed by the acceptance criteria of Generic Letter 86-10, "Fire Endurance Acceptance Test Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Training Within the Same Fire Area," Supplement 1, functionality of the cable at these elevated temperatures shall be demonstrated. Qualification demonstration of these cables shall be performed in accordance with the electrical testing requirements of Generic Letter 86-10, Supplement 1, Attachment 1, "Attachment Methods for Demonstrating Functionality of Cables Protected by Raceway Fire Barrier Systems During and After Fire Endurance Test Exposure."* *Exception No. 2: ERFBS systems employed prior to the issuance of Generic Letter 86-10, Supplement 1, are acceptable providing that the system successfully met the limiting end point temperature requirements as specified by the AHJ at the time of acceptance.*			

ATTACHMENT B

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

109 Pages

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

NFPA 805 Section: 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 maloperation of those components needed to meet the nuclear safety criteria shall be included. Availability and reliability of equipment selected shall be evaluated.

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

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

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

The results of this review conclude that the CNS NSCA has been performed consistent with (i.e., aligns with) the deterministic methodology guidance, criteria, and

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

assumptions from Chapter 3 of NEI 00-01 except as noted within this document.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance

3.1 Safe Shutdown Systems and Path Development	This section discusses the identification of systems available and necessary to perform the required safe shutdown functions. It also provides information on the process for combining these systems into safe shutdown paths. Appendix R Section III.G.1.a requires that the capability to achieve and maintain hot shutdown be free of fire damage. It is expected that the term "free of fire damage" will be further clarified in a forthcoming Regulatory Issue Summary. Appendix R Section III.G.1.b requires that repairs to systems and equipment necessary to achieve and maintain cold shutdown be completed within 72 hours. It is the intent of the NRC that requirements related to the use of manual operator actions will be addressed in a forthcoming rulemaking.
	
	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 important to achieve and maintain hot shutdown. Safe shutdown systems are selected so that the capability to perform these required functions is a part of each safe shutdown path. The functions important to post-fire safe shutdown generally include, but are not limited to the following:
	
	- Reactivity Control
	- Pressure Control Systems
	- Inventory Control Systems
	- Decay Heat Removal Systems
	- Process Monitoring
	- Support Systems
	    * Electrical systems
	    * Cooling systems
	
	These functions are of importance because they have a direct bearing on the safe shutdown goal of being able to achieve and maintain hot shutdown which ensures the integrity of the fuel, the reactor pressure vessel, and the primary containment. If these functions are preserved, then the plant will be safe because the fuel, the reactor and the primary containment will not be damaged. By assuring that this equipment is not damaged and remains functional, the protection of the health and safety of the public is assured.
	
	In addition to the above listed functions, Generic Letter 81-12 specifies consideration of associated circuits with the potential for spurious equipment operation and/or loss of power source, and the common enclosure failures. Spurious operations/actuations can affect the accomplishment of the post-fire safe shutdown functions listed above. Typical examples of the effects of the spurious operations of concern are the following:
	
	- A loss of reactor pressure vessel/reactor coolant inventory in excess of the safe shutdown makeup capability

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

- A flow loss or blockage in the inventory makeup or decay heat removal systems being used for the required safe shutdown path.

Spurious operations are of concern because they have the potential to directly affect the ability to achieve and maintain hot shutdown, which could affect the fuel and cause damage to the reactor pressure vessel or the primary containment. Common power source and common enclosure concerns could also affect these and must be addressed.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

CNS systems / functions / components required to achieve and maintain "safe and stable" plant conditions post-fire per the Nuclear Safety Performance Criteria of NFPA 805 are identified in CNS Calculation NEDC 11-019, Section 9.0, Nuclear Safety Capability Assessment Analysis Model Development.

NFPA 805 allows more flexibility than the previous deterministic programs based on 10 CFR 50 Appendix R (and NEI 00-01, Chapter 3) since NFPA 805 only requires the licensee to maintain the fuel in a safe and stable condition. NFPA 805, Section 1.6.56 defines safe and stable condition. For CNS to be in a safe and stable condition, it will not be necessary to perform a transition to cold shutdown as currently required under 10 CFR 50, Appendix R.

The identification and analysis of these systems / functions / components includes addressing associated circuit issues for spurious operations, high/low pressure interfaces, common power supplies, and common enclosures.

A computer database tool, EDISON/SAFE, is utilized to demonstrate that the Nuclear Safety Performance Criteria of NFPA 805 are met for each fire area of the plant.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 7.5, 9.0, 10.0 and 11.0)

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

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

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.1.1	Criteria/Assumptions	The following criteria and assumptions may be considered when identifying systems available and necessary to perform the required safe shutdown functions and combining these systems into safe shutdown paths.

Applicability	Comments
Not Applicable	None

Alignment Statement

Not Applicable

Alignment Basis

Introductory Section. Refer to following sub-sections for detailed guidance and bases.

Reference Documents

None

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

NEI 00-01 Ref

3.1.1.1 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

[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

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 9.0, Nuclear Safety Capability Assessment Analysis Model Development, identifies the performance criteria, systems, equipment and logic development in order to achieve a safe and stable condition. Calculation NEDC 11-019, Section 12.0 describes the analysis results with the individual fire area assessments contained within Appendix F of Calculation NEDC 11-019. The shutdown paths and methods described in these sections of this calculation along with those depicted in the analysis model contained in EDISON/SAFE are consistent with the information provided in the referenced GE Report.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 9.0, 12.0 and Appendix F)

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.2 Criteria/Assumptions	[BWR] GE Report GE-NE-T43-00002-00-03-R01 provides a discussion on the BWR Owners' Group (BWROG) position regarding the use of Safety Relief Valves (SRVs) and low pressure systems (LPCI/CS) for safe shutdown. The BWROG position is that the use of SRVs and low pressure systems is an acceptable methodology for achieving redundant safe shutdown in accordance with the requirements of 10CFR50 Appendix R Sections III.G.1 and III.G.2. The NRC has accepted the BWROG position and issued an SER dated Dec. 12, 2000.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

ADS is used to reduce plant pressure for the use of the low pressure CS system to achieve a safe and stable condition. Calculation NEDC 11-019 details the CS, LPCI and ADS systems. The safety relief valves within ADS are credited to rapidly depressurize the reactor to allow for the use of the low pressure systems in lieu of the high pressure systems for reactor makeup. LPCI is modeled within the analysis but is not the preferred method of low pressure makeup.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 9.1.2, 9.4.2.3, 9.4.2.4 and 9.4.2.7)

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

NEI 00-01 Ref

3.1.1.3 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

[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 cool down rate must be controlled to prevent the formation of a bubble in the reactor head. Therefore, feedwater (either auxiliary or emergency) flow rates as well as steam release must be controlled.

Applicability

Not Applicable

Comments

None

Alignment Statement

Not Applicable

Alignment Basis

CNS is a BWR.

Reference Documents

None

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.4 Criteria/Assumptions	The classification of shutdown capability as alternative shutdown is made independent of the selection of systems used for shutdown. Alternative shutdown capability is determined based on an inability to assure the availability of a redundant safe shutdown path. Compliance to the separation requirements of Sections III.G.1 and III.G.2 may be supplemented by the use of manual actions to the extent allowed by the regulations and the licensing basis of the plant, repairs (cold shutdown only), exemptions, deviations, GL 86-10 fire hazards analyses or fire protection design change evaluations, as appropriate. These may also be used in conjunction with alternative shutdown capability.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Alternate shutdown capability is provided at CNS as a redundant path exterior to the Main Control Room. Alternative shutdown capability is comprised of 3 control panels in the Alternate Shutdown Room. These control panels provide isolation, control and indication for ADS, HPCI, RHR and REC. CNS Operations Emergency Procedure 5.4FIRE-S/D, “Fire Induced Shutdown From Outside the Control Room,” governs the use of the Alternate Shutdown Panels.

Unlike 10 CFR 50 Appendix R, NFPA 805 makes no distinction for alternate and dedicated shutdown. Alternate Shutdown Panels are the primary control stations for implementation of the 10 CFR 50 Appendix R alternate shutdown strategy in the event of a fire that requires the evacuation of the Main Control Room. Based on the definition provided in RG 1.205, and the additional guidance provided in FAQ 07-0030 Revision 5 (ML110070485), Alternate Shutdown Panels are also considered to be the Primary Control Station for NFPA 805.

The use of Alternate Shutdown Panels have been transitioned to NFPA 805 as the Primary Control Stations for meeting the NSPC in the event of a fire that requires evacuation of the Main Control Room.

Reference Documents

CNS Operations Emergency Procedure 5.4 FIRE-S/D, "Fire Induced Shutdown From Outside the Control Room," Revision 44

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.5 Criteria/Assumptions	At the onset of the postulated fire, all safe shutdown systems (including applicable redundant trains) are assumed operable and available for post-fire safe shutdown. Systems are assumed to be operational with no repairs, maintenance, testing, Limiting Conditions for Operation, etc. in progress. The units are assumed to be operating at full power under normal conditions and normal lineups.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criteria / assumption listed in Section 3.1.1.5 of NEI 00-01 is explicitly stated in Section 5.1.1 of Calculation NEDC 11-019.

All systems credited to achieve a safe and stable condition are assumed to be operational at the onset of a postulated fire with the unit operating at full power.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.1.1)

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.6 Criteria/Assumptions	No Final Safety Analysis Report accidents or other design basis events (e.g. loss of coolant accident, earthquake), single failures or non-fire induced transients need be considered in conjunction with the fire.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criteria / assumption listed in Section 3.1.1.6 of NEI 00-01 is explicitly stated in Section 5.1.1 of Calculation NEDC 11-019.

No additional design basis accidents need to be considered in conjunction with the postulated fire analysis.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.1.1)

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

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

NEI 00-01 Ref

3.1.1.7 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

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

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criteria / assumption listed in Section 3.1.1.7 of NEI 00-01 is explicitly stated in Section 5.1.1 of Calculation NEDC 11-019.

Offsite power can be used as a source of power for the NSCA, if desired. All equipment required to support the portion of offsite power relied upon to achieve the NSPC has been included in the NSCA model.

The electrical distribution systems in the CNS NSCA, inclusive of offsite power capability, is addressed in Section 9.1 of this calculation and is intertwined at the component level and not modeled as a stand-alone system.

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

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 5.1.1 and 5.2.1)

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

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

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Both safety-related and non-safety related equipment are credited to achieve a safe and stable condition. Calculation NEDC 11-019 does not restrict the use of only safety-related systems and components to achieve the nuclear safety performance criteria.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.1.1)

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

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

NEI 00-01 Ref

3.1.1.9 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

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

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

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

The nuclear safety goals, objectives and performance criteria of NFPA 805, as discussed above, allow more flexibility than the previous deterministic programs based on 10 CFR 50 Appendix R (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 <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 <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 CNS to be in a safe and stable condition, it will not be necessary to perform a transition to cold shutdown as currently required under 10 CFR 50, Appendix R. Therefore, the unit will remain at or below the temperature defined by a hot shutdown plant operating state for the event.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 7.5)

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.1.10 Criteria/Assumptions	Manual initiation from the main control room or emergency control stations of systems required to achieve and maintain safe shutdown is acceptable where permitted by current regulations or approved by NRC; automatic initiation of systems selected for safe shutdown is not required but may be included as an option.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criterion / assumption listed in Section 3.1.1.10 of NEI 00-01 is explicitly stated in Section 5.1.1 of Calculation NEDC 11-019.

The CNS NSCA credits Main Control Room operator actions to align NSCA systems / functions / components. The CNS NSCA does not credit automatic initiation of NSCA systems / functions / components unless specifically modeled and analyzed.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.1.1)

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

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

NEI 00-01 Ref

3.1.1.11 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

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

Not Applicable

Comments

None

Alignment Statement

Not Applicable

Alignment Basis

CNS is a single unit facility.

Reference Documents

None

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2 Shutdown Functions	The following discussion on each of these shutdown functions provides guidance for selecting the systems and equipment required for safe 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
Not Applicable	None

Alignment Statement

Not Applicable

Alignment Basis

Introductory Section. Refer to following sub-sections for detailed guidance and bases.

Reference Documents

None

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

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

NEI 00-01 Ref

3.1.2.1 Reactivity Control

NEI 00-01 Section 3 Guidance

[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.

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Reactivity control will be accomplished by insertion of the control rods and will result from an automatic RPS trip or from operator initiation of a manual trip. This action will de-energize the RPS to actuate a reactor scram. The CNS NSCA does not credit automatic initiation of NSCA systems / functions / components unless specifically modeled and analyzed.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 5.1.1 and 9.1.1)

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

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

NEI 00-01 Ref

3.1.2.1 Reactivity Control

NEI 00-01 Section 3 Guidance

[PWR] Makeup/Charging

There must be a method for ensuring that adequate shutdown margin is maintained by ensuring borated water is utilized for RCS makeup/charging.

Applicability

Not Applicable

Comments

None

Alignment Statement

Not Applicable

Alignment Basis

CNS is a BWR.

Reference Documents

None

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.2 Pressure Control Systems	[BWR] Safety Relief Valves (SRVs) The SRVs are opened to maintain hot shutdown conditions or to depressurize the vessel to allow injection using low pressure systems. These are operated manually. Automatic initiation of the Automatic Depressurization System is not a required function.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Overpressure protection is provided by the SRVs in the self-activated spring lift mode. This mode of operation is not susceptible to fire damage. The SRVs are also actuated by the operator or ADS to reduce plant pressure for the use of low pressure CS system. The ADS system is only credited in manual mode for depressurization. Use of the ADS Inhibit switch blocks spurious ADS auto-initiation signal.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 9.1.2 and 9.4.2.7.3)

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

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

NEI 00-01 Ref

3.1.2.2 Pressure Control Systems

NEI 00-01 Section 3 Guidance

[PWR] Makeup/Charging

RCS pressure is controlled by controlling the rate of charging/makeup to the RCS. Although utilization of the pressurizer heaters and/or auxiliary spray reduces operator burden, neither component is required to provide adequate pressure control. Pressure reductions are made by allowing the RCS to cool/shrink, thus reducing pressurizer level/pressure. Pressure increases are made by initiating charging/makeup to maintain pressurizer level/pressure. Manual control of the related pumps is acceptable.

Applicability

Not Applicable

Comments

None

Alignment Statement

Not Applicable

Alignment Basis

CNS is a BWR.

Reference Documents

None

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

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

NEI 00-01 Ref

3.1.2.3 Inventory Control

NEI 00-01 Section 3 Guidance

[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.

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Reactor coolant make-up can be achieved by isolation of the Reactor Coolant System and control of the vessel coolant level by injecting water into the isolated reactor vessel. The RCS pressure boundary is necessary to achieve inventory and pressure control. The required components to limit coolant loss are contained within the analysis. Reactor make-up is provided by the operation of the CS system, the HPCI system or the RCIC system. Note that depressurization is required to reduce plant pressure for the use of the low pressure CS system. Only manual operation of the HPCI, RCIC and CS systems are credited to achieve a safe and stable condition.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 9.1.2, 9.4.2.1.1, 9.4.2.2.1 and 9.4.2.3.1)

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

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

NEI 00-01 Ref

3.1.2.3 Inventory Control

NEI 00-01 Section 3 Guidance

[PWR] Systems selected for the inventory control function should be capable of maintaining level to achieve and maintain hot shutdown. Typically, the same components providing inventory control are capable of providing pressure control. Manual initiation of these systems is acceptable. Automatic initiation functions are not required.

Applicability

Not Applicable

Comments

None

Alignment Statement

Not Applicable

Alignment Basis

CNS is a BWR.

Reference Documents

None

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.2.4 Decay Heat Removal	[BWR] Systems selected for the decay heat removal function(s) should be capable of: - Removing sufficient decay heat from primary containment, to prevent containment over-pressurization and failure. - Satisfying the net positive suction head requirements of any safe shutdown systems taking suction from the containment (suppression pool). - Removing sufficient decay heat from the reactor to achieve cold shutdown. This does not restrict the use of other systems.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Decay Heat is removed initially by natural circulation within the reactor pressure vessel and automatic mechanical operation of the SRVs. The SRVs discharge steam from the reactor vessel to the suppression pool. The emerging steam is condensed in this pool, and the heat absorbed by the suppression pool is removed by the RHR System operating in the Suppression Pool Cooling (SPC) mode and ultimately transferred to the river via the Service Water (SW) System. The required system logics for suppression pool cooling include satisfying net positive suction head requirements.

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

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 9.1.3)

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

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

NEI 00-01 Ref

3.1.2.4 Decay Heat Removal

NEI 00-01 Section 3 Guidance

[PWR] Systems selected for the decay heat removal function(s)) should be capable of:

- Removing sufficient decay heat from the reactor to reach hot shutdown conditions. Typically, this entails utilizing natural circulation in lieu of forced circulation via the reactor coolant pumps and controlling steam release via the Atmospheric Dump valves.

- Removing sufficient decay heat from the reactor to reach cold shutdown conditions.

This does not restrict the use of other systems.

Applicability

Not Applicable

Comments

None

Alignment Statement

Not Applicable

Alignment Basis

CNS is a BWR.

Reference Documents

None

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

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

NEI 00-01 Ref

3.1.2.5 Process Monitoring

NEI 00-01 Section 3 Guidance

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

BWR

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

PWR

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

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

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

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

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

Alignment Basis

The following instruments are modeled in the Nuclear Boiler Instrumentation (NBI) system, based on existing operating procedures:

Reactor coolant level and pressure
Suppression chamber temperature and level

The indicating ranges of these instruments cover the normal operating bands and will operate throughout the scenario. Other tank levels and diagnostic instrumentation such as flow or system pressures are modeled directly in the logics for the system or component that requires the indication and are not included in this performance goal.

Other diagnostic instrumentation contained within the model includes the follow:

HPCI Pump Discharge Pressure Indication
HPCI Turbine Inlet and Outlet Pressure Indication
HPCI Turbine Speed Indication
ECST Level Indication
RCIC Turbine Speed Indication
CS Pump A and B Discharge Flow Indication
RHR System Loop A and B Flow Indication

When shutting down from outside the Main Control Room, the following instruments are available at the Primary Control Station in the Alternate Shutdown Room:

RPV Level
Reactor Pressure (at HPCI Turbine Steam Inlet)
Suppression Chamber Temperature and Level
ECST Level

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 9.1.5)

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

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

NEI 00-01 Ref

3.1.2.6.1 Electrical Systems

NEI 00-01 Section 3 Guidance

AC Distribution System

Power for the Appendix R safe shutdown equipment is typically provided by a medium voltage system such as 4.16 KV Class 1E busses either directly from the busses or through step down transformers/load centers/distribution panels for 600, 480 or 120 VAC loads. For redundant safe shutdown performed in accordance with the requirements of Appendix R Section III.G.1 and 2, power may be supplied from either offsite power sources or the emergency diesel generator depending on which has been demonstrated to be free of fire damage. No credit should be taken for a fire causing a loss of offsite power. Refer to Section 3.1.1.7.

DC Distribution System

Typically, the 125VDC distribution system supplies DC control power to various 125VDC control panels including switchgear breaker controls. The 125VDC distribution panels may also supply power to the 120VAC distribution panels via static inverters. These distribution panels typically supply power for instrumentation necessary to complete the process monitoring functions. For fire events that result in an interruption of power to the AC electrical bus, the station batteries are necessary to supply any required control power during the interim time period required for the diesel generators to become operational. Once the diesels are operational, the 125 VDC distribution system can be powered from the diesels through the battery chargers.

[BWR] Certain plants are also designed with a 250VDC Distribution System that supplies power to Reactor Core Isolation Cooling and/or High Pressure Coolant Injection equipment.

The DC control centers may also supply power to various small horsepower Appendix R safe shutdown system valves and pumps. If the DC system is relied upon to support safe shutdown without battery chargers being available, it must be verified that sufficient battery capacity exists to support the necessary loads for sufficient time (either until power is restored, or the loads are no longer required to operate).

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

The power distribution system is contained within the NSCA model as equipment-to-equipment logics. 4.16 KV power is provided from offsite sources or diesel generators. This power is stepped down through transformers for the necessary VAC levels. 250V and 125V DC power is provided with batteries that are supported with battery chargers.

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

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Appendix C)

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

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

NEI 00-01 Ref

3.1.2.6.2 Cooling Systems

NEI 00-01 Section 3 Guidance

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

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

HVAC Systems may be required to assure that safe shutdown equipment remains within its operating temperature range, as specified in manufacturer's literature or demonstrated by suitable test methods, and to assure protection for plant operations staff from the effects of fire (smoke, heat, toxic gases, and gaseous fire suppression agents).

HVAC systems may be required to support safe shutdown system operation, based on plant-specific configurations. Typical uses include:

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

Plant-specific evaluations are necessary to determine which HVAC systems are essential to safe shutdown equipment operation.

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Various plant systems are required to support the systems and components selected to accomplish the previously defined safety functions.

The following are modeled as Systems in the SUPPORT Performance Goal:

Reactor Equipment Cooling (REC): This system supplies cooling to critical heat exchangers and coolers. If both pumps are inoperable, emergency operation is

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

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

possible by cross-connecting to the Service Water system. This allows modeling Train A and Train B of both the normal and the emergency lineup as four different systems and at least one of the four are required to survive.

Service Water: This system cools the diesel generator coolers and the REC and RHR heat exchangers. EDISON/SAFE models Train A and Train B as two systems, and requires the same-train Service Water to cool the REC and RHR systems. Both diesel coolers can be supplied from either Train A and Train B Service Water. Components of the Service Water system that are specific to individual cooling loads are modeled as support components for the cooled system.

Other critical support functions are modeled in as equipment logics:

HVAC Systems. HVAC cooling is required and modeled for the battery rooms, diesel generators, critical switchgear rooms, Core Spray Pump Rooms (including the RCIC turbine) and the HPCI Room. Plant evaluations have confirmed that other areas of the plant do not require HVAC cooling in order to protect credited equipment from overheating or to ensure habitability.

Diesel Generator Support Auxiliaries. Provides direct support in equipment logics for the corresponding Diesel Generator. This includes cooling water valves, ventilation, starting air, fuel oil transfer pumps and jacket water pumps.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 9.1.4, 9.4.4.5 and 9.4.4.6)

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

NEI 00-01 Ref

3.1.3 Methodology for Shutdown System Selection

NEI 00-01 Section 3 Guidance

Refer to hardcopy of NEI 00-01 Rev 1 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

Not Applicable

Comments

None

Alignment Statement

Not Applicable

Alignment Basis

Introductory section. Refer to following subsection detailed guidance and bases.

Reference Documents

None

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.3.1 Identify Safe Shutdown Functions	Review available documentation to obtain an understanding of the available plant systems and the functions required to achieve and maintain safe shutdown. Documents such as the following may be reviewed: - Operating Procedures (Normal, Emergency, Abnormal) - System descriptions - Fire Hazard Analysis - Single-line electrical diagrams - Piping and Instrumentation Diagrams (P&IDs) - [BWR] GE Report GE-NE-T43-00002-00-01-R02 entitled "Original Shutdown Paths for the BWR"

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The necessary documentation was reviewed in support of the development of the NSCA model to achieve and maintain a safe and stable condition.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 6.0)

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.3.2 Identify Combinations of Systems That Satisfy Each Safe Shutdown Function	Given the criteria/assumptions defined in Section 3.1.1, identify the available combinations of systems capable of achieving the safe shutdown functions of reactivity control, pressure control, inventory control, decay heat removal, process monitoring and support systems such as electrical and cooling systems (refer to Section 3.1.2). This selection process does not restrict the use of other systems. In addition to achieving the required safe shutdown functions, consider spurious operations and power supply issues that could impact the required safe shutdown function.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The method to achieve a safe and stable condition and system path relationships used to analyze post-fire safe shutdown are maintained in EDISON/SAFE as described in Calculation NEDC 11-019, Section 11.0. Each safe shutdown function (methods) and credited system is included in logic diagrams. The supporting systems are illustrated as logical paths necessary to satisfy each function through AND/OR statements.

Components whose spurious operation could adversely affect safe shutdown functions are included in the system level logics as applicable. Associated Circuits by Common Power Supply have been addressed as described in Calculation NEDC 11-019, Section 10.3 as applicable. There are no coordination issues that required the inclusion of circuits to the analysis model.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 10.3 and 11.0)

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.3.3 Define Combination of Systems for Each Safe Shutdown Path	Select combinations of systems with the capability of performing all of the required safe shutdown functions and designate this set of systems as a safe shutdown path. In many cases, paths may be defined on a divisional basis since the availability of electrical power and other support systems must be demonstrated for each path. During the equipment selection phase, identify any additional support systems and list them for the appropriate path.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Different combinations of systems that successfully achieve the nuclear safety functions criteria are illustrated on logic diagrams. The NSCA Model relationships are maintained in the EDISON/SAFE database as described in Calculation NEDC 11-019, Section 11.0.

Additional supplementary information is provided for each of the credited systems in Section 9.1 of Calculation NEDC 11-019.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 9.1 and 11.0)

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.1.3.4 Assign Shutdown Paths to Each Combination of Systems	Assign a path designation to each combination of systems. The path will serve to document the combination of systems relied upon for safe shutdown in each fire area. Refer to Attachment 1 to this document for an example of a table illustrating how to document the various combinations of systems for selected shutdown paths.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Path designations are assigned to each specific system using Train A or B designation where applicable. Each path is a specific combination of systems and components supporting the success of a method, as illustrated in logic diagrams. The analysis of each fire area ensures that at least one success path exists for each fire area.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Appendix F)

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

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

NEI 00-01 Ref

3.2 Safe Shutdown Equipment Selection

NEI 00-01 Section 3 Guidance

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

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

All selected equipment in support of a system required to achieve a safe and stable condition is appropriately designated as such in Appendix A of Calculation NEDC 11-019. The equipment list relates back to the systems. The specific equipment that support each system path are documented within Calculation NEDC 11-019, Tables 9.4.2-1 through 9.4.4-8 and are illustrated in the logic diagrams within Appendix E of Calculation NEDC 11-019.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Appendix A, Appendix E and Tables 9.4.2-1 through 9.4.4-8)

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

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

NEI 00-01 Ref

3.2.1 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

Consider the following criteria and assumptions when identifying equipment necessary to perform the required safe shutdown functions:

Applicability

Not Applicable

Comments

None

Alignment Statement

Not Applicable

Alignment Basis

Introductory Section. Refer to following sub-sections for detailed guidance and bases.

Reference Documents

None

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

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

NEI 00-01 Ref

3.2.1.1 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

Safe shutdown equipment can be divided into two categories. Equipment may be categorized as (1) primary components or (2) secondary components. Typically, the following types of equipment are considered to be primary components:

- Pumps, motor operated valves, solenoid valves, fans, gas bottles, dampers, unit coolers, etc.

- All necessary process indicators and recorders (i.e., flow indicator, temperature indicator, turbine speed indicator, pressure indicator, level recorder)

- Power supplies or other electrical components that support operation of primary components (i.e., diesel generators, switchgear, motor control centers, load centers, power supplies, distribution panels, etc.).

Secondary components are typically items found within the circuitry for a primary component. These provide a supporting role to the overall circuit function. Some secondary components may provide an isolation function or a signal to a primary component via either an interlock or input signal processor. Examples of secondary components include flow switches, pressure switches, temperature switches, level switches, temperature elements, speed elements, transmitters, converters, controllers, transducers, signal conditioners, hand switches, relays, fuses and various instrumentation devices.

Determine which equipment should be included on the Safe Shutdown Equipment List (SSEL). As an option, include secondary components with a primary component(s) that would be affected by fire damage to the secondary component. By doing this, the SSEL can be kept to a manageable size and the equipment included on the SSEL can be readily related to required post-fire safe shutdown systems and functions.

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Appendix A of Calculation NEDC 11-019 contains the required equipment list. While the list does not specifically list equipment with primary or secondary designations, it does list the equipment as to what system it is a part of. The equipment list combined with the system and equipment logics implicitly show primary and secondary relationships and designations.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Appendix A)

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

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

NEI 00-01 Ref

3.2.1.2 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

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

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criterion / assumption listed in Section 3.2.1.2 of NEI 00-01 is explicitly stated in Section 5.1.2 of Calculation NEDC 11-019.

Fire damage is evaluated with respect to the ability to manually open or close the valve should it be necessary to do so as a part of the post-fire safe shutdown scenario.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 5.1.2 & 5.2.3)

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

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

NEI 00-01 Ref	**NEI 00-01 Section 3 Guidance**
3.2.1.3 Criteria/Assumptions	Assume that manual valves are in their normal position as shown on P&IDs or in the plant operating procedures.

Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criterion / assumption listed in Section 3.2.1.3 of NEI 00-01 is explicitly stated in Section 5.1.2 of Calculation NEDC 11-019.

Manual valves are assumed to be in their normal operating position for the analysis.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.1.2)

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

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

NEI 00-01 Ref	**NEI 00-01 Section 3 Guidance**
3.2.1.4 Criteria/Assumptions	Assume that a check valve closes in the direction of potential flow diversion and seats properly with sufficient leak tightness to prevent flow diversion. Therefore, check valves do not adversely affect the flow rate capability of the safe shutdown systems being used for inventory control, decay heat removal, equipment cooling or other related safe shutdown functions.

Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criterion / assumption listed in Section 3.2.1.4 of NEI 00-01 is explicitly stated in Section 5.1.2 of Calculation NEDC 11-019.

Check valves are assumed to operate properly for the analysis.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.1.2)

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.2.1.5 Criteria/Assumptions	Instruments (e.g., resistance temperature detectors, thermocouples, pressure transmitters, and flow transmitters) are assumed to fail upscale, midscale, or downscale as a result of fire damage, whichever is worse. An instrument performing a control function is assumed to provide an undesired signal to the control circuit.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criterion / assumption listed in Section 3.2.1.5 of NEI 00-01 is explicitly stated in Section 5.1.2 of Calculation NEDC 11-019.

Instruments 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.

In addition, the analysis includes instruments which provide permissive or controlling signals to safe shutdown components, or which can cause spurious operation. These instruments are modeled in direct support of the affected component using equipment logics, cable logics, or a combination of both.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 5.1.2 and 9.1.5)

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

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

NEI 00-01 Ref

3.2.1.6 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

Identify equipment that could spuriously operate or mal-operate and impact the performance of equipment on a required safe shutdown path during the equipment selection phase. Consider Bin 1 of RIS 2004-03 during the equipment identification process.

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Support components were identified where credit was taken for proper control and/or operation of other main or support components (e.g., solenoid valve supporting air-operated valve). In addition, components that could cause spurious actuation of another component, loss of control of another component, or another component adopting an unwanted position/status were analyzed for cable selection (e.g., transmitter supporting process control loop or relay supporting control circuit). Spurious components were divided into the following two categories:
a) Components that could cause spurious operations when energized or de-energized; and
b) Components that could transmit spurious signals.
The components of Category a) were analyzed for cable selection based on receipt of an unwanted signal resulting in spurious operation of the main component to an undesirable position. The components of Category b) were considered "permissive" and analyzed accordingly. For the purpose of spurious signal analysis, the selected cables included all cables that could transmit a spurious signal as a result of a fire-induced cable failure. This included conductors that could operate a relay having a contact(s) in the control circuit being analyzed.

RIS 2004-03, Bin 1 circuit configurations (i.e., those most likely to fail under fire scenarios) were considered. This included conductor-to-conductor shorts within a multi-conductor cable, and cable-to-cable interactions. The analysis did not limit the number of cables that may be damaged by fire.

During the NFPA 805 transition, NPPD performed a series of MSO Expert Panel systems reviews. Results were fed back into the NSCA and Fire PRA, as necessary. The treatment of MSOs is provided in the applicable section of the LAR Transition Report.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 10.2.1.2)

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.2.1.7 Criteria/Assumptions	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

As part of the NFPA 805 Transition, the impact of fire on instrument tubing sensing lines was evaluated considering the fire area location of the instrument sensing lines. The sensing lines for the applicable process monitoring instruments are included in the NSCA model, and evaluated similar to a cable, such that the instrument is assumed to fail in areas containing its associated tubing unless an evaluation notes otherwise. The sensing lines for applicable process monitoring instruments are welded steel therefore the pressure boundary will not be breached as a result of fire damage. For instruments that are credited as active or for instruments whose failure could be detrimental, the fire zone location of the instrument, together with the fire zone(s) where the instrument's sensing line is located, if different than that of the instrument, are associated in EDISON/SAFE with the instrument. In this manner, a fire would fail the instrument in the fire zone containing the instrument and in any fire zone containing the sensing line.

The effects of exposure fire damage on instrumentation tubing is not an issue as original construction requirements only allowed socket weld, butt weld, or screwed joints. CNS does not use brazed or soldered joints. Copper tubing may be used in limited pneumatic applications however it is not used in instrument lines.

Reference Documents

Contract E70-3, "Piping Tubing and Fittings for IC," Revision 2

GE Specification 22A1427AE, "Process Instrument Piping and Tubing," Revision 1

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 9.5)

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.2.2 Methodology for Equipment Selection	Refer to NEI 00-01 Rev 1 Figure 3-3 for a flowchart illustrating the various steps involved in selecting safe shutdown equipment. Use the following methodology to select the safe shutdown equipment for a post-fire safe shutdown analysis: [Refer to hardcopy of NEI 00-01 for Figure]

Applicability	Comments
Not Applicable	None

Alignment Statement

Not Applicable

Alignment Basis

Introductory section. Refer to following subsection detailed guidance and bases.

Reference Documents

None

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

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

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.2.2.1	Identify the System Flow Path for Each Shutdown Path	Mark up and annotate a P&ID to highlight the specific flow paths for each system in support of each shutdown path. Refer to Attachment 2 for an example of an annotated P&ID illustrating this concept. [Refer to hardcopy of NEI 00-01 for Attachment 2]

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Marked up and annotated P&ID drawings were primarily used as an aide in reviewing the credited system flow paths during the development of system and equipment logics that reside in the NSCA model and database.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 9.4)

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

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

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Nuclear Safety Capability Assessment Analysis Model Development detailed within NEDC 11-019, identifies the overall process utilized to identify the combinations of plant components for each plant system that is identified as being required to satisfy each of the Nuclear Safety Performance Criteria (NSPC) from Section 1.5.1 of NFPA 805.

A review of P&IDs, electrical drawings, etc. was performed to identify the NSCA systems, and to identify and develop the NSCA system-to equipment logic relationships (i.e., Boolean logic / success paths) and the NSCA equipment-to-equipment logic success path relationships (i.e., success paths).

Mechanical and electrical system components such as pumps, air-operated valves, motor-operated valves, and solenoid-operated valves, fans, heaters, electrically controlled circuit breakers, transformers, switchgear, motor control centers, batteries, battery chargers, inverters, distribution panels, automatic transfer switches, diesel generators and engines, strainers, instrumentation, and dampers, etc., which have an active function in achieving a safe and stable condition are included in the NSCA.

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

Control panels and discrete electrical and instrumentation components such as hand switches, relays, starters, fuses, indicating lights, molded case and other non-electrically operated circuit breakers, electrical disconnects, pull boxes, junction boxes, terminal boxes, signal converters, amplifiers, bistables, relay cards, instrument power supplies, etc. (excluding the transmitting devices and indicating devices), are not explicitly identified or included in the NFPA 805 Nuclear Safety

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

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

Equipment List, Appendix A of NEDC 11-019. These secondary components or sub-components are represented in the NSCA by virtue of the circuit conductors and cables that interconnect them to the primary component.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 9.0)

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

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

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

All selected equipment in support of a system required to achieve a safe and stable condition is appropriately designated as such in Appendix A of Calculation NEDC 11-019. The equipment list relates back to the systems. The specific equipment that support each system path are documented within Calculation NEDC 11-019, Tables 9.4.2-1 through 9.4.4-8 and are illustrated in the logic diagrams within Appendix E of Calculation NEDC 11-019.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Appendix A, Appendix E and Tables 9.4.2-1 through 9.4.4-8)

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

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

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The Nuclear Safety Equipment List identifies the credited equipment for the NSCA. Calculation NEDC 11-019, Appendix A, specifies the data included on the NFPA 805 Nuclear Safety Equipment List for each piece of equipment:

- Equipment Designation
- Description
- Fire Zone
- System
- Normal Position
- Failed Position
- HSD Position
- Remarks

While some of the suggested information to be included on the table has been excluded from the NFPA 805 Nuclear Safety Equipment List, the necessary information is captured in the NSCA database (EDISON/SAFE) in order to successfully perform the post-fire separation analysis.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Appendix A)

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

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

NEI 00-01 Ref

3.2.2.5 Identify Dependencies Between Equipment, Supporting Equipment, Safe Shutdown Systems and Safe Shutdown Paths

NEI 00-01 Section 3 Guidance

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.

[Refer to hardcopy of NEI 00-01 for Attachment 4]

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

The NSCA database (EDISON/SAFE) provides the tool to establish relational ties between selected methods, systems, components and cable that are credited to achieve a safe and stable condition.

Different combinations of equipment that successfully support the NSCA performance goal paths are illustrated on the logic diagrams in Calculation NEDC 11-019, Appendix E. The various equipment combinations can be achieved by choosing different logical branches that satisfy a particular system path. Support systems may not be included in the system logic diagrams. These relationships may instead be applied at the equipment logic level within the database. Equipment logics are defined in textual format in Section 9.4. Boolean logics associating each Equipment with their supporting cables were developed and logic statements entered into EDISON/SAFE. Equipment credited in support of the NSCA model and equipment logics are tabulated in Appendix C of Calculation NEDC 11-019.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 9.2, Appendix C and Appendix E)

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

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

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

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

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

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3 Safe Shutdown Cable Selection and Location	This section provides industry guidance on the recommended methodology and criteria for selecting safe shutdown cables and determining their potential impact on equipment required for achieving and maintaining safe shutdown of an operating nuclear power plant for the condition of an exposure fire. The Appendix R safe shutdown cable selection criteria are developed to ensure that all cables that could affect the proper operation or that could cause the maloperation of safe shutdown equipment are identified and that these cables are properly related to the safe shutdown equipment whose functionality they could affect. Through this cable-to-equipment relationship, cables become part of the safe shutdown path assigned to the equipment affected by the cable.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 10.0 describes the circuit analysis methodology used to meet the NFPA 805 requirements. Cables that are required to support the NFPA 805 equipment are maintained in the EDISON/SAFE database. The database provides cable data and circuit analysis data for each component and cable credited to achieve a safe and stable condition.

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 10.0)

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1 Criteria/Assumptions	To identify an impact to safe shutdown equipment based on cable routing, the equipment must have cables that affect it identified. Carefully consider how cables are related to safe shutdown equipment so that impacts from these cables can be properly assessed in terms of their ultimate impact on safe shutdown system equipment. Consider the following criteria when selecting cables that impact safe shutdown equipment:

Applicability	Comments
Not Applicable	None

Alignment Statement

Not Applicable

Alignment Basis

Introductory section. refer to following subsection for detailed guidance and bases.

Reference Documents

None

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.1 Criteria/Assumptions	The list of cables whose failure could impact the operation of a piece of safe shutdown equipment includes more than those cables connected to the equipment. The relationship between cable and affected equipment is based on a review of the electrical or elementary wiring diagrams. To assure that all cables that could affect the operation of the safe shutdown equipment are identified, investigate the power, control, instrumentation, interlock, and equipment status indication cables related to the equipment. Consider reviewing additional schematic diagrams to identify additional cables for interlocked circuits that also need to be considered for their impact on the ability of the equipment to operate as required in support of postfire safe shutdown. As an option, consider applying the screening criteria from Section 3.5 as a part of this section. For an example of this see Section 3.3.1.4.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The circuit analysis methodology is described in Calculation NEDC 11-019, Section 10.0, Circuit Identification and Analysis. This methodology includes consideration of hot shorts (external and internal), open circuits, shorts to ground, and spurious signals (including interlock and permissive circuits).

The EDISON/SAFE database provides circuit analysis data and relational tie between each component and cable credited in the analysis model.

Development of the safe shutdown separation analysis for CNS required that circuit identification and analysis be performed for each electrically controlled and/or operated NSCA component. This task was performed through a detailed engineering review of circuit schematics, connection diagrams, cable block diagrams, and/or other relevant controlled plant documentation. The purpose of circuit identification and analysis was to identify and classify all cables supporting a given safe shutdown component such that a component/cable logic could be developed in preparation for computerized separation analysis.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 10.0)

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

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

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 10.0 identifies the types of cables considered and selection criteria including control, permissive/interlock, indication/signal and annunciation circuits that could result in the spurious actuation of NSCA Equipment.

The EDISON/SAFE database provides relational tie between each component and cable credited to achieve a safe and stable condition.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 10.0)

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.3 Criteria/Assumptions	Electrical devices such as relays, switches and signal resistor units are considered to be acceptable isolation devices. In the case of instrument loops, review the isolation capabilities of the devices in the loop to determine that an acceptable isolation device has been installed at each point where the loop must be isolated so that a fault would not impact the performance of the safe shutdown instrument function.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criterion / assumption listed in Section 3.3.1.3 of NEI 00-01 is explicitly stated in Section 5.1.3 of the Calculation NEDC 11-019.

Cables containing circuit conductors required for post-fire operations, including circuits that could cause spurious operation are included within equipment cable logics for each of the NSCA Equipment. These cables are classified as "required." Cables that functioned as a portion of the control circuit, but that could not disable the control circuit required for safe and stable hot shutdown, were classified as "not required," and were not included in the cable logic.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 5.1.3, 10.2.2.1 and 10.2.3.2)

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.1.4 Criteria/Assumptions	Screen out cables for circuits that do not impact the safe shutdown function of a component (i.e., annunciator circuits, space heater circuits and computer input circuits) unless some reliance on these circuits is necessary. However, they must be isolated from the component's control scheme in such a way that a cable fault would not impact the performance of the circuit.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Cable selection was performed for each electrically controlled and/or operated NSCA component per the guidance outlined in NEI-00-01 "Guidance for Post-Fire Circuit Analysis," and insights provided in NUREG/CR-6850 "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities."

Cables associated with an NSCA component were classified as "required" or "not required" depending upon the possible effects that fire-induced cable failure could have upon the function(s) of the respective component.

Not required cables are generally only identified in the database for primary scheme (i.e., "on-scheme") cables that are determined not to be required for the NSCA, NPO, or Fire PRA. These cables are identified in the EDISON/SAFE database at the discretion of the preparer and reviewer of the circuit identification and analysis for each NSCA component to document that the primary scheme cables were indeed included and addressed in the circuit identification and analysis activity.

Cables that functioned as a portion of the control circuit, but that could not disable the control circuit required for safe and stable hot shutdown, were classified as "not required," and were not included in the cable logic.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 10.2.1 and 10.2.2.1)

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	**NEI 00-01 Section 3 Guidance**
3.3.1.5 Criteria/Assumptions	For each circuit requiring power to perform its safe shutdown function, identify the cable supplying power to each safe shutdown and/or required interlock component. Initially, identify only the power cables from the immediate upstream power source for these interlocked circuits and components (i.e., the closest power supply, load center or motor control center). Review further the electrical distribution system to capture the remaining equipment from the electrical power distribution system necessary to support delivery of power from either the offsite power source or the emergency diesel generators (i.e., onsite power source) to the safe shutdown equipment. Add this equipment to the safe shutdown equipment list. Evaluate the power cables for this additional equipment for associated circuits concerns.
Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Electrical power sources (e.g., AC/DC bus, MCC or distribution panel) providing motive power and/or control power for active components were identified and tabulated for each NSCA component. These electrical power sources were considered "active" components and subject to circuit analysis and cable selection.

Electrical power sources (e.g., AC/DC bus, MCC or distribution panel) providing motive power and/or control power for passive components were identified and tabulated for each NSCA component. These electrical power sources were considered as "active" safe shutdown components and subject to additional circuit analysis and cable selection if their failure (e.g., power supply de-energization or energization resulting from a fire-induced cable failure) could result in the spurious operation of the host passive component. If power supply failure could not result in spurious operation of the host passive component, the power supply was considered "passive," and not subject to circuit analysis and/or cable selection.

Calculation NEDC 11-019, Section 10.3, Associated Circuits of Concern Study, identifies the approach to address circuits of concerns by common power supply.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 10.2.2.2, 10.2.3 and 10.3)

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	**NEI 00-01 Section 3 Guidance**
3.3.1.6 Criteria/Assumptions	The automatic initiation logics for the credited post-fire safe shutdown systems are not required to support safe shutdown. Each system can be controlled manually by operator actuation in the main control room or emergency control station. If operator actions outside the MCR are necessary, those actions must conform to the regulatory requirements on manual actions. However, if not protected from the effects of fire, the fire-induced failure of automatic initiation logic circuits must not adversely affect any post-fire safe shutdown system function.

Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The cable selection process outlined in Calculation NEDC 11-019, Section 10.0 considers the affects of automatic, spurious and permissive signals.

Cables required to ensure positive remote manual control capability of a component were selected to provide the intended post-fire function. Credit was not taken for receipt of automatic/process signals initiating or assisting in initiation or operation of a system. Automatic signals that provided an automatic start (or trip) signal to components, however, were considered in the cable selection process as potential spurious concerns.

CNS takes credit for Operator Manual Actions (hot shutdown) that are not currently approved nor allowed by the NRC. These OMAs are currently characterized as compensatory actions and considered variances from deterministic requirements (VFDRs) that are being addressed / analyzed in accordance NFPA 805 FAQ 07-0030 and FAQ 08-0054 through the Fire Risk Evaluation process as part of transition to the new NFPA 805 licensing basis.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 10.0 and 10.2.1.1)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref

3.3.1.7 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

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

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 10.3 identifies the approach taken to address associated circuits concerns by common power supply.

Coordination of circuit protective devices has been evaluated and demonstrated via the performance and maintenance of several Breaker/Fuse Coordination Studies. As such, there was not a need for inclusion of associated circuits of this kind into the separation analysis.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 10.3)

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.3.2	Associated Circuit Cables	Appendix R, Section III.G.2, requires that separation features be provided for equipment and cables, including associated nonsafety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve hot shutdown. The three types of associated circuits were identified in Reference 6.1.5 and further clarified in a NRC memorandum dated March 22, 1982 from R. Mattson to D. Eisenhut, Reference 6.1.6. They are as follows: - Spurious actuations - Common power source - Common enclosure Cables Whose Failure May Cause Spurious Actuations Safe shutdown system spurious actuation concerns can result from fire damage to a cable whose failure could cause the spurious actuation/mal-operation of equipment whose operation could affect safe shutdown. These cables are identified in Section 3.3.3 together with the remaining safe shutdown cables required to support control and operation of the equipment. Common Power Source Cables The concern for the common power source associated circuits is the loss of a safe shutdown power source due to inadequate breaker/fuse coordination. In the case of a fire-induced cable failure on a non-safe shutdown load circuit supplied from the safe shutdown power source, a lack of coordination between the upstream supply breaker/fuse feeding the safe shutdown power source and the load breaker/fuse supplying the non-safe shutdown faulted circuit can result in loss of the safe shutdown bus. This would result in the loss of power to the safe shutdown equipment supplied from that power source preventing the safe shutdown equipment from performing its required safe shutdown function. Identify these cables together with the remaining safe shutdown cables required to support control and operation of the equipment. Refer to Section 3.5.2.4 for an acceptable methodology for analyzing the impact of these cables on post-fire safe shutdown. Common Enclosure Cables The concern with common enclosure associated circuits is fire damage to a cable whose failure could propagate to other safe shutdown cables in the same enclosure either because the circuit is not properly protected by an isolation device (breaker/fuse) such that a fire-induced fault could result in ignition along its length, or by the fire propagating along the cable and into an adjacent fire area. This fire spread to an adjacent fire area could impact safe shutdown equipment in that fire area, thereby resulting in a condition that exceeds the criteria and assumptions of this methodology (i.e., multiple fires). Refer to Section 3.5.2.5 for an acceptable methodology for analyzing the impact of these cables on post-fire safe shutdown.

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The circuit analysis methodology is described in Calculation NEDC 11-019, Section 10.0. This methodology addresses spurious operation and includes consideration of hot shorts (external and internal), open circuits, shorts to ground, and spurious signals (including interlock and permissive circuits).

Associated circuits by common power supply are discussed in Calculation NEDC 11-019, Section 10.3. Associated Circuit by Common Enclosure is presented in Calculation NEDC 11-019, Section 10.3.4.

Refer to subsequent sections (3.5.2.4 & 3.5.2.5) for alignment details.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 10.0, 10.3.3 and 10.3.4)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

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

Applicability	**Comments**
Not Applicable	None

Alignment Statement

Not Applicable

Alignment Basis

Introductory section. Refer to subsequent sections for evaluation of specific criteria/guidance.

Reference Documents

None

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

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

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Appropriate drawings for applicable electrical equipment listed in the NFPA 805 Nuclear Safety Equipment List, Appendix A of NEDC 11-019 were reviewed during the cable selection process. Component work sheets and cable block diagrams have been created as appropriate. Any Input drawings used during the cable selection process have been identified on the worksheets/block diagrams. Power supplies have been identified for all active equipment.

Refer to Section 3.5.2.4 for alignment details related to coordination studies.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.3.2 Identify Interlocked Circuits and Cables Whose Spurious Operation or Mal-operation Could Affect Shutdown	In reviewing each control circuit, investigate interlocks that may lead to additional circuit schemes, cables and equipment. Assign to the equipment any cables for interlocked circuits that can affect the equipment. While investigating the interlocked circuits, additional equipment or power sources may be discovered. Include these interlocked equipment or power sources in the safe shutdown equipment list (refer to NEI 00-01 Rev 1 Figure 3-3) if they can impact the operation of the equipment under consideration. [Refer to hardcopy of NEI 00-01 for Figure]

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The circuit analysis methodology is described in Calculation NEDC 11-019, Section 10.0. This methodology includes consideration of interlock and permissive circuits.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 10.0)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.3.3 Assign Cables to the Safe Shutdown Equipment	Given the criteria/assumptions defined in Section 3.3.1, identify the cables required to operate or that may result in mal-operation of each piece of safe shutdown equipment. Tabulate the list of cables potentially affecting each piece of equipment in a relational database including the respective drawing numbers, their revision and any interlocks that are investigated to determine their impact on the operation of the equipment. In certain cases, the same cable may support multiple pieces of equipment. Relate the cables to each piece of equipment, but not necessarily to each supporting secondary component. If adequate coordination does not exist for a particular circuit, relate the power cable to the power source. This will ensure that the power source is identified as affected equipment in the fire areas where the cable may be damaged.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Circuit analysis data for each component and cable credited to achieve a safe and stable condition are contained in a relational database (EDISON/SAFE). Section 11.0 of Calculation NEDC 11-019 describes the development of the model in the analysis database.

There were no cases of inadequate coordination identified for a particular circuit. Refer to Section 3.5.2.4 for alignment details related to coordination studies.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 11.0)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.5	Circuit Analysis and Evaluation	This section on circuit analysis provides information on the potential impact of fire on circuits used to monitor, control and power safe shutdown equipment. Applying the circuit analysis criteria will lead to an understanding of how fire damage to the cables may affect the ability to achieve and maintain post-fire safe shutdown in a particular fire area. This section should be used in conjunction with Section 3.4, to evaluate the potential fire-induced impacts that require mitigation. Appendix R Section III.G.2 identifies the fire-induced circuit failure types that are to be evaluated for impact from exposure fires on safe shutdown equipment. Section III.G.2 of Appendix R requires consideration of hot shorts, shorts-to-ground and open circuits.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Section 10.0 of Calculation NEDC 11-019 describes the circuit analysis methodology used to meet the NFPA 805 requirements.

The fire induced cable damage postulated to occur within the NSCA includes hot shorts, shorts to ground, and open circuits.

Cables that are required to support the NFPA 805 equipment are maintained in the analysis database EDISON/SAFE. The database provides cable and circuit analysis data for each component and cable credited to achieve a safe and stable condition.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 10.0 and 10.3.2.1)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.5.1	Criteria/Assumptions	Apply the following criteria/assumptions when performing fire-induced circuit failure evaluations.

Applicability	Comments
Not Applicable	None

Alignment Statement

Not Applicable

Alignment Basis

Introductory Section. Refer to subsequent sections for evaluation of specific criteria/assumptions.

Reference Documents

None

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref

3.5.1.1 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

Consider the following circuit failure types on each conductor of each unprotected safe shutdown cable to determine the potential impact of a fire on the safe shutdown equipment associated with that conductor.

- A hot short may result from a fire-induced insulation breakdown between conductors of the same cable, a different cable or from some other external source resulting in a compatible but undesired impressed voltage or signal on a specific conductor. A hot short may cause a spurious operation of safe shutdown equipment.

- An open circuit may result from a fire-induced break in a conductor resulting in the loss of circuit continuity. An open circuit may prevent the ability to control or power the affected equipment. An open circuit may also result in a change of state for normally energized equipment. (e.g. [for BWRs] loss of power to the Main Steam Isolation Valve (MSIV) solenoid valves due to an open circuit will result in the closure of the MSIVs). Note that RIS 2004-03 indicates that open circuits, as an initial mode of cable failures, are considered to be of very low likelihood. The risk-informed inspection process will focus on failures with relatively high probabilities.

- A short-to-ground may result from a fire-induced breakdown of a cable insulation system, resulting in the potential on the conductor being applied to ground potential. A short-to-ground may have all of the same effects as an open circuit and, in addition, a short-to-ground may also cause an impact to the control circuit or power train of which it is a part.

Consider the three types of circuit failures identified above to occur individually on each conductor of each safe shutdown cable on the required safe shutdown path in the fire area.

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions from NEI 00-01 Chapter 3, lists criteria / assumptions pertaining to the NSCA fire area assessment. The criteria / assumptions listed in Section 3.5.1.1 of NEI 00-01 are explicitly stated in the Section 5.1.5 of Calculation NEDC 11-019.

The three circuit failure types have all been identified and considered within the NSCA. Calculation NEDC 11-019 describes these circuit failure types within Section 10.3, Associated Circuits of Concern Study.

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 5.1.5 and 10.3)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.1.2 Criteria/Assumptions	Assume that circuit contacts are positioned (i.e., open or closed) consistent with the normal mode/position of the safe shutdown equipment as shown on the schematic drawings. The analyst must consider the position of the safe shutdown equipment for each specific shutdown scenario when determining the impact that fire damage to a particular circuit may have on the operation of the safe shutdown equipment.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions from NEI 00-01 Chapter 3, lists criteria / assumptions pertaining to the NSCA fire area assessment. The criteria / assumptions listed in Section 3.5.1.2 of NEI 00-01 are explicitly stated in Section 5.1.5 of Calculation NEDC 11-019.

All relay, position switch, and control switch contacts in the control circuits are in the position or status that corresponds to the normal condition of the device. Test and transfer switches in control circuits are in their normal positions. This assumption is reasonable and is consistent with the single fire assumption identified above.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 5.1.5 and 5.2.1)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref

3.5.1.3 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

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

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions from NEI 00-01 Chapter 3, lists criteria / assumptions pertaining to the NSCA fire area assessment. The criteria / assumptions listed in Section 3.5.1.3 of NEI 00-01 are explicitly stated in Section 5.1.5 of Calculation NEDC 11-019.

Fire induced damage that causes spurious operation exists until action has been taken to isolate the given circuit. The fire is not assumed to clear the circuit fault. The analysis utilizes the following mitigating methods to resolve spurious operation; plant modification, pre-fire de-energization and recovery action.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 5.1.5 and 10.3.2.5)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.1.4 Criteria/Assumptions	When both trains are in the same fire area outside of primary containment, all cables that do not meet the separation requirements of Section III.G.2 are assumed to fail in their worst case configuration.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criteria / assumption listed in Section 3.5.1.4 of NEI 00-01 is explicitly stated in Section 5.1.5 of Calculation NEDC 11-019.

The deterministic separation requirements of NFPA 805 Section 4.2.3 are similar and assume the same failure criteria of Section III.G.2. Additionally, NFPA 805 allows for a performance based analysis of fire areas that do not meet the deterministic requirements. NFPA 805 Section 4.2.4 details the requirements of the performance based analysis.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.1.5)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.1.5 Criteria/Assumptions	The following guidance provides the NRC inspection focus from Bin 1 of RIS 2004-03 in order to identify any potential combinations of spurious operations with higher risk significance. Bin 1 failures should also be the focus of the analysis; however, NRC has indicated that other types of failures required by the regulations for analysis should not be disregarded even if in Bin 2 or 3. If Bin 1 changes in subsequent revisions of RIS 2004-03, the guidelines in the revised RIS should be followed. Cable Failure Modes. For multiconductor cables testing has demonstrated that conductor-to-conductor shorting within the same cable is the most common mode of failure. This is often referred to as "intra-cable shorting." It is reasonable to assume that given damage, more than one conductor-to-conductor short will occur in a given cable. A second primary mode of cable failure is conductor-to-conductor shorting between separate cables, commonly referred to as "inter-cable shorting." Inter-cable shorting is less likely than intra-cable shorting. Consistent with the current knowledge of fire-induced cable failures, the following configurations should be considered: A. For any individual multiconductor cable (thermoset or thermoplastic), any and all potential spurious actuations that may result from intra-cable shorting, including any possible combination of conductors within the cable, may be postulated to occur concurrently regardless of number. However, as a practical matter, the number of combinations of potential hot shorts increases rapidly with the number of conductors within a given cable. For example, a multiconductor cable with three conductors (3C) has 3 possible combinations of two (including desired combinations), while a five conductor cable (5C) has 10 possible combinations of two (including desired combinations), and a seven conductor cable (7C) has 21 possible combinations of two (including desired combinations). To facilitate an inspection that considers most of the risk presented by postulated hot shorts within a multiconductor cable, inspectors should consider only a few (three or four) of the most critical postulated combinations. B. For any thermoplastic cable, any and all potential spurious actuations that may result from intra-cable and inter-cable shorting with other thermoplastic cables, including any possible combination of conductors within or between the cables, may be postulated to occur concurrently regardless of number. (The consideration of thermoset cable inter-cable shorts is deferred pending additional research.) C. For cases involving the potential damage of more than one multiconductor cable, a maximum of two cables should be assumed to be damaged concurrently. The spurious actuations should be evaluated as previously described. The consideration of more than two cables being damaged (and subsequent spurious actuations) is deferred pending additional research. D. For cases involving direct current (DC) circuits, the potential spurious operation due to failures of the associated control cables (even if the spurious operation requires two concurrent hot shorts of the proper polarity, e.g., plus-to-plus and

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

minus-to-minus) should be considered when the required source and target conductors are each located within the same multiconductor cable.

E. Instrumentation Circuits. Required instrumentation circuits are beyond the scope of this associated circuit approach and must meet the same requirements as required power and control circuits. There is one case where an instrument circuit could potentially be considered an associated circuit. If fire-induced damage of an instrument circuit could prevent operation (e.g., lockout permissive signal) or cause maloperation (e.g., unwanted start/stop/reposition signal) of systems necessary to achieve and maintain hot shutdown, then the instrument circuit may be considered an associated circuit and handled accordingly.

Likelihood of Undesired Consequences

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

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

CNS performed a Fire Protection self assessment (April 2005) to review the existing Appendix R Safe Shutdown Analysis with respect to NRC RIS 2004-03, Bin 1 issues for various cable failure modes.

The procedure for NFPA 805 / FPRA cable selection and circuit analysis (EPM-DP-EP-004) references the RIS mention above. However, understanding that RIS applies primarily for the Reactor Oversight Process (ROP), it does not place the same limitations or restrictions on circuit failure modes analysis performed under the self assessment as part of the new licensing basis proposed under NFPA 805. Specifically no limitations have been placed on cable type (thermoplastic/thermoset) or short failure modes (intra-/inter-cable). In addition, as part of the NFPA 805 transition, NPPD performed a series of expert panel systems reviews to address the potential for multiple spurious operations (MSOs). Results were fed back into the NSCA and Fire PRA as necessary. The treatment of MSOs is to be documented in the applicable section of the LAR Transition Report.

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Reference Documents

P1731-RPT-001, "CNS Associated Circuits Assessment for RIS 2004-03," Revision 0

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.2 Types of Circuit Failures	Appendix R requires that nuclear power plants must be designed to prevent exposure fires from defeating the ability to achieve and maintain post-fire safe shutdown. Fire damage to circuits that provide control and power to equipment on the required safe shutdown path and any other equipment whose spurious operation/mal-operation could affect shutdown in each fire area must be evaluated for the effects of a fire in that fire area. Only one fire at a time is assumed to occur. The extent of fire damage is assumed to be limited by the boundaries of the fire area. Given this set of conditions, it must be assured that one redundant train of equipment capable of achieving hot shutdown is free of fire damage for fires in every plant location. To provide this assurance, Appendix R requires that equipment and circuits required for safe shutdown be free of fire damage and that these circuits be designed for the fire-induced effects of a hot short, short-to-ground, and open circuit. With respect to the electrical distribution system, the issue of breaker coordination must also be addressed. This section will discuss specific examples of each of the following types of circuit failures: - Open circuit - Short-to-ground - Hot short.

Applicability	Comments
Not Applicable	None

Alignment Statement

Not Applicable

Alignment Basis

Introductory Section. Refer to subsequent sections for evaluation of specific circuit failure types.

Reference Documents

None

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.2.1 Circuit Failures Due to an Open Circuit	This section provides guidance for addressing the effects of an open circuit for safe shutdown equipment. An open circuit is a fire-induced break in a conductor resulting in the loss of circuit continuity. An open circuit will typically prevent the ability to control or power the affected equipment. An open circuit can also result in a change of state for normally energized equipment. For example, a loss of power to the main steam isolation valve (MSIV) solenoid valves [for BWRs] due to an open circuit will result in the closure of the MSIV. NOTE: The EPRI circuit failure testing indicated that open circuits are not likely to be the initial fire-induced circuit failure mode. Consideration of this may be helpful within the safe shutdown analysis. Consider the following consequences in the safe shutdown circuit analysis when determining the effects of open circuits: - Loss of electrical continuity may occur within a conductor resulting in deenergizing the circuit and causing a loss of power to, or control of, the required safe shutdown equipment. - In selected cases, a loss of electrical continuity may result in loss of power to an interlocked relay or other device. This loss of power may change the state of the equipment. Evaluate this to determine if equipment fails safe. - Open circuit on a high voltage (e.g., 4.16 kV) ammeter current transformer (CT) circuit may result in secondary damage. [Refer to hardcopy of NEI 00-01 for Figure 3.5.2-1 that shows example of open circuits]

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The fire induced cable damage postulated to occur includes hot shorts, shorts to ground, and open circuits. Multiple simultaneous circuit failures are postulated in the circuit identification and analysis (affecting multiple cables, affecting multiple conductors within cables). No limit is prescribed to the number or type circuit failures that are postulated to occur except as modified by the following: In consideration of spurious actuations, all possible functional failure states must be evaluated as a result of one or more of the above failure modes. For three-phase AC electrical circuits, the probability of getting a hot short on all three phases in the proper sequence to cause spurious operation of a motor is considered sufficiently low as to not require evaluation except for any cases involving high/low pressure interfaces. For ungrounded DC circuits, if it would require that two shorts of the proper polarity without grounding cause the spurious operation, then no further evaluation is necessary except for cases involving high/ low pressure interfaces.

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Calculation NEDC 11-071, Evaluation of Current Transformer Open Secondary, during a Fire, concludes the following: a Current Transformer (CT) secondary open circuit will produce very short duration high voltage pulses when the circuit cable has high cable insulation resistance (IR). However, based on cable fire data, high cable IR during a fire event is indicative of an intact cable, not one with an open conductor. More likely, an open conductor exists when the cable is damaged by the fire and the cable IR is 100 ohms or less. At this degraded level the CT voltage pulses are much lower. The possibility of the much lower voltages causing a secondary fire at the CT location is unlikely.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.2.3)

Calculation NEDC 11-071, "Evaluation of Current Transformer Open Secondary during a Fire," Revision 0

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

NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.2.2 Circuit Failures Due to a Short-to-Ground	This section provides guidance for addressing the effects of a short-to-ground on circuits for safe shutdown equipment. A short-to-ground is a fire-induced breakdown of a cable insulation system resulting in the potential on the conductor being applied to ground potential. A short-to-ground can cause a loss of power to or control of required safe shutdown equipment. In addition, a short-to-ground may affect other equipment in the electrical power distribution system in the cases where proper coordination does not exist. Consider the following consequences in the post-fire safe shutdown analysis when determining the effects of circuit failures related to shorts-to-ground: - A short to ground in a power or a control circuit may result in tripping one or more isolation devices (i.e. breaker/fuse) and causing a loss of power to or control of required safe shutdown equipment. - In the case of certain energized equipment such as HVAC dampers, a loss of control power may result in loss of power to an interlocked relay or other device that may cause one or more spurious operations. [Refer to hardcopy of NEI 00-01 for Figures 3.5.2-2 and 3.5.2-3 that shows example of short-to-ground on ungrounded and grounded circuits]

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The fire-induced cable damage postulated to occur includes hot shorts, shorts-to-ground, and open circuits. Multiple simultaneous circuit failures are postulated in the circuit identification and analysis (affecting multiple cables, affecting multiple conductors within cables). No limit is prescribed to the number or type of circuit failures that are postulated to occur, except as modified by the following: In consideration of spurious actuations, all possible functional failure states must be evaluated as a result of one or more of the above failure modes. For three-phase AC electrical circuits, the probability of getting a hot short on all three phases in the proper sequence to cause spurious operation of a motor is considered sufficiently low as to not require evaluation, except for any cases involving high/low pressure interfaces. For ungrounded DC circuits, if it would require that two shorts of the proper polarity without grounding cause the spurious operation, then no further evaluation is necessary except for cases involving high/ low pressure interfaces.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.2.3)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.2.3 Circuit Failures Due to a Hot Short	This section provides guidance for analyzing the effects of a hot short on circuits for required safe shutdown equipment. A hot short is defined as a fire induced insulation breakdown between conductors of the same cable, a different cable or some other external source resulting in an undesired impressed voltage on a specific conductor. The potential effect of the undesired impressed voltage would be to cause equipment to operate or fail to operate in an undesired manner. Consider the following specific circuit failures related to hot shorts as part of the post-fire safe shutdown analysis: - A hot short between an energized conductor and a de-energized conductor within the same cable may cause a spurious actuation of equipment. The spuriously actuated device (e.g., relay) may be interlocked with another circuit that causes the spurious actuation of other equipment. This type of hot short is called a conductor-to-conductor hot short or an internal hot short. - A hot short between any external energized source such as an energized conductor from another cable (thermoplastic cables only) and a de-energized conductor may also cause a spurious actuation of equipment. This is called a cable-to-cable hot short or an external hot short. Cable-to-cable hot shorts between thermoset cables are not postulated to occur pending additional research. [Refer to hardcopy of NEI 00-01 for Figures 3.5.2-4 and 3.5.2-5 that shows example of hot shorts on ungrounded and grounded circuits]

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Consider the following circuit failure types on each conductor of each unprotected safe shutdown cable to determine the potential impact of a fire on the safe shutdown equipment associated with that conductor. A hot short may result from a fire-induced insulation breakdown between conductors of the same cable, a different cable, or from some other external source resulting in a compatible but undesired impressed voltage or signal on a specific conductor. A hot short may cause a spurious operation of safe shutdown equipment.

Circuit identification and analysis for the CNS NSCA includes consideration for multiple concurrent hot shorts, open circuits, and shorts-to-ground.

The fire induced cable damage postulated to occur includes hot shorts, shorts-to-ground, and open circuits. In consideration of spurious actuations, all possible

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

functional failure states must be evaluated as a result of one or more of the above failure modes.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 5.1.5, 10.0 and 10.3.2.1)

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.2.4 Circuit Failures Due to Inadequate Circuit Coordination	The evaluation of associated circuits of a common power source consists of verifying proper coordination between the supply breaker/fuse and the load breakers/fuses for power sources that are required for safe shutdown. The concern is that, for fire damage to a single power cable, lack of coordination between the supply breaker/fuse and the load breakers/fuses can result in the loss of power to a safe shutdown power source that is required to provide power to safe shutdown equipment. A coordination study should demonstrate the coordination status for each required common power source. For coordination to exist, the time-current curves for the breakers, fuses and/or protective relaying must demonstrate that a fault on the load circuits is isolated before tripping the upstream breaker that supplies the bus. Furthermore, the available short circuit current on the load circuit must be considered to ensure that coordination is demonstrated at the maximum fault level. The methodology for identifying potential associated circuits of a common power source and evaluating circuit coordination cases of associated circuits on a single circuit fault basis is as follows: - Identify the power sources required to supply power to safe shutdown equipment. - For each power source, identify the breaker/fuse ratings, types, trip settings and coordination characteristics for the incoming source breaker supplying the bus and the breakers/fuses feeding the loads supplied by the bus. - For each power source, demonstrate proper circuit coordination using acceptable industry methods. - For power sources not properly coordinated, tabulate by fire area the routing of cables whose breaker/fuse is not properly coordinated with the supply breaker/fuse. Evaluate the potential for disabling power to the bus in each of the fire areas in which the associated circuit cables of concern are routed and the power source is required for safe shutdown. Prepare a list of the following information for each fire area: - Cables of concern. - Affected common power source and its path. - Raceway in which the cable is enclosed. - Sequence of the raceway in the cable route. - Fire zone/area in which the raceway is located. For fire zones/areas in which the power source is disabled, the effects are mitigated by appropriate methods.

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

- Develop analyzed safe shutdown circuit dispositions for the associated circuit of concern cables routed in an area of the same path as required by the power source. Evaluate adequate separation based upon the criteria in Appendix R, NRC staff guidance, and plant licensing bases.

Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Circuit coordination has been evaluated by NPPD under Breaker/Fuse Coordination Studies NEDC 86-105B, "Critical AC Coordination", NEDC 86-105D, "Critical DC Coordination", NEDC 86-105F, "Non-critical AC Coordination" and NEDC 09-028, "120V Fuse Coordination". The results of these calculations are summarized within Section 10.3.3 of Calculation NEDC 11-019.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 10.3.3)

Calculation NEDC 86-105B, "Critical AC Coordination, " Revision 8

Calculation NEDC 86-105D, "Critical DC Coordination"

Calculation NEDC 86-105F, "Non-critical AC Coordination" Revision 6

Calculation NEDC 09-028, "120V Fuse Coordination," Revision 0

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.5.2.5 Circuit Failures Due to Common Enclosure Concerns	The common enclosure associated circuit concern deals with the possibility of causing secondary failures due to fire damage to a circuit either whose isolation device fails to isolate the cable fault or protect the faulted cable from reaching its ignition temperature, or the fire somehow propagates along the cable into adjoining fire areas. The electrical circuit design for most plants provides proper circuit protection in the form of circuit breakers, fuses and other devices that are designed to isolate cable faults before ignition temperature is reached. Adequate electrical circuit protection and cable sizing are included as part of the original plant electrical design maintained as part of the design change process. Proper protection can be verified by review of as-built drawings and change documentation. Review the fire rated barrier and penetration designs that preclude the propagation of fire from one fire area to the next to demonstrate that adequate measures are in place to alleviate fire propagation concerns.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Review of applicable documentation determined that the application and sizing of cable and circuit protection devices along with breaker/fuse coordination has been implemented into plant design to eliminate the requirement to postulate circuit failures due to common enclosures.

Electrical penetrations are fire-sealed at their boundary penetrations with fire stops installed equivalent to those required for the boundary. Exceptions are required to be addressed and justified through engineering evaluations.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 3.0, 10.3.4 and 10.3.4.1)

Burns & Roe, "Engineering Criteria Document for Cooper Nuclear Station," Dated 6/3/70

CNS Design Criteria Document DCD-5, "DC Electrical Distribution System," Revision 0

CNS Design Criteria Document DCD-4, "AC Electrical Distribution System," Revision 0

Calculation NEDC 09-028, "120V Fuse Coordination," Revision 0

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NFPA 805 Section: 2.4.2.2 Nuclear Safety Capability Circuit Analysis

Calculation NEDC 86-105D, "Critical DC Coordination"

Calculation NEDC 86-105F, "Non-critical AC Coordination," Revision 6

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NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location

Physical location of equipment and cables shall be identified.

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.3.4 Identify Routing of Cables	Identify the routing for each cable including all raceway and cable endpoints. Typically, this information is obtained from joining the list of safe shutdown cables with an existing cable and raceway database.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The routing information for each cable required for the NSCA has been entered into EDISON/SAFE for use in the analysis. The routing information will be used to determine cable fire zones to support the NSCA analysis.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 10.3.5)

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NFPA 805 Section: 2.4.2.3 Nuclear Safety Equipment and Cable Location

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.3.3.5 Identify Location of Raceway and Cables by Fire Area	Identify the fire area location of each raceway and cable endpoint identified in the previous step and join this information with the cable routing data. 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

The methodology for the identification of locating credited equipment and cables is described in Section 10.3.5 of Calculation NEDC 11-019. Location information is provided on a fire zone basis for each equipment and cable credited in the analysis.

The routing information (tray, conduit, etc.) of each required cable was reviewed to determine the fire zone locations. Cable fire zones were determined by a review of the plant equipment and cable and conduit layout drawings. This information was entered into EDISON/SAFE for use in the NSCA model.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 10.3.5)

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

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

NEI 00-01 Ref		NEI 00-01 Section 3 Guidance
3.4	Fire Area Assessment and Compliance Strategies	By determining the location of each component and cable by fire area and using the cable to equipment relationships described above, the affected safe shutdown equipment in each fire area can be determined. Using the list of affected equipment in each fire area, the impacts to safe shutdown systems, paths and functions can be determined. Based on an assessment of the number and types of these impacts, the required safe shutdown path for each fire area can be determined. The specific impacts to the selected safe shutdown path can be evaluated using the circuit analysis and evaluation criteria contained in Section 3.5 of this document. 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

Aligns

Alignment Basis

The EDISON/SAFE module provides the capability to perform an automated analysis utilizing a model composed of plant systems, equipment, cables, and their physical locations. The analysis uses a Boolean logic evaluation method that supports success path relationships enabling analysis and graphic display capability of each fire zone/fire area. Consequently, EDISON/SAFE contains the complete Cooper Nuclear Station (CNS) Nuclear Safety Capability Assessment (NSCA) analysis including the equipment, cable, fire zone/area, logic, analysis results and compliance strategy information necessary to support the NSCA.

Refer to subsequent sections for evaluation of specific criteria/assumptions.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 12.1)

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	**NEI 00-01 Section 3 Guidance**
3.4.1 Criteria/Assumptions	The following criteria and assumptions apply when performing fire area compliance assessment to mitigate the consequences of the circuit failures identified in the previous sections for the required safe shutdown path in each fire area.

Applicability	**Comments**
Not Applicable	None

Alignment Statement

Not Applicable

Alignment Basis

Introductory section. Refer to subsequent sections for evaluation of specific criteria/guidance.

Reference Documents

None

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	**NEI 00-01 Section 3 Guidance**
3.4.1.1 Criteria/Assumptions	Assume only one fire in any single fire area at a time.

Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criteria / assumption listed in Section 3.4.1.1 of NEI 00-01 is explicitly stated in Section 5.1.4 of Calculation NEDC 11-019.

The NSCA assumes only one fire in a single fire area at a time.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.1.4)

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.2 Criteria/Assumptions	Assume that the fire may affect all unprotected cables and equipment within the fire area. This assumes that neither the fire size nor the fire intensity is known. This is conservative and bounds the exposure fire that is required by the regulation.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criteria / assumption listed in Section 3.4.1.2 of NEI 00-01 is explicitly stated in Section 5.1.4 of Calculation NEDC 11-019.

All unprotected equipment and cables are fire-affected within a fire area for the NSCA deterministic analysis that is required to determine the VFDRs. Additional performance-based analysis may be utilized per NFPA 805.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.1.4)

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref

3.4.1.3 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

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

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

Calculation NEDC 11-019, Section 5.1, NSCA Criteria / Assumptions for NEI 00-01, lists criteria / assumptions pertaining to the NSCA model development and component selection. This criteria / assumption listed in Section 3.4.1.3 of NEI 00-01 is explicitly stated in Section 5.1.4 of Calculation NEDC 11-019.

The NSCA requires one success path for each performance goal be available to achieve a safe and stable condition. This compliance strategy is determined by performing a deterministic analysis and resolving any fire-affected equipment and/or cables within that success path. Any failures within the success path that require OMA are further investigated with the use of performance-based methods. All failures within the credited compliance strategies are analyzed and resolved to ensure the credited strategy is properly protected and available.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 5.1.4)

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	**NEI 00-01 Section 3 Guidance**
3.4.1.4 Criteria/Assumptions	Use manual actions where appropriate to achieve and maintain post-fire safe shutdown conditions in accordance with NRC requirements.

Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Operator manual actions are considered variances from deterministic requirements (VFDRs) that are being addressed / analyzed in accordance FAQ 30 / 54 through the Fire Risk Evaluation process as part of transition to the new NFPA 805 licensing basis.

Recovery actions can be performed as part of a performance-based, risk-informed approach subject to the limitations of Chapter 4 of NFPA 805 to mitigate a spurious actuation or achieve and maintain a Nuclear Safety Performance Criterion. The NFPA 805 recovery action assessment documented in CNS Calculation NEDC 11-020 “Recovery Action Transition," documents the results of an evaluation and documentation of recovery actions used as resolutions to VFDRs at the Cooper Nuclear Station.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 12.3)

NEDC 11-020, “Recovery Action Transition”, Rev 0

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref

3.4.1.5 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

Where appropriate to achieve and maintain cold shutdown within 72 hours, use repairs to equipment required in support of post-fire shutdown.

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

The 72-hour requirement from NEI 00-01 is only applicable to the 10 CFR 50 Appendix R licensing basis. Per Section 1.3.1 of NFPA 805, given a fire, a plant is not required to transition to cold shutdown within 72 hours but instead provide reasonable assurance to achieve and maintain the fuel in a safe and stable condition. For CNS, the required end state of "safe and stable" under NFPA 805 will be met when the plant is in a stable hot shutdown configuration.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 7.5)

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

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref

3.4.1.6 Criteria/Assumptions

NEI 00-01 Section 3 Guidance

Appendix R compliance requires that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage (III.G.1.a). When cables or equipment, including associated circuits, are within the same fire area outside primary containment and separation does not already exist, provide one of the following means of separation for the required safe shutdown path(s):

- Separation of cables and equipment and associated nonsafety 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 nonsafety circuits of redundant trains within the same fire area by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area (III.G.2.b).

- Enclosure of cable and equipment and associated non-safety circuits of one redundant train within a fire area in a fire barrier having a one-hour rating. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area (III.G.2.c).

For fire areas inside noninerted containments, the following additional options are also available:

- Separation of cables and equipment and associated nonsafety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards (III.G.2.d);

- Installation of fire detectors and an automatic fire suppression system in the fire area (III.G.2.e); or

- Separation of cables and equipment and associated non-safety circuits of redundant trains by a noncombustible radiant energy shield (III.G.2.f).

Use exemptions, deviations and licensing change processes to satisfy the requirements mentioned above and to demonstrate equivalency depending upon the plant's license requirements.

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

Alignment Basis

As part of Transition to NFPA 805, similar deterministic approaches (NFPA 805, Section 4.2.3) or performance-based approaches (NFPA 805 Section 4.2.4) will be used to achieve compliance with the new regulation. If performance-based methods are used then variances from deterministic requirements (VFDRs) will be identified and addressed using the fire risk evaluation process.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 12.1.2)

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	**NEI 00-01 Section 3 Guidance**
3.4.1.7 Criteria/Assumptions	Consider selecting other equipment that can perform the same safe shutdown function as the impacted equipment. In addressing this situation, each equipment impact, including spurious operations, is to be addressed in accordance with regulatory requirements and the NPP's current licensing basis.

Applicability	**Comments**
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Numerous systems are used to accomplish the NFPA 805 Performance Goals. This includes a list of equipment comprising the system and the logical relationship between the system and the equipment. The system function may be accomplished by more than one set of equipment and this is shown by an "OR" logic relationship. The system-equipment logical relationship is shown in both a tabular and graphic format.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 5.1.4 and 9.4)

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.1.8 Criteria/Assumptions	Consider the effects of the fire on the density of the fluid in instrument tubing and any subsequent effects on instrument readings or signals associated with the protected safe shutdown path in evaluating post-fire safe shutdown capability. This can be done systematically or via procedures such as Emergency Operating Procedures.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

As part of the NFPA 805 Transition, the impact of fire on instrument tubing sensing lines was evaluated considering the fire area location of the instrument sensing lines. The sensing lines for the applicable process monitoring instruments are included in the NSCA model, and evaluated similar to a cable, such that the instrument is assumed to fail in areas containing its associated tubing unless an evaluation notes otherwise. The sensing lines for applicable process monitoring instruments are welded steel; therefore, the pressure boundary will not be breached as a result of fire damage. For instruments that are credited as active or for instruments whose failure could be detrimental, the fire zone location of the instrument, together with the fire zone(s) where the instrument's sensing line is located, if different than that of the instrument, are associated in EDISON/SAFE with the instrument. In this manner, a fire would fail the instrument in the fire zone containing the instrument and in any fire zone containing the sensing line.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 9.5)

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NFPA 805 Section: 2.4.2.4 Fire Area Assessment

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

Applicability	Comments
Not Applicable	None

Alignment Statement

Not Applicable

Alignment Basis

Introductory Section. Refer to subsequent sections for evaluation of specific guidance.

Reference Documents

None

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

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.2.1 Identify the Affected Equipment by Fire Area	Identify the safe shutdown cables, equipment and systems located in each fire area that may be potentially damaged by the fire. Provide this information in a report format. The report may be sorted by fire area and by system in order to understand the impact to each safe shutdown path within each fire area (see Attachment 5 for an example of an Affected Equipment Report). [Refer to hardcopy of NEI 00-01 for Attachment]

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

An automated analysis is completed within EDISON/SAFE to determine the fire affected equipment and cables for each fire area. The analysis steps through the logic statements (system logic statements, performance goal success paths, equipment-equipment success paths, equipment-cable success paths, and equipment-system success paths) beginning with the initially failed items list (systems, equipment, and cables). Initially failed items list are located in the fire zone being analyzed, or specified at the beginning of the analysis, or have a resolution type of initially failed for the fire zone or zones being analyzed.

The analysis structure and results are displayed as a tree structure beginning with performance goals on the left leading to system logic. The system logic display shows all equipment in every success path. This tree structure diagram of the EDISON/SAFE model shows the results of the analysis in the following way:

Failed equipment, cables, systems, and performance goals (displayed in red text)
Resolved equipment, cables, systems (displayed in blue text, with check mark)
Design Change (ACP) logics, if applicable (displayed in bold text)
All items that show with black text are unaffected by the initial failures used in this analysis.

The failed equipment and cables can be listed in report format for each fire area from EDISON/SAFE.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 11.1.3)

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

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref

3.4.2.2 Determine the Shutdown Paths Least Impacted By a Fire in Each Fire Area

NEI 00-01 Section 3 Guidance

Based on a review of the systems, equipment and cables within each fire area, determine which shutdown paths are either unaffected or least impacted by a postulated fire within the fire area. Typically, the safe shutdown path with the least number of cables and equipment in the fire area would be selected as the required safe shutdown path. Consider the circuit failure criteria and the possible mitigating strategies, however, in selecting the required safe shutdown path in a particular fire area. Review support systems as a part of this assessment since their availability will be important to the ability to achieve and maintain safe shutdown. For example, impacts to the electric power distribution system for a particular safe shutdown path could present a major impediment to using a particular path for safe shutdown. By identifying this early in the assessment process, an unnecessary amount of time is not spent assessing impacts to the frontline systems that will require this power to support their operation.

Based on an assessment as described above, designate the required safe shutdown path(s) for the fire area. Identify all equipment not in the safe shutdown path whose spurious operation or mal-operation could affect the shutdown function. Include these cables in the shutdown function list. For each of the safe shutdown cables (located in the fire area) that are part of the required safe shutdown path in the fire area, perform an evaluation to determine the impact of a fire-induced cable failure on the corresponding safe shutdown equipment and, ultimately, on the required safe shutdown path.

When evaluating the safe shutdown mode for a particular piece of equipment, it is important to consider the equipment's position for the specific safe shutdown scenario for the full duration of the shutdown scenario. It is possible for a piece of equipment to be in two different states depending on the shutdown scenario or the stage of shutdown within a particular shutdown scenario. Document information related to the normal and shutdown positions of equipment on the safe shutdown equipment list.

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

EDISON/SAFE provides a demonstration that at least one post-fire success path is established including the documentation of any engineering justification, recovery actions, etc., needed to ensure the path is available. EDISON/SAFE resolutions (masks) are provided in the form of a protection remark, or compliance strategy, used to disposition the failure of a specific component or cable during the performance of the separation analysis for a specific fire area. Resolutions are only applied to components and cables associated with the systems selected to be utilized to achieve a safe and stable condition. The NFPA 805 systems selected to achieve this condition are those least affected (if at all) by the fire thus minimizing the number of variances that require resolution.

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

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

Fire Area compliance strategies for each fire area are contained within Appendix F of Calculation NEDC 11-019. The compliance strategies include resolutions, recovery actions and credited system success paths for each fire area.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 11.1, 12.1.2 and Appendix F)

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

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.2.3 Determine Safe Shutdown Equipment Impacts	Using the circuit analysis and evaluation criteria contained in Section 3.5 of this document, determine the equipment that can impact safe shutdown and that can potentially be impacted by a fire in the fire area, and what those possible impacts are.

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

The EDISON/SAFE database electronically identifies the equipment that is affected by a fire within a specified fire area. This is completed through logical relationships that have been created within the database. NSCA model development is detailed within Section 9.0 of Calculation NEDC 11-019.

The NSCA analysis is run in EDISON/SAFE, which indicates possible instances where the performance goals are not met. Each of these instances is reviewed. If there is a design reason that the instance is not a problem, e.g. the battery charger fails but the capacity of the battery is sufficient to perform the required action, the explanation(s) are entered into EDISON/SAFE as resolutions and the analysis rerun.
Remaining instances where performance goals are not met once EDISON/SAFE is rerun are considered "Variances from Deterministic Requirements" (VFDRs).
During the NFPA 805 transition process, instances of logic failure were handled using Fire Risk Evaluations (FREs).
During the FRE process, the logic failure is reviewed using risk-informed, performance-based methodologies with the possible results including:
-No impact, risk insignificant;
-Procedure change, preemptive action;
-Physical plant modification; and
-Recovery action (RA).
The results of the Fire Risk Evaluation are reflected in EDISON/SAFE as resolutions. This process is repeated until all the performance goals in EDISON/SAFE are shown to be successful.

Fire Area compliance strategies for each fire area are contained within Appendix F of Calculation NEDC 11-019. The compliance strategies include resolutions, recovery actions and credited system success paths for each fire area.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Sections 9.0, 12.1.2 and Appendix F)

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

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref

3.4.2.4 Develop a Compliance Strategy or Disposition to Mitigate the Effects Due to Fire Damage to Each Required Component or Cable

NEI 00-01 Section 3 Guidance

The available deterministic methods for mitigating the effects of circuit failures are summarized as follows (see Figure 1-2):

- Provide a qualified 3-fire rated barrier.
- Provide a 1-hour fire rated barrier with automatic suppression and detection.
- Provide separation of 20 feet or greater with automatic suppression and detection and demonstrate that there are no intervening combustibles within the 20 foot separation distance.
- Reroute or relocate the circuit/equipment, or perform other modifications to resolve vulnerability.
- Provide a procedural action in accordance with regulatory requirements.
- Perform a cold shutdown repair in accordance with regulatory requirements.
- Identify other equipment not affected by the fire capable of performing the same safe shutdown function.
- Develop exemptions, deviations, Generic Letter 86-10 evaluation or fire protection design change evaluations with a licensing change process.

Additional options are available for non-inerted containments as described in 10 CFR 50 Appendix R section III.G.2.d, e and f.

[Refer to hardcopy of NEI 00-01 for Figure]

Applicability

Applicable

Comments

None

Alignment Statement

Aligns

Alignment Basis

The NSCA methodology is contained within Section 1.3 of Calculation NEDC 11-019. The NFPA 805 analysis is completed using a deterministic or performance based approach in each fire area at CNS. Circuit failures are analyzed to determine if any mitigating factors can resolve the identified cable failures. Failures that cannot be resolved without the use of a recovery action (i.e. separation issues) are evaluated as a VFDR. VFDRs are analyzed within the PRA analysis to determine acceptability. If not acceptable, alternate means of compliance are pursued.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 1.3)

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

NFPA 805 Section: 2.4.2.4 Fire Area Assessment

NEI 00-01 Ref	NEI 00-01 Section 3 Guidance
3.4.2.5 Document the Compliance Strategy or Disposition Determined to Mitigate the Effects Due to Fire Damage to Each Required Component or Cable	Assign compliance strategy statements or codes to components or cables to identify the justification or mitigating actions proposed for achieving safe shutdown. The justification should address the cumulative effect of the actions relied upon by the licensee to mitigate a fire in the area. Provide each piece of safe shutdown equipment, equipment not in the path whose spurious operation or mal-operation could affect safe shutdown, and/or cable for the required safe shutdown path with a specific compliance strategy or disposition. Refer to Attachment 6 for an example of a Fire Area Assessment Report documenting each cable disposition. [Refer to hardcopy of NEI 00-01 for Attachment]

Applicability	Comments
Applicable	None

Alignment Statement

Aligns

Alignment Basis

Per Calculation NEDC 11-019, Section 12.0, the analysis is performed to determine the ability to achieve a safe and stable condition given a fire in any area of the plant. The EDISON/SAFE module provides the capability to perform an automated analysis utilizing a model composed of plant systems, equipment, cables, and their physical locations. The analysis uses a Boolean logic evaluation method that supports success path relationships enabling analysis and graphic display capability of each fire zone/fire area. Consequently, EDISON/SAFE contains the complete CNS NSCA analysis including the equipment, cable, fire zone/area, logic, analysis results and strategy information necessary to support NFPA 805 compliance.

The compliance strategy to achieve a safe and stable condition for each fire area is contained within Appendix F of Calculation NEDC 11-019.

Reference Documents

Calculation NEDC 11-019, "Nuclear Safety Capability Assessment," Revision 0 - (Section 12.0 and Appendix F)

ATTACHMENT C

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

236 Pages

The nuclear safety goal of NFPA 805 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. A safe and stable condition is defined as the ability to maintain *Keff* <0.99, with a reactor coolant temperature at or below the requirements for hot shutdown. The B-3 table documents this for 'at-power' modes of operation.

To meet this nuclear safety goal, fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition and the licensee must demonstrate the ability to maintain one success path of required equipment effectively free of fire damage. This was accomplished at CNS by developing and analyzing a comprehensive list of systems and equipment to identify those critical components required to achieve and maintain the fuel in a safe and stable hot shutdown state following a fire from at power conditions. Key assumptions associated with the Table B-3 selected equipment and systems are detailed within CNS Calculation NEDC 11-019.

In general, the fire area analyses consider control of the unit from the Control Room or from the Alternate Shutdown (ASD) Room (i.e., Primary Control Station).

The strategies documented in Table B-3 are the assured strategy for each fire area. The location and severity of the fire dictates whether the bounding (i.e., worst case full area involved in the fire) compliance strategy will be implemented for a fire.

Evaluation of Nuclear Safety Performance Criteria

NFPA 805 Section 1.5.1 states, "Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met." The sections below provide a description of how the performance criteria are assured at CNS.

Reactivity Control

NFPA 805 Section 1.5.1(a) states for the Reactivity Control performance criteria:

> *Reactivity control shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting shall occur rapidly enough such that fuel design limits are not exceeded.*

Reactivity control will be accomplished by insertion of the control rods and will result from an automatic RPS trip or from operator initiation of a manual trip. This action will de-energize the RPS to actuate a reactor scram. For boiling water reactors (BWRs), adequate shutdown margin is assured without the need for borated charging water and source range monitoring is not required.

Inventory and Pressure Control

NFPA 805 Section 1.5.1(b) states for the Inventory and Pressure Control performance criteria:

> *With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling coolant level such that subcooling is maintained for a PWR and shall be capable of maintaining or rapidly restoring reactor water level above top of active fuel for a BWR such that fuel clad damage as a result of a fire is prevented.*

Reactor coolant make-up can be achieved by isolation of the Reactor Coolant System and control of vessel coolant level by injecting water into the isolated Reactor Pressure Vessel (RPV). Systems for which the primary purpose is to inject water into the RPV are grouped under this Performance Goal, even if they also assist in removing decay heat.

Maintaining the RCS pressure boundary integrity is necessary to achieve inventory and pressure control. Coolant loss is limited by modeling the ability to isolate the following pressure boundaries:

- Control Rod Drive (CRD) vents
- High Pressure Coolant Injection (HPCI) System
- Main Steam to the Main Turbine and other paths
- Safety Relief Valves (SRVs - spurious concern)
- Steam Supply to Reactor Core Isolation Cooling (RCIC) Turbine
- Residual Heat Removal (RHR) System
- Reactor Water Cleanup (RWCU) System

Initially, reactor makeup is provided by operation of the Core Spray (CS) system, the HPCI system or the RCIC system. The Operator or the Automatic Depressurization System (ADS) is utilized to reduce plant pressure for use of the low pressure CS System. The HPCI and RCIC systems consist of a steam driven centrifugal pump capable of injecting high pressure water into the RPV. This also aids in removing decay heat.

Overpressure protection is provided by the SRVs in the self-activated spring lift mode. This mode of operation is not susceptible to fire damage.

Instrumentation required for Reactor Inventory and Pressure Control (Reactor Coolant Level and Pressure) is modeled in the Process Monitoring Performance Goal.

When shutting down from outside the Control Room, Inventory and Pressure control is accomplished by maintaining the RCS pressure boundary integrity and by operating HPCI from the Alternate Shutdown (ASD) Room.

Decay Heat Removal

NFPA 805 Section 1.5.1(c) states for the Decay Heat Removal performance criteria:

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

Decay Heat is removed initially by natural circulation within the RPV and automatic (mechanical) operation of the SRVs. The SRVs discharge steam from the RPV to the Suppression Pool. The emerging steam is condensed in this pool, and the heat absorbed by the Suppression Pool is removed by the RHR system operating in the Suppression Pool Cooling (SPC) mode and ultimately transferred to the river via the Service Water (SW) system. The required system logics for SPC include satisfying net positive suction head requirements.

When HPCI or RCIC is used for makeup, steam is also vented to the suppression chamber via the operating turbine. These systems are modeled under the Inventory and Pressure Performance Goal since their primary purpose is makeup, and they are not included in this Performance Goal.

Instrumentation required for decay heat removal (suppression chamber level and temperature) are modeled in the Process Monitoring Performance Goal.

When steam pressure is reduced such that HPCI and/or RCIC turbine operation cannot be sustained, the SRVs are manually opened to further reduce RPV pressure such that the CS system can be used to provide core cooling and maintain RPV inventory.

When shutting down from outside the Control Room, decay heat removal is accomplished by operating the RHR system in Low Pressure Coolant Injection (LPCI) mode from the ASD Room. Three SRVs are available to be operated from the ASD Room to depressurize the RPV.

Fuel pool cooling is required in order to prevent boiling and the resulting loss of inventory, which can cause damage to the stored fuel cells when they are uncovered. However, per CNS Updated Safety Analysis Report Section X-3, Spent Fuel Storage:

> *Since the spent fuel pool temperature will initially be less than 150°F, the decay heat will take at least 4 hours to heat the spent fuel pool water to 212°F. Four hours is sufficient time to establish adequate makeup to the spent fuel pool prior to the onset of bulk boiling. Also, under bulk boiling conditions, the temperature of the fuel will not exceed 350°F. This is an acceptable temperature from the standpoint of fuel rod integrity and surface corrosion.*

Plant procedures require logging the temperature every four hours in the event that cooling is lost. A number of options are available for replenishing the water to prevent uncovering the fuel, including the use of fire hoses or cross-ties to the RHR system. It is, therefore, unnecessary to model the Fuel Pool Cooling system.

Vital Auxiliaries

NFPA 805 Section 1.5.1(d) states for the Vital Auxiliaries performance criteria:

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

Various plant systems are required to support the systems and components selected to accomplish the previously defined safety functions. Not all of these plant systems are modeled as system logics but are instead modeled in component logics for the supported component.

The following are modeled as Systems in the SUPPORT Performance Goal:

- Reactor Equipment Cooling (REC): This system supplies cooling to critical heat exchangers and coolers. If both pumps are inoperable, emergency operation is possible by cross-connecting to the SW system. This allows modeling Train A and Train B of both the normal and the emergency lineup as four different systems and at least one of the four are required to survive.

- Service Water: This system cools the diesel generator coolers and the REC and RHR Heat Exchangers. SAFE models Train A and Train B as two systems, and requires the same-train SW to cool the REC and RHR systems. Both Diesel Generator coolers can be supplied from either SW Train A or Train B. Components of the SW system that are specific to individual cooling loads are modeled as support components for the cooled system.

Other critical support functions are modeled in as component logics:

- HVAC Systems. HVAC cooling is required and modeled for the Battery Rooms, Diesel Generators, Critical Switchgear Rooms, Core Spray Pump Rooms (including the RCIC turbine) and the HPCI Room using component logics. Plant evaluations have confirmed that other areas of the plant do not require HVAC cooling in order to protect credited equipment from overheating or to ensure habitability.

- Diesel Generator Support Auxiliaries. Provides direct support in component logics for the corresponding Diesel Generator. This includes cooling water valves, ventilation, starting air, fuel oil transfer pumps and jacket water pumps.

- Electrical Supply. Each bus, breaker, motor control center, panel, etc., is modeled in direct support of the specific load.

When shutting down from outside the Control Room, REC Pumps C and D can be operated from the ASD Room to provide cooling, or isolated to allow for the REC emergency mode.

Process Monitoring

NFPA 805 Section 1.5.1(e) states for the Process Monitoring performance criteria:

> *Process monitoring shall be capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have been achieved and are being maintained.*

NEI 00-01 contains guidance in Section 3.1.2.5 for choosing appropriate instrumentation for process monitoring. The following instruments are modeled in the Reactor Vessel Instrumentation (RVI) system, based on existing operating procedures:

- Reactor coolant level and pressure
- Suppression chamber temperature and level

The indicating ranges of these instruments cover the normal operating bands and will operate throughout the scenario. Other tank levels and diagnostic instrumentation such as flow or system pressures are modeled directly in the logics for the system or component that requires the indication and are not included in this performance goal.

In addition to instruments required for indication, the analysis includes instruments which provide permissive or controlling signals to safe shutdown components or which can cause spurious operation. These instruments are modeled in direct support of the affected component using component logics, cable logics, or a combination of both.

When shutting down from outside the Control Room, the following instruments are available in the ASD Room:

- Reactor coolant level
- Reactor Pressure (at HPCI Turbine Steam Inlet)
- Suppression chamber temperature and level
- Emergency Condensate Storage Tank level
- Diagnostic instrumentation for alternate shutdown systems
- Level indication for tanks needed for alternate shutdown

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

Fire Zone	Description
7A	RHR Service Water Booster Pump and Service Air Compressor Areas
7B	Emergency Condensate Storage Tank Area
8C	RPS Room 1A
8D	Seal Water Pump Area and Corridor

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train B will be operated to maintain Suppression Pool temperature.	CBA-02 CBA-05
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indication [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train B to maintain RPV level.	CBA-02 CBA-05
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A	
Vital Auxiliaries	Mechanical: -REC will be supplied by SW Train B to provide the cooling supply to the ECCS. -SW Train B will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160V Bus 1G - 125/250 VDC Train B is available HVAC: - CS Train B - Quad area cooling - AC Switchgear Room 1G - Essential Control Building HVAC system - DC Switchgear Room 1B - Essential Control Building HVAC system - Battery Rooms 1B - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1B - Essential Control Building HVAC system	CBA-01 CBA-03 CBA-04 CBA-06

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-032 - Evaluation of DC SWGR Rooms 1A and 1B and Battery Rooms 1A and 1B Fire Barrier Separation

Purpose This evaluation was written to justify fire dampers provided in supply ventilation ductwork routed through the barrier between each Switchgear Room and the adjacent Seal Water Pump Area and Corridor that have not been installed in the plane of the barrier, as required by the manufacturers installation guidelines. Also, the dampers at the wall boundary where the ventilation ductwork penetrates the barriers from Battery Rooms 1A and 1B to the Seal Water Pump Area and Corridor have been blocked open to ensure exhaust system operability. In addition, the fire doors in these barriers have NFPA 80 code deviations.

Conclusion Based on the lack of significant combustible loading and fire hazards in the areas, the presence of 3-hour rated dampers that will provide significant fire separation regardless of the mounting position, and installed fire protection features including detection system coverage and manual suppression capabilities, the fire damper configurations are adequate for the fire hazards of the adjacent areas. The fire door code deviations include excessive door-to-door gap clearances and modifications made to ensure proper door operation. Considering the lack of significant combustible loading and fire hazards in the area and administrative procedures controlling door operation, these deviations do not represent a significant decrease in fire safety.

Basis The bases that justify this conclusion are summarized as follows:

• The combustible loading classification for Switchgear Rooms 1A and 1B are "LOW" and is mainly attributed to miscellaneous plastics and transient loading allowances. There are few fixed combustibles in any of the zones and the equivalent fire severities are minimal. The combustible loading classification in the Corridor is also "LOW" and consists of miscellaneous cable, hose and plastic, and transient allowances.
• The lack of significant combustible materials in the zones and the lack of intervening combustibles significantly reduce the chance of fire propagation between zones. Transient combustibles are controlled by plant procedures, effectively reducing the possibility of a fire involving transient materials.
• Each fire zone is provided with detection system coverage. In the event of a fire in the Switchgear Rooms or Corridor, detection system actuation will result in rapid fire brigade response and subsequent manual extinguishment utilizing hose stations and portable extinguishers strategically located in adjacent fire zones.
• Based on the material and installation of the ventilation ducts, they are considered to prevent fire propagation for up to 1 hour.

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire hazards of the areas, and adequate separation has been provided. In general, minor variations to the configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.

Basis

The bases that justify this conclusion are summarized as follows:

• Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
• Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade.
• Ventilation systems can typically be used for smoke and heat removal.
• Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
• The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202.
• The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105.
• The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201.
• The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307.
• The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104.
• The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103.
• The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104.
• The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6.
• The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited for the acceptability of door R7.
• The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for the acceptability of doors R101 and R102.
• The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102.

- The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115.
- Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.

Table B-3 Fire Area Transition

Fire Area		**Description**
CB-A		Control Building Basement, Control Building 903 Corridor and RPS Room 1A
EEEE Title		EE 09-040 - Evaluation of Auxiliary Relay Room and RPS Room 1B Appendix R Fire Barriers
	Purpose	This evaluation justifies the adequacy of ventilation duct penetrations, without fire-rated dampers, routed through the barriers that separate the Auxiliary Relay Room, RPS Room 1B, and RPS Room 1A on the 903'-6" Elevation of the Control Building. Additionally, a fire rating cannot be assigned to the barriers due to miscellaneous door discrepancies concerning doors H102, H103, and H104. • Door H102 separates RPS Room 1B (Fire Area CB-C) from the Seal Water Pump Area and Corridor (Fire Area CB-A) • Doors H103 and H104 separate the Auxiliary Relay Room (Fire Area CB-D) from the Seal Water Pump Area and Corridor (Fire Area CB-A)
	Conclusion	Based on the installed fire protection features, including detection system coverage and manual fire suppression capabilities, and the lack of significant fire hazards, and the types of combustible materials, the ventilation ductwork is sufficient to provide protection from spread of fire prior to fire brigade response and subsequent manual extinguishment, if necessary. Discrepancies concerning doors H102, H103, and H104 are relatively minor, and include door clearances, and minor modification for proper operation. These discrepancies are therefore not considered to affect fire safety. The separation that has been provided is considered to be adequate for the fire hazards of the areas.
	Basis	The bases that justify this conclusion are summarized as follows: • There are typically few fixed combustibles in any of the zones, and the equivalent fire severities are minimal. The exception is the Auxiliary Relay Room, which contains a significant amount of cable insulation in open cable trays routed at the ceiling elevation. The content of the RPS Room consists mainly of batteries. The combustibles in the Corridor mainly consist of miscellaneous cable, hose and plastic, and transient allowances. Cables in the Corridor are routed in conduit, and potential ignition sources in the zones are limited to energized electrical equipment, typically in cabinets, and potential transient sources. • A cable tray fire in the Auxiliary Relay Room would be characterized as a slow propagating fire that has the potential to produce significant amounts of smoke. Based on the installed detection system coverage, fire brigade response and manual extinguishment would occur prior to the fire being able to breach the barrier via the ductwork. • The lack of significant combustible materials in the remaining zones and the lack of intervening combustibles significantly reduce the chance of fire propagation between zones. • Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of a fire.

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

• The supply registers in the Auxiliary Relay Room are provided with 3-hour rated fire dampers (HV-AD-AD1434 and HV-AD-AD1435). A 3-hour fire-rated damper has been provided in the Auxiliary Relay Room at the exhaust register (HV-AD-AD1436).
• Sheet metal ductwork is recognized by the NFPA to provide up to 1-hour fire separation when properly hung, and the fire is stopped.
• Each fire zone is provided with detection system coverage. In the event of a fire in the RPS Room, Auxiliary Relay Room, or Corridor, detection system actuation will result in rapid fire brigade response and subsequent manual extinguishment utilizing hose stations and portable extinguishers strategically located in adjacent fire zones.

EEEE Title EE 09-042 - Evaluation of 1-Hour Marinite Wall in Battery Room 1B

Purpose This evaluation is written to address the fire protection adequacy of the 1-hour fire-rated wall enclosure located in Battery Room 1B.

Conclusion Based on a review of the potential fire hazards located in Battery Room 1B, with the 1-hour fire-rated wall enclosure and the existing fire detection system, the existing barrier provided is adequate to prevent the fire spreading from Battery Room 1B to the 1-hour fire-rated enclosure.

Basis The bases that justify this conclusion are summarized as follows:

• The combustible loading classification for Battery Room 1B is "LOW" and is mainly attributed to plastics and transient loading allowances. Battery Room 1B has an equivalent fire severity of 30 minutes. The combustible loading consists of cables routed in conduit riser. The lack of significant combustible materials in the zones and the lack of intervening combustibles significantly reduce the chance of fire propagation between zones.
• Based on the low combustible loading in Battery Room 1B, a fire will not be able to develop to such intensity as to challenge the as-installed configuration.
• Battery Room 1B is provided with smoke detection system coverage. Detection system actuation will prompt rapid fire brigade response and manual extinguishment via portable extinguishers and manual hose stations strategically located in adjacent zones. Pre-fire plans are available for Battery Room 1B.
• Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of a fire.
• Safe shutdown can be accomplished independent of Battery Room 1B.

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

EEEE Title

LBDCR 2004-023 - Evaluation of a FHA Revision to Relocate the Fire Barrier Between Fire Area IV/Fire Zone 8D and Fire Area VII/Fire Zone 24

Purpose

The purpose of this evaluation is to demonstrate the relocated fire barrier between Fire Area IV/Fire Zone 8D and Fire Area VIII/Fire Zone 24 is adequate. The section of the barrier that is being changed is the wall area adjacent to, and including, door H100. The barrier is being moved to the vestibule walls and ceiling on the north side of H100. Door H100 will no longer be considered part of the fire barrier. Door H114 on the east side of the vestibule will be evaluated to show it is adequate for the fire hazards that are present. Door H114 is located on the east side of the vestibule. The door is a single leaf unrated metal door. The door has been barricaded with a 1/2 in. thick carbon steel plate plug welded to the frame of the door. The west side of door H114 has been barricaded with concrete blocks stacked approximately 6 ft high and 3 ft deep.

Conclusion

The walls and ceiling of the vestibule can be credited as a 3-hour fire rating. Based on the fire hazard analysis, fire door H114 is adequate for the hazards associated with the area.

Basis

The bases that justify this conclusion are summarized as follows:

- The automatic smoke detection system that alarms in the Control Room will result in prompt fire brigade response and manual fire brigade extinguishment. Based on low combustible loading and detection in the area, the fire zone boundaries are adequate to prevent fire spread to adjacent fire areas and fire zones. Pre-fire plans are available for fire brigade use in responding to fire events in the fires zones.
- An automatic wet pipe sprinkler system has been provided in the Swing Charger Room adjacent to the west wall of Fire Zone 8C.
- Given a fire in this zone, safe shutdown can be accomplished as verified by the Appendix R Safe Shutdown Analysis Report.
- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- The construction features associated with door H114 and the vestibule are similar to door R115. Door R115 was evaluated under EE 09-035.
- Based on the construction of the barrier and door H114 with welded steel plate and the fire severity present in Fire Area VIII/Fire Zone 24 side of the barrier, the Fire Area VIII/Fire Zone 24 fire would not be expected to breach the barrier. Also, a fire starting in Fire Area VIII/Fire Zone 24 would be mitigated by the automatic suppression system so that it would not provide a significant challenge to barrier. A fire starting in Fire Area IV/Fire Zone 8D would not be expected to spread to Fire Area VIII/Fire Zone 24 due to lack of combustibles in the area, lack of combustibles inside the vestibule, and the construction features of door H114.
- The walls and ceiling of the vestibule can be credited as a 3-hour fire rating. Based on the fire hazard analysis, fire door H114 is adequate for the hazards associated with the area.

Table B-3 Fire Area Transition

Fire Area	**Description**
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A
EEEE Title	EE 97-121 - Appendix R Fire Protection Evaluation of Control Building 882' Underground Cable Manholes

Purpose

This analysis is written to address the adequacy of cable manhole barriers located on the basement level of the Control Building, 882'-6" Elevation (Fire Area IV/Fire Zone 7A). Non-rated fire barriers separate the Control Building Basement Area from the individual cable manholes located below the basement, which function as a connection point for cables routed in underground ducts from the Intake Structure or the Diesel Generator Building to the Control Building. Due to the presence of safe shutdown cables in the manholes, the adequacy of the manhole barriers must be assessed to ensure that fire spread is precluded to the Control Building Basement at the 882' Elevation and to the opposite division manholes.

Conclusion

Based on a review of the potential fire hazards located in the Control Building Basement and in the manholes, adequate fire protection is provided to prevent fire spread to and from the manholes located below the Control Building Basement..

Basis

The bases that justify this conclusion are summarized as follows:

- Each manhole contains a limited amount of fire resistant cable insulation and no ignition sources. As a result, fire potential in the manhole is limited, a significant fire could not be sustained, and due to the lack of intervening combustibles and the metal plates, propagation to either the Control Building Basement Area or to the redundant division manholes is not possible.
- A review of the Control Building Basement Area also indicates limited potential for fire spread into the manholes. Combustible loading in the basement area is low and consists mainly of lube oil associated with the RHR Service Water (RHRSW) Booster Pumps and Station Air Compressors.
- A 4" steel I-beam floor dike is also provided on the floor in the basement area to separate the main basement floor area from the floor area of the manholes. This floor dike will prevent combustible liquid spills from impacting the manholes.
- Smoke detection, portable extinguishers, and a manual fire hose are provided in the area.
- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- Each manhole is approximately 8' apart on center. A substantial metal plate (approximately 1/4" thick and approximately 3' x 3') covers the access opening to each manhole. Cables routed through the manhole do not penetrate or cross over to the redundant division manholes. Cables exiting manholes are typically routed in 4" steel conduits and the conduits are routed in the soil fill below the basement slab.

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

Variances from Deterministic Requirements (VFDR)

CBA-01

Description

Establish vital auxiliaries by powering the credited 4160G Bus from Emergency Transformer (EE-CB-4160G-1GE).

The Nuclear Safety Performance Criteria (NSPC) require at least one train of electrical power available post-fire. There is cable fire damage to the DG Fuel Oil (DGDO) Pumps, EE-CB-4160DG2-EG2 (DG48, DG53, DG54, DG57, H574), EE-CB-4160F-1FS, EE-CB-4160G-1GE (H572, H573), EE-CB-4160G-1GS (H555), EE-SWGR-4160DG2 and Isolate-DG2 (A54, DG45, DG48, DG50, DG51, DG63, DG87, DG88, DG89, H470, H485, H573, H574). The 4160F-1FS, even with the potential cable damage, will trip to protect the Emergency Transformer from potential faults on the 4160F Bus and allow the Emergency Transformer to be used in this area. Cable damage to the EE-CB-4160DG2-EG2, EESWGR- 4160DG2, and Isolate-DG2 would allow for the DG to place itself on the 4160G Bus, potentially out-of-phase with offsite power that may be present on the bus. Based on cable damage, even in "isolate" the failure mode is possible to place the DG on the bus until control power fuses are pulled in the DG2 area and the breaker controlled locally (specifically DG53 and DG57 with an external hot short).

During the normal transfer sequence when 4160B is removed from the 4160G Bus (EE-CB-4160G-1GB and/or EE-CB-4160B-1BG open), DG2 would start automatically and the EE-CB-4160G-1GS would close. Once DG2 is up to speed, it checks the status of the 4160G Bus and closes its output breaker if power is still required to the bus. The operator can open EE-CB-4160G-1GS and/or EE-CB-4160G-1GE from the Control Room, but based on normal operations for the GS breaker (if there is no power on the bus) and/or cable damage on the GE breaker (H572 and H573), either breaker may re-close on its own once the operator has opened it from the Control Room.

Action to open EE-CB-4160G-1GE locally may not be able to be accomplished in time to preclude paralleling the DG out-of-phase onto the credited 4160G Bus. The four cables in question are all affected in fire scenarios: 7A-TS11, CPSR-A.FD4, CPSR-B.FD4, CPSR-C.FD4, SW-P-A.FD4, SW-P-B.FD4, SW-P-C.FD4, and SW-P-D.FD4.

Fire damage to interlock cables (H572 and H573) between 1GE and EG2/GEN EG2 CONTROL and RELAY PANEL may also cause automatic tripping of the 4160G-1GE breaker and inability to close it from the Control Room, or allow for auto-closure of breaker. EE-CB-4160G-1GE is normally closed and is desired open to isolate the DG from the 4160G Bus if the Emergency Transformer is used. The

Table B-3 Fire Area Transition

Fire Area		Description
CB-A		Control Building Basement, Control Building 903 Corridor and RPS Room 1A
		Emergency Transformer is available. The ability to open from the Control Room is still available, but based on potential cable damage the 4160-1GE breaker may re-close on its own. This is a separation issue for Vital Auxiliaries.
	Disposition	A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria. Risk: Acceptable with Recovery Action. See LAR Attachment G, Table G-1 for Recovery Action details. Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

CBA-02

Description

Preventing a full or partial loss of Service Water for supporting Decay Heat Removal and Inventory and Pressure Control. (EE-CB-4160G-SWP1B, SW-MOV-37MV, and SW-MOV-MO89B).

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. CS Train B is credited for Inventory Control, SPC Train B is credited for Decay Heat Removal, REC Train B provides cooling for RHR Pump 1D and the RHR and CS Pump Rooms (Quads), and SW Train B provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The complete loss of Service Water or its diversion could challenge these NSPC.

Fire in the area could result in damage to the 4KV Bus 1G UV circuit which would cause a trip signal to the SW Pump 1B breaker. Inability to keep the SW Pump 1B breaker closed would result in loss of all Service Water.

Damage to SW-MOV-37MV cables (MY348 and MY350) could remove the ability to close the valve if already open. This would divert flow to non-critical loads and the non-credited SW loop with only a single Service Water Pump available. This reduction in flow may not allow sufficient cooling to the REC and RHR Heat Exchangers.

NOTE: Even though SW-MOV-37MV is a potential NRC Information Notice 92-18 concern, the only location with an affected cable is in Fire Zone 7A, where the cable that causes the concern is protected via the duct bank.

Cable damage to both RHRSW Booster Pump breakers would result in the inability to close either breaker, causing SW-MOV-MO89B to fail closed without the ability to open from the Control Room. This would result in the loss of SW flow to the RHR Heat Exchanger for SPC Train B.

This is a separation issue for Decay Heat Removal and Inventory and Pressure Control.

Table B-3 Fire Area Transition

Fire Area		Description
CB-A		Control Building Basement, Control Building 903 Corridor and RPS Room 1A
	Disposition	A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria. Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth. See LAR Attachment G, Table G-1 for Recovery Action details. Modification: None
CBA-03		
	Description	Loss of Critical Switchgear cooling due to cable damage to damper AD-1408 (HV-FAN-SF-SWGR-1G and HV-FAN-EF-SWGR-1G). Fire damage to cables will not preclude operation of the HV-FAN-EF-SWGR-1G and HV-FAN-SFSWGR-1G fans from the 1G AC Switchgear Room. Cable damage will affect the operation of ventilation damper AD-1408 by energizing its solenoid and keeping the damper open. Damper AD-1407 has its cable routed in dedicated conduit in area CB-A. The NSCA model requires that either train of Switchgear Room cooling fans be available to ensure both AC and DC Switchgear remain available post-fire. This is a separation issue for Vital Auxiliaries, Electrical Power Distribution.
	Disposition	A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria. Risk: Acceptable Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

CBA-04

Description

4160G UV Circuit cable damage affects automatic / remote operation from the Control Room of Critical Pumps for Safe Shutdown (CS Pump 1B and RHR Pump 1D).

Fire in the area could result in damage to the 4KV Bus 1G UV circuit that will cause a trip signal to the CS Pump 1B, RHR Pump 1D, and SW Pump 1B breakers. SW Pump 1B is covered separately under CBA-2 for ensuring SW flow. NSPC requires the ability to maintain level within the core. CS Train B is the credited train for inventory control in this area. RHR Train B is credited to support SPC mode of RHR following a fire in this area. NSPC requires a means of Decay Heat Removal post-fire. SPC Train B is credited for Decay Heat Removal.

Inability to keep the Core Spray Pump 1B breaker closed would result in loss of ability to add makeup water via the CS system.

Inability to keep the RHR Pump 1D breaker closed would result in loss of RHR flow to the RHR Heat Exchanger, resulting in the loss of SPC Train B.

This is a separation issue for Vital Auxiliaries, Electrical Power Distribution, in support of Inventory and Pressure Control, and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

CBA-05

Description

Preventing a full or partial loss of Service Water due to clogging of SW-STNR-B.

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. CS Train B is credited for Inventory Control and SPC Train B is credited for Decay Heat Removal. REC Train B provides cooling for RHR Pump 1D and the RHR and CS Pump Rooms (Quads), and SW Train B provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The complete loss of Service Water or its diversion could challenge these NSPC.

Cable damage to MTX1 results in a loss of the automatic features of the SW Train B strainer. The loss of the automatic features could result in the clogging of the strainer, resulting is a complete loss of SW flow.

This is a separation issue for Inventory and Pressure Control and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

CBA-06

Description

Loss of both A and B Train Battery Chargers due to Cable damage (EE-CHG-125-1B).

Fire damage to cable MTX10 causes a loss of Main Supply Power to the B Train Battery Charger. The NSPC require at least one train of DC power be available post-fire and both A and B Train Battery Chargers are affected. The B Train batteries are the credited train post-fire.

This is a separation issue for Vital Auxiliaries.

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

Disposition A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Modification.

Modification: Item S-2.3 of LAR Attachment S, Table S-2.

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
7A	Detection	Ionization	R	N	N	N	Y	Y	N
7A	Feature	Fire Barrier	N/A	N/A	N	N	N	Y	N
7B	Detection	Ionization	R	N	N	N	N	N	N
8C	Detection	Ionization	R	N	N	N	Y	Y	N
8D	Detection	Ionization	R	N	N	N	Y	Y	N
8D	Suppression	Automatic Wet-Pipe	P	N/A	N	N	N	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	Description
CB-A	Control Building Basement, Control Building 903 Corridor and RPS Room 1A

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. In the event of normal operation, the wet-pipe system in the Swing Charger Room will not have an adverse effect, as the chargers and the disconnect switch cabinets are spray tight and the conduits are sealed. A pipe rupture from within Fire Zone 7A could spray several of the RHRSW Booster Pumps and may also spray EE-MCC-T. However, the equipment is spray-tight and would not be adversely affected. A pipe rupture of the supply pipe in the corridor of Fire Zone 8D could discharge water through the steel equipment access hatch and down the stairwell to Fire Zone 7A. There is no direct effect due to floor drains, equipment pedestals, and the physical distance to equipment. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

A 1-hour rated concrete enclosure containing Division II control and motive power feeds is located along the south wall of Fire Zone 7A.

Table B-3 Fire Area Transition

Fire Area	Description
CB-A-1	Control Building 903, DC Switchgear Room 1A and Battery Room 1A

Fire Zone	Description
8E	Battery Room 1A
8H	DC Switchgear Room 1A

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train B will be operated to maintain Suppression Pool temperature.	None
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, SW, RHR flow indications, and REC pressure indication [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train B to maintain RPV level.	CBA1-02
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
CB-A-1	Control Building 903, DC Switchgear Room 1A and Battery Room 1A	
Vital Auxiliaries	Mechanical: -REC Train B will be operated to provide the cooling supply to the ECCS. -SW Train B will be operated to provide the cooling supply to the REC system and RHR Train B Heat Exchanger. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160V Bus 1G - 125/250 VDC Train B is available HVAC: - HPCI/CS Train B - Quad area cooling - AC Switchgear Room 1G - Essential Control Building HVAC system. - DC Switchgear Room 1B - Battery Room 1B - Auxiliary Relay Room and RPS MG Set Room 1B	CBA1-01 CBA1-03

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	Description
CB-A-1	Control Building 903, DC Switchgear Room 1A and Battery Room 1A

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-032 - Evaluation of DC SWGR Rooms 1A and 1B and Battery Rooms 1A and 1B Fire Barrier Separation

Purpose This evaluation was written to justify fire dampers provided in supply ventilation ductwork routed through the barrier between each Switchgear Room and the adjacent Seal Water Pump Area and Corridor that have not been installed in the plane of the barrier, as required by the manufacturers installation guidelines. Also, the dampers at the wall boundary where the ventilation ductwork penetrates the barriers from Battery Rooms 1A and 1B to the Seal Water Pump Area and Corridor have been blocked open to ensure exhaust system operability. In addition, the fire doors in these barriers have NFPA 80 code deviations.

Conclusion Based on the lack of significant combustible loading and fire hazards in the areas, the presence of 3-hour rated dampers that will provide significant fire separation regardless of the mounting position, and installed fire protection features including detection system coverage and manual suppression capabilities, the fire damper configurations are adequate for the fire hazards of the adjacent areas. The fire door code deviations include excessive door-to-door gap clearances and modifications made to ensure proper door operation. Considering the lack of significant combustible loading and fire hazards in the area and administrative procedures controlling door operation, these deviations do not represent a significant decrease in fire safety.

Basis The bases that justify this conclusion are summarized as follows:

- The combustible loading classification for Switchgear Rooms 1A and 1B are "LOW" and is mainly attributed to miscellaneous plastics and transient loading allowances. There are few fixed combustibles in any of the zones and the equivalent fire severities are minimal. The combustible loading classification in the Corridor is also "LOW" and consists of miscellaneous cable, hose and plastic, and transient allowances.
- The lack of significant combustible materials in the zones and the lack of intervening combustibles significantly reduce the chance of fire propagation between zones. Transient combustibles are controlled by plant procedures, effectively reducing the possibility of a fire involving transient materials.
- Each fire zone is provided with detection system coverage. In the event of a fire in the Switchgear Rooms or Corridor, detection system actuation will result in rapid fire brigade response and subsequent manual extinguishment utilizing hose stations and portable extinguishers strategically located in adjacent fire zones.
- Based on the material and installation of the ventilation ducts, they are considered to prevent fire propagation for up to 1 hour.

Table B-3 Fire Area Transition

Fire Area		**Description**
CB-A-1		Control Building 903, DC Switchgear Room 1A and Battery Room 1A
EEEE Title		EE 86-2 - Evaluation of a Ventilation Opening Through the Cable Spreading Room Floor Appendix R Fire Barrier
	Purpose	The purpose of this engineering evaluation is to document the acceptability of fire damper HV-AD-AD1556 located in the floor/ceiling boundary where the ventilation ductwork penetrates from Battery Room 1A to the Cable Spreading Room. The damper has been blocked open to ensure exhaust system operability, as accidental closure would render the exhaust system inoperable. In addition, steel pipe has been used as ventilation ductwork in the Cable Spreading Room, which is not typical for ventilation.
	Conclusion	Based on the significant construction of the assembly, the lack of significant fire hazards and combustible loading, and the presence of installed fire protection features including fire detection and fixed fire suppression, the configuration that has been provided is adequate for the fire hazards of the areas.
	Basis	The bases that justify this conclusion are summarized as follows: • The duct assembly in the Cable Spreading Room is of significant steel construction. There is reasonable assurance that if a fire in the Cable Spreading Room was to occur, it would not breach the ductwork (i.e., the pipe) based on the significant pipe construction, and be able to propagate down into the Battery Room. • Although the combustible loading classification in the Cable Spreading Room is "HIGH," due to the large quantity of cable insulation in the area, installed fire protection features are provided to mitigate the effects of a fire. • The combustible loading classification in the Battery Room is "LOW" and consists mainly of plastic associated with battery cases. The lack of significant fire hazards and combustible loading in Battery Room 1A precludes the possibility of a fire developing to such intensity as to breach the ventilation ductwork in the Battery Room and then breach the significant pipe construction provided in the Cable Spreading Room. • Automatic detection system coverage will alert the fire brigade of fire conditions and automatic suppression system coverage will limit the intensity of a fire in the Cable Spreading Room. • Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.

Table B-3 Fire Area Transition

Fire Area	**Description**
CB-A-1	Control Building 903, DC Switchgear Room 1A and Battery Room 1A

Variances from Deterministic Requirements (VFDR)

CBA1-01

Description

Loss of critical electrical components due to loss of Control Building HVAC/Switchgear Cooling (HV-FAN-SF-SWGR-1G, HV-FAN-EF-SWGR-1G, EE-MCC-TX, EE-PNL-CDP1B, EE-SWGR-125B, and EE-BAT-125B).

The NSCA model requires that either train of Switchgear Room cooling fans be available to ensure the Switchgear remains available post-fire. The NSCA model also requires that either train of Battery power be available post-fire.

Fire damage to cables will not preclude operation of the EF-SWGR-1G and SF-SWGR-1G fans from the 1G AC Switchgear Room. Cable damage will affect the operation of ventilation damper AD-1408 by energizing its solenoid and keeping the damper open. Damper AD-1407 has its cable routed in dedicated conduit in area CB-A.

Fire-induced failure of cables for Control Building HVAC and/or a fire in the Train A DC Switchgear and/or Battery Room will result in loss of cooling to EE-SWGR-125B/EE-BAT-125B/EE-PNL-CDP1B/EE-MCC-TX. This would result in a loss of all DC panels. With temperatures affecting operations, cooling via open compartments is effective to ensure operation of the panels.

This is a separation issue for Vital Auxiliaries, Electrical Power Distribution.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
CB-A-1	Control Building 903, DC Switchgear Room 1A and Battery Room 1A

CBA1-02

Description

Preventing a Reactor Recirculation (RR) Pump Seal LOCA due to inability to secure the RR Pumps from the Control Room with the REC Non-Critical Header secured (RR Pump Breaker for 4160C-1CS).

Breaker F/FDR to the 4160V Bus from the Startup Transformer: This is a normally available, required open breaker that provides motive power to RR Pump A. The RR Pumps are required to trip to prevent a potential seal LOCA when REC is not available to provide cooling. This would challenge the NSPC for Inventory Control. Remote operation of the breaker from the Control Room is lost due to EE-PNL-AA1 cable DC311 being potentially damaged, and therefore a loss of control power to the breaker.

NOTE: REC Non-Critical Header is secured in this fire area to address potential containment over-pressure.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

CBA1-03

Description

Establish vital auxiliaries by ensuring power to the credited 4160G Bus from Emergency Transformer (EE-CB-4160F-1FS).

The NSPC require at least one train of electrical power available post-fire. Fire damage to control power panels and supply cables on the 4160F Bus would not allow breakers to open automatically if required, and therefore, cause a loss of the credited 4160G Bus. EE-CB-4160F-1FS would normally close during the transfer process to the Emergency Transformer to power the 4160F Bus.

This is a separation issue for Vital Auxiliaries.

Table B-3 Fire Area Transition

Fire Area	Description
CB-A-1	Control Building 903, DC Switchgear Room 1A and Battery Room 1A

Disposition A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
8E	Detection	Ionization	R	N	N	N	Y	Y	N
8H	Detection	Ionization	R	N	N	N	Y	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. There are no fixed suppression systems in the area and the Battery Rooms and DC Switchgear Rooms on the 903'-6" Elevation of the Control Building are not subject to any adverse effects by water intrusion from fire suppression systems. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
CB-B	Control Building 903, DC Switchgear Room 1B and Battery Room 1B

Fire Zone	Description
8F	Battery Room 1B
8G	DC Switchgear Room 1B

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A will be operated to maintain Suppression Pool temperature.	None
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - Flow indications for CS, RHR, and SW [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train A to maintain RPV level.	CBB-03
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
CB-B	Control Building 903, DC Switchgear Room 1B and Battery Room 1B	
Vital Auxiliaries	Mechanical: -REC will be supplied by SW Train A to provide the cooling supply to the ECCS. -SW Train A will be operated to provide the cooling supply to the REC system and RHR Train A Heat Exchanger. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160V Bus 1F - 125/250 VDC Train A is available HVAC: - RCIC/CS Train A - Quad area cooling - AC Switchgear Room 1F - Essential Control Building HVAC system - DC Switchgear Room 1A - Essential Control Building HVAC system - Battery Room 1A - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1A	CBB-01 CBB-02

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	**Description**
CB-B	Control Building 903, DC Switchgear Room 1B and Battery Room 1B

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-032 - Evaluation of DC SWGR Rooms 1A and 1B and Battery Rooms 1A and 1B Fire Barrier Separation

Purpose This evaluation was written to justify fire dampers provided in supply ventilation ductwork routed through the barrier between each Switchgear Room and the adjacent Seal Water Pump Area and Corridor that have not been installed in the plane of the barrier, as required by the manufacturers installation guidelines. Also, the dampers at the wall boundary where the ventilation ductwork penetrates the barriers from Battery Rooms 1A and 1B to the Seal Water Pump Area and Corridor have been blocked open to ensure exhaust system operability. In addition, the fire doors in these barriers have NFPA 80 code deviations.

Conclusion Based on the lack of significant combustible loading and fire hazards in the areas, the presence of 3-hour rated dampers that will provide significant fire separation regardless of the mounting position, and installed fire protection features including detection system coverage and manual suppression capabilities, the fire damper configurations are adequate for the fire hazards of the adjacent areas. The fire door code deviations include excessive door-to-door gap clearances and modifications made to ensure proper door operation. Considering the lack of significant combustible loading and fire hazards in the area and administrative procedures controlling door operation, these deviations do not represent a significant decrease in fire safety.

Basis The bases that justify this conclusion are summarized as follows:

• The combustible loading classification for Switchgear Rooms 1A and 1B are "LOW" and is mainly attributed to miscellaneous plastics and transient loading allowances. There are few fixed combustibles in any of the zones and the equivalent fire severities are minimal. The combustible loading classification in the Corridor is also "LOW" and consists of miscellaneous cable, hose and plastic, and transient allowances.
• The lack of significant combustible materials in the zones and the lack of intervening combustibles significantly reduce the chance of fire propagation between zones. Transient combustibles are controlled by plant procedures, effectively reducing the possibility of a fire involving transient materials.
• Each fire zone is provided with detection system coverage. In the event of a fire in the Switchgear Rooms or Corridor, detection system actuation will result in rapid fire brigade response and subsequent manual extinguishment utilizing hose stations and portable extinguishers strategically located in adjacent fire zones.
• Based on the material and installation of the ventilation ducts, they are considered to prevent fire propagation for up to 1 hour.

Table B-3 Fire Area Transition

Fire Area		**Description**
CB-B		Control Building 903, DC Switchgear Room 1B and Battery Room 1B
EEEE Title		EE 09-042 - Evaluation of 1-Hour Marinite Wall in Battery Room 1B
	Purpose	This evaluation is written to address the fire protection adequacy of the 1-hour fire-rated wall enclosure located in Battery Room 1B.
	Conclusion	Based on a review of the potential fire hazards located in Battery Room 1B, with the 1-hour fire-rated wall enclosure and the existing fire detection system, the existing barrier provided is adequate to prevent the fire spreading from Battery Room 1B to the 1-hour fire-rated enclosure.
	Basis	The bases that justify this conclusion are summarized as follows: • The combustible loading classification for Battery Room 1B is "LOW" and is mainly attributed to plastics and transient loading allowances. Battery Room 1B has an equivalent fire severity of 30 minutes. The combustible loading consists of cables routed in conduit riser. The lack of significant combustible materials in the zones and the lack of intervening combustibles significantly reduce the chance of fire propagation between zones. • Based on the low combustible loading in Battery Room 1B, a fire will not be able to develop to such intensity as to challenge the as-installed configuration. • Battery Room 1B is provided with smoke detection system coverage. Detection system actuation will prompt rapid fire brigade response and manual extinguishment via portable extinguishers and manual hose stations strategically located in adjacent zones. Pre-fire plans are available for Battery Room 1B. • Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of a fire. • Safe shutdown can be accomplished independent of Battery Room 1B.

Table B-3 Fire Area Transition

Fire Area	Description
CB-B	Control Building 903, DC Switchgear Room 1B and Battery Room 1B

Variances from Deterministic Requirements (VFDR)

CBB-01

Description

Loss of critical electrical components due to loss of Control Building HVAC/Switchgear Cooling (HV-FAN-SF-SWGR-1F, HV-FAN-EF-SWGR-1F, EE-MCC-LX, and EE-PNL-CDP1A).

The NSCA model requires that either train of Switchgear Room cooling fans be available to ensure the Switchgear remains available post-fire. The NSCA model also requires that either train of Battery power be available post-fire.

Fire damage to cables will not preclude operation of the EF-SWGR-1F and SF-SWGR-1F fans from the 1F AC Switchgear Room. Cable damage will affect the operation of ventilation damper AD-1405 by energizing its solenoid and keeping the damper open.

Fire-induced failure of cables for Control Building HVAC and/or a fire in the Train B DC Switchgear and/or Battery room will result in loss of cooling to EE-PNL-CDP1A/EE-MCC-LX. This would result in a loss of all DC panels. With temperatures affecting operations, cooling via open compartments is effective to ensure operation of the panels.

This is a separation issue for Vital Auxiliaries.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
CB-B	Control Building 903, DC Switchgear Room 1B and Battery Room 1B

CBB-02

Description

Establish vital auxiliaries by ensuring power to the credited 4160F Bus from Emergency Transformer (EE-CB-4160G-1GS).

The NSPC require at least one train of electrical power available post-fire.

Fire damage to control power panels and supply cables on the 4160G Bus would not allow breakers to open automatically, if required, and therefore cause a loss of the credited 4160F Bus. EE-CB-4160G-1GS would normally close during the transfer process to the Emergency Transformer to power the 4160G Bus.

This is a separation issue for Vital Auxiliaries.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

CBB-03

Description

Preventing a RR Pump Seal LOCA due to inability to secure the RR Pumps from the Control Room with the REC Non-Critical Header secured (RR Pump Breakers for 4160D-1DS).

Breaker F/FDR to the 4160V Bus from the Startup Transformer: This is a normally available, required open breaker that provides motive power to the RR Pump B. The RR Pump is required to trip to prevent a potential RR Seal LOCA when REC is not available to provide cooling. This would challenge the NSPC for Inventory and Pressure Control. Remote operation of the breaker(s) from the Control Room is lost due to EE-PNL-BB11 cable DC331 being potentially damaged, and therefore, a loss of control power to the breaker.

NOTE: REC Non-Critical Header is secured in this fire area to address potential Containment over-pressure.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Table B-3 Fire Area Transition

Fire Area	**Description**
CB-B	Control Building 903, DC Switchgear Room 1B and Battery Room 1B

Disposition A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
8F	Detection	Ionization	R	N	N	N	Y	Y	N
8G	Detection	Ionization	R	N	N	N	Y	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	Description
CB-B	Control Building 903, DC Switchgear Room 1B and Battery Room 1B

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. There are no fixed suppression systems in the area, and the Battery Rooms and DC Switchgear Rooms on the 903'-6" Elevation of the Control Building are not subject to any adverse effects by water intrusion from fire suppression systems. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
CB-C	RPS Room 1B

Fire Zone	Description
8B	RPS Room 1B

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A will be operated to maintain Suppression Pool temperature.	CBC-02
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - RHR, SW, and CS flow indication, and REC pressure indication [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train A to maintain RPV level.	CBC-01
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip, or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
CB-C	RPS Room 1B	
Vital Auxiliaries	Mechanical: -REC Train A will be operated to provide the cooling supply to the ECCS. -SW Train A will be operated to provide the cooling supply to the REC system and RHR Train A Heat Exchanger. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160V Bus 1F - 125/250 VDC Train A and B are available HVAC: - HPCI/CS Train A - Quad area cooling - AC Switchgear Rooms - Essential Control Building HVAC system - DC Switchgear Rooms - Essential Control Building HVAC system - Battery Rooms - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1A - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	Description
CB-C	RPS Room 1B

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-040 - Evaluation of Auxiliary Relay Room and RPS Room 1B Appendix R Fire Barriers

Purpose

This evaluation justifies the adequacy of ventilation duct penetrations, without fire-rated dampers, routed through the barriers that separate the Auxiliary Relay Room, RPS Room 1B, and RPS Room 1A on the 903'-6" Elevation of the Control Building. Additionally, a fire rating cannot be assigned to the barriers due to miscellaneous door discrepancies concerning doors H102, H103, and H104.

- Door H102 separates RPS Room 1B (Fire Area CB-C) from the Seal Water Pump Area and Corridor (Fire Area CB-A)
- Doors H103 and H104 separate the Auxiliary Relay Room (Fire Area CB-D) from the Seal Water Pump Area and Corridor (Fire Area CB-A)

Conclusion

Based on the installed fire protection features, including detection system coverage and manual fire suppression capabilities, and the lack of significant fire hazards, and the types of combustible materials, the ventilation ductwork is sufficient to provide protection from spread of fire prior to fire brigade response and subsequent manual extinguishment, if necessary. Discrepancies concerning doors H102, H103, and H104 are relatively minor, and include door clearances, and minor modification for proper operation. These discrepancies are therefore not considered to affect fire safety. The separation that has been provided is considered to be adequate for the fire hazards of the areas.

Basis

The bases that justify this conclusion are summarized as follows:

- There are typically few fixed combustibles in any of the zones, and the equivalent fire severities are minimal. The exception is the Auxiliary Relay Room, which contains a significant amount of cable insulation in open cable trays routed at the ceiling elevation. The content of the RPS Room consists mainly of batteries. The combustibles in the Corridor mainly consist of miscellaneous cable, hose and plastic, and transient allowances. Cables in the Corridor are routed in conduit, and potential ignition sources in the zones are limited to energized electrical equipment, typically in cabinets, and potential transient sources.
- A cable tray fire in the Auxiliary Relay Room would be characterized as a slow propagating fire that has the potential to produce significant amounts of smoke. Based on the installed detection system coverage, fire brigade response and manual extinguishment would occur prior to the fire being able to breach the barrier via the ductwork.
- The lack of significant combustible materials in the remaining zones and the lack of intervening combustibles significantly reduce the chance of fire propagation between zones.

Table B-3 Fire Area Transition

Fire Area	Description
CB-C	RPS Room 1B
	• Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of a fire. • The supply registers in the Auxiliary Relay Room are provided with 3-hour rated fire dampers (HV-AD-AD1434 and HV-AD-AD1435). A 3-hour fire-rated damper has been provided in the Auxiliary Relay Room at the exhaust register (HV-AD-AD1436). • Sheet metal ductwork is recognized by the NFPA to provide up to 1-hour fire separation when properly hung, and the fire is stopped. • Each fire zone is provided with detection system coverage. In the event of a fire in the RPS Room, Auxiliary Relay Room, or Corridor, detection system actuation will result in rapid fire brigade response and subsequent manual extinguishment utilizing hose stations and portable extinguishers strategically located in adjacent fire zones.

Table B-3 Fire Area Transition

Fire Area	Description
CB-C	RPS Room 1B

Variances from Deterministic Requirements (VFDR)

CBC-01

Description Ability to operate SRV's automatically/remotely from the Control Room is lost due to cable fire damage (Pilot Valves SPV71E, SPV71G, and SPV71H).

The NSPC require the ability to reduce pressure in the RPV to maintain water inventory above the active fuel using low pressure injection systems. Automatic control of these valves is lost, to support operation of the CS Pumps for Inventory Control.

This is a separation issue for Inventory and Pressure Control.

Disposition A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

CBC-02

Description Preventing a loss of SW cooling flow to credited RHR Heat Exchanger (SW-MOV-MO89A).

SW Train A is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC.

SPC Train A is the credited train of Decay Heat Removal. SW-MOV-MO89A is normally closed and with the loss of RHR-LOGIC-A power, the valve will not spuriously open. However, the loss of RHR logic will lock in the closed signal on the valve, securing SW flow to the RHR Heat Exchanger, and therefore result in a loss of SPC Train A.

This is a separation issue for Decay Heat Removal.

Table B-3 Fire Area Transition

Fire Area	Description
CB-C	RPS Room 1B

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
8B	Detection	Ionization	R	N	N	N	Y	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. There are no fixed suppression systems in the area and the RPS MG Set Room on the 903'-6" Elevation of the Control Building is not subject to any adverse effects by water intrusion from fire suppression systems. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

Fire Zone	Description
10A	Computer Room
10B	Control Room and SAS Corridor
8A	Auxiliary Relay Room
9A	Cable Spreading Room
9B	Cable Expansion Room

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	RHR system will use the alternate lineup to maintain temperature.	CBD-01 CBD-02 CBD-09 CBD-10 CBD-12
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV level and pressure [from Primary Control Station] - Suppression Pool level and temperature [from Primary Control Station] - Emergency Condensate Storage Tank level [from Primary Control Station] -HPCI flow, pressure, and turbine speed [from Primary Control Station]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. HPCI system will use alternate lineup to control RPV pressure and to maintain RPV level.	CBD-01 CBD-02 CBD-04 CBD-06 CBD-08 CBD-11 CBD-13 CBD-14
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	CBD-12

Table B-3 Fire Area Transition

Fire Area	Description	
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room	
Vital Auxiliaries	Mechanical: -REC will be supplied by SW Train B to provide the cooling supply to the ECCS in the alternate lineup. -SW Train B will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers in the alternate lineup. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160V Bus 1G - 125/250 VDC Train B is available HVAC: - HPCI/CS Train B - Quad area cooling - AC Switchgear Room 1G - Essential Control Building HVAC system - DC Switchgear Room 1B - Battery Room 1B - Auxiliary Relay Room and RPS MG Set Room 1B	CBD-03 CBD-05 CBD-07 CBD-15

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area — **Description**

CB-D — Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

hazards of the areas, and adequate separation has been provided. In general, minor variations to the configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.

Basis

The bases that justify this conclusion are summarized as follows:

- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade.
- Ventilation systems can typically be used for smoke and heat removal.
- Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
- The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202.
- The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105.
- The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201.
- The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104.
- The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104.
- The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6.
- The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited for the acceptability of door R7.
- The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

the acceptability of doors R101 and R102.
• The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102.
• The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115.
• Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.

EEEE Title EE 09-036 - Evaluation of Cable Expansion Room Penetration Seals

Purpose This evaluation documents the adequacy of the non-fire rated expansion joints in the Cable Expansion Room Appendix R fire barriers. In addition, conduit penetrations exist in the floor/ceiling assembly penetrating to the Office Building Corridor below that are not sealed with grout to the depth specified in the design details.

Conclusion Based on the fire protection features provided in the Cable Expansion Room, including automatic suppression and detection system coverage, the ability to achieve safe shutdown independent of the area, and the lack of safe shutdown equipment/cables and combustible materials in the fire zone adjacent to the Cable Expansion Room, the configurations, as provided, are considered adequate.

Basis The bases that justify this conclusion are summarized as follows:

• The expansion joints are provided with 14 gauge metal covers. The metal covers will provide some degree of fire protection to impede the spread of fire, smoke, and hot gases to the adjacent fire zones.
• An automatic sprinkler system and smoke detectors are provided in the Cable Expansion Room. In the event of a fire in the area, detection system actuation will result in alarm in the Control Room, fire brigade response, and subsequent manual extinguishment utilizing hose stations and portable extinguishers.
• The presence of automatic detection and suppression ensures that a fire will be limited by automatic or manual suppression, such that breaching of the barriers via the expansion joints or conduit penetrations will be limited.
• Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of a fire.

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

EEEE Title EE 09-040 - Evaluation of Auxiliary Relay Room and RPS Room 1B Appendix R Fire Barriers

Purpose

This evaluation justifies the adequacy of ventilation duct penetrations, without fire-rated dampers, routed through the barriers that separate the Auxiliary Relay Room, RPS Room 1B, and RPS Room 1A on the 903'-6" Elevation of the Control Building. Additionally, a fire rating cannot be assigned to the barriers due to miscellaneous door discrepancies concerning doors H102, H103, and H104.

- Door H102 separates RPS Room 1B (Fire Area CB-C) from the Seal Water Pump Area and Corridor (Fire Area CB-A)
- Doors H103 and H104 separate the Auxiliary Relay Room (Fire Area CB-D) from the Seal Water Pump Area and Corridor (Fire Area CB-A)

Conclusion

Based on the installed fire protection features, including detection system coverage and manual fire suppression capabilities, and the lack of significant fire hazards, and the types of combustible materials, the ventilation ductwork is sufficient to provide protection from spread of fire prior to fire brigade response and subsequent manual extinguishment, if necessary. Discrepancies concerning doors H102, H103, and H104 are relatively minor, and include door clearances, and minor modification for proper operation. These discrepancies are therefore not considered to affect fire safety. The separation that has been provided is considered to be adequate for the fire hazards of the areas.

Basis

The bases that justify this conclusion are summarized as follows:

- There are typically few fixed combustibles in any of the zones, and the equivalent fire severities are minimal. The exception is the Auxiliary Relay Room, which contains a significant amount of cable insulation in open cable trays routed at the ceiling elevation. The content of the RPS Room consists mainly of batteries. The combustibles in the Corridor mainly consist of miscellaneous cable, hose and plastic, and transient allowances. Cables in the Corridor are routed in conduit, and potential ignition sources in the zones are limited to energized electrical equipment, typically in cabinets, and potential transient sources.
- A cable tray fire in the Auxiliary Relay Room would be characterized as a slow propagating fire that has the potential to produce significant amounts of smoke. Based on the installed detection system coverage, fire brigade response and manual extinguishment would occur prior to the fire being able to breach the barrier via the ductwork.
- The lack of significant combustible materials in the remaining zones and the lack of intervening combustibles significantly reduce the chance of fire propagation between zones.
- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of a fire.

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room
	• The supply registers in the Auxiliary Relay Room are provided with 3-hour rated fire dampers (HV-AD-AD1434 and HV-AD-AD1435). A 3-hour fire-rated damper has been provided in the Auxiliary Relay Room at the exhaust register (HV-AD-AD1436). • Sheet metal ductwork is recognized by the NFPA to provide up to 1-hour fire separation when properly hung, and the fire is stopped. • Each fire zone is provided with detection system coverage. In the event of a fire in the RPS Room, Auxiliary Relay Room, or Corridor, detection system actuation will result in rapid fire brigade response and subsequent manual extinguishment utilizing hose stations and portable extinguishers strategically located in adjacent fire zones.

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

EEEE Title EE 86-2 - Evaluation of a Ventilation Opening Through the Cable Spreading Room Floor Appendix R Fire Barrier

Purpose The purpose of this engineering evaluation is to document the acceptability of fire damper HV-AD-AD1556 located in the floor/ceiling boundary where the ventilation ductwork penetrates from Battery Room 1A to the Cable Spreading Room. The damper has been blocked open to ensure exhaust system operability, as accidental closure would render the exhaust system inoperable. In addition, steel pipe has been used as ventilation ductwork in the Cable Spreading Room, which is not typical for ventilation.

Conclusion Based on the significant construction of the assembly, the lack of significant fire hazards and combustible loading, and the presence of installed fire protection features including fire detection and fixed fire suppression, the configuration that has been provided is adequate for the fire hazards of the areas.

Basis The bases that justify this conclusion are summarized as follows:

- The duct assembly in the Cable Spreading Room is of significant steel construction. There is reasonable assurance that if a fire in the Cable Spreading Room was to occur, it would not breach the ductwork (i.e., the pipe) based on the significant pipe construction, and be able to propagate down into the Battery Room.
- Although the combustible loading classification in the Cable Spreading Room is "HIGH," due to the large quantity of cable insulation in the area, installed fire protection features are provided to mitigate the effects of a fire.
- The combustible loading classification in the Battery Room is "LOW" and consists mainly of plastic associated with battery cases. The lack of significant fire hazards and combustible loading in Battery Room 1A precludes the possibility of a fire developing to such intensity as to breach the ventilation ductwork in the Battery Room and then breach the significant pipe construction provided in the Cable Spreading Room.
- Automatic detection system coverage will alert the fire brigade of fire conditions and automatic suppression system coverage will limit the intensity of a fire in the Cable Spreading Room.
- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

EEEE Title EE 86-5 - Evaluation of HVAC Ducts and Fire Door Between the Control Room and Controlled Corridor

Purpose This engineering evaluation is being prepared to document the acceptability of the three ventilation duct penetrations that are not provided with fire dampers located in the Control Room south wall, which provides Appendix R fire barrier separation between the Control Room and the Controlled Corridor.

Conclusion Based on the construction of the assembly, the lack of significant fire hazards and combustible loading and the presence of installed fire protection features including fire detection, the configuration that has been provided is adequate for the fire hazards of the areas.

Basis The bases that justify this conclusion are summarized as follows:

- The Control Room is constantly attended. Any anticipated fire would be quickly discovered and extinguished in its incipient stage.
- The two larger vent ducts are at an elevation approximately 12 feet above the floor, which is above the suspended ceilings in the corridor. It is, therefore, unlikely that a fire would cause direct flame exposure to the duct in the corridor.
- The corridor area is essentially void of any significant combustible loading with the exception of miscellaneous combustibles stored in closed metal lockers. There are essentially no combustibles normally located in the corridor near the duct penetrations. It is extremely unlikely that the penetrations would ever be exposed to a significant fire.
- Fire propagation from the Control Room to the corridor is considered inconsequential from a safe shutdown standpoint since the office corridor does not contain any safe shutdown cables or equipment.
- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- There are no openings, such as ventilation grills, on either side of the wall in the large ducts to allow flame propagation to enter the ductwork.
- The Battery Room exhaust duct is a Schedule 80 pipe. Fire propagation into or through this pipe is not considered a credible event, as it is of significant construction so as to prevent fire spread.

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

Variances from Deterministic Requirements (VFDR)

CBD-01

Description

Preventing a full or partial loss of SW for supporting Decay Heat Removal and Inventory and Pressure Control from the starter racks and ASD Room with SW supplying REC (EE-CB-4160G-SWP1B, REC-FIS-24-ASD, REC-MOV-695MV-ASD, REC-MOV-714MV-ASD, SW-AOV-TCV451B-ASD, SW-MOV-37MV, SW-MOV-651MV-ASD, SW-MOV-887MV-ASD, SW-MOV-889MV-ASD, and SW-STNR-B).

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. HPCI is credited for Inventory Control, SPC Train B is credited for Decay Heat Removal, REC Train B provides cooling for RHR Pump 1D and the RHR and HPCI Rooms (Quads), and SW Train B provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The complete loss or diversion of SW to REC could challenge these NSPC.

(1) Based on cable failures, EE-CB-4160G-SWP1B may trip open.
(2) HPCI Room cooling flow may not be available due to cable failures to FIS indication.
(3) REC-MOV-695MV is not controlled from the ASD Room, and cable failure removes the ability to remotely close the valve prior to leaving the Control Room.
(4) REC-MOV-714MV is not controlled from ASD Room, and cable failure removes the ability to remotely close the valve prior to leaving the Control Room.
(5) Cable damage may cause SW-AOV-TCV451B to fail closed, securing the Reactor Building SW flow returning from the REC system, removing cooling flow to the HPCI and RHR Pumps.
(6) SW-MOV-37MV is not controlled from the ASD Room, and may need to be closed to isolate the diversion of SW flow from the critical SW and REC loads to the non-critical SW loads.
(7) SW-MOV-651MV is not controlled from the ASD Room, and needs to be closed to prevent flow diversion from the SW system to the REC Heat Exchanger, which is bypassed in ASD lineup.
(8) SW-MOV-887 is not controlled from the ASD Room, and needs to be open to supply SW to the REC Critical Header.
(9) SW-MOV-889 is not controlled from the ASD Room, and needs to be open to supply SW to the REC Critical Header.
(10) SW-STNR-B based on cable failures may lose its automatic features, resulting in clogging of the strainer and reducing/securing SW flow.

This is a separation issue for Inventory and Pressure Control and Decay Heat Removal.

Table B-3 Fire Area Transition

Fire Area		Description
CB-D		Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

CBD-02

Description

Ability to close MSIVs automatically/remotely from the Control Room is lost due to cable fire damage and Control Room abandonment.

MSIVs (MS-AOV-AO80A, MS-AOV-AO80B, MS-AOV-AO80C, and MS-AOV-AO80D) may not go closed from the Control Room due to cable damage. The NSPC requires a means of isolating the RPV to ensure sufficient water inventory for Decay Heat Removal.

This is a separation issue for Inventory and Pressure Control and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

CBD-03

Description

Establish Vital Auxiliaries and control by powering the credited 4160G Bus from the Emergency Transformer, and MCC-R from its alternate power supply (EE-CB-4160F-1FS, EE-CB-4160G-1GB, EE-CB-4160G-1GE, EE-CB-4160G-1GS, EE-CB-4160G-RSWP1B-OCT, EE-CB-4160G-RSWP1D-OCT, and EE-CB-4160G-SS1G).

The NSCA require at least one source of AC power be available. Offsite power from the Emergency Transformer is unavailable using the normal transfer mode due to fire damage to supply breakers feeding the 4160G Bus.

There are multiple breaker and cables affected on the 4160F Bus. Multiple cables that support EE-CB-4160F-1FS are also damaged due to fire, and may close the breaker.

The EE-CB-4160G-1GB breaker is normally closed, and should open as part of the transfer process due to loss of the Start-Up Transformer, but cable damage to H561 may not allow it to open. The DG and Emergency Transformer are not rated to normally carry both the Vital and Non-Vital buses. Additionally, the DG2/1GS breakers are interlocked with the BG/GB breakers, and will not close if the breaker is not opened.

Cable damage (C730, H571, H572, H573) due to fire could close EE-CB-4160G-1GE, or blow its control power fuses. The breaker is normally closed, and needs to be open to allow for control of the DG and not allow damage to the DG or the bus if the DG came onto the bus out-of-phase.

Based on cable failure (C730, H551, H552, H553, H555, X10) due to fire damage, breaker EE-CB-4160G-1GS may trip open or blow control power fuses, and not close.

EE-CB-4160G-RSWP1B-OCT may not trip due to potential fire damage to the control and load cables, resulting in loss of the credited 4160G Bus.

EE-CB-4160G-RSWP1D-OCT may not trip due to potential fire damage to the control and load cables, resulting in loss of the credited 4160G Bus.

Cable damage to H542 and H543 may cause EE-CB-4160G-SS1G to trip open. Opening this breaker causes a loss of power to the 480V G Bus. The breaker is normally closed, and desired closed to support NSPC requirements for Process Monitoring and Decay Heat Removal.

Table B-3 Fire Area Transition

Fire Area	**Description**
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

This is a separation issue for Vital Auxiliaries.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDRs meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

CBD-04

Description

Ability to Close SRVs automatically/remotely from the Control Room is lost due to cable fire damage (Pilot Valves SPV71A, SPV71B, SPV71C, SPV71D, SPV71E, SPV71F, SPV71G, and SPV71H).

The NSCA require the ability to isolate the RPV to maintain water inventory above the top of active fuel. Spurious operations could result in the opening of up to eight ADS (SRV) valves. The NSPC could be challanged if the affected ADS valves are not returned to their fail-safe closed position within 18 minutes for single spurious operation, and a shorter time if more than one ADS valve spuriously opens.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

CBD-05

Description

Loss of Battery/DC Switchgear Room/Critical Switchgear cooling due to cable damage to cooling fans and dampers (EE-BAT-125-1B, EE-BAT-250-1B, EE-SWGR-125B, EE-SWGR-250B, HV-FAN-EF-SWGR-1G, and HV-FAN-SF-SWGR-1G).

Fire-induced failure of cables for Control Building HVAC will result in loss of cooling to Battery Room 1B. The NSCA model requires that either train of battery power be available post-fire. With temperatures affecting operation, cooling via open compartment is effective to ensure operation of the batteries.

Fire damage to cables will not preclude operation of the EF-SWGR-1G and SF-SWGR-1G fans from the 1G AC Switchgear Room. Cable damage will affect the operation of ventilation damper AD-1408 by energizing its solenoid and keeping the damper open. Damper AD-1407 has its cable routed in dedicated conduit in area CB-A. The NSCA model requires that either train of Switchgear Room cooling fans be available to ensure the Switchgear remains available post-fire.

This is a separation issue for Vital Auxiliaries.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

CBD-06

Description

Preventing a RR Pump Seal LOCA due to inability to secure the RR Pumps from the Control Room with the REC Non-Critical header secured (RR Pump Breakers for 4160C-1CS and 4160D-1DS).

Breakers F/FDR to the 4160V Buses from the Startup Transformer: These are normally available, required open breakers that provide motive power to the RR Pumps. The RR Pumps are required to trip to prevent potential RR Pump Seal LOCA when REC is not available to provide cooling. This would challenge the NSPC for Inventory and Pressure Control. Remote operation of the breaker(s) from the Control Room is lost due to cables H248, H261 and M1 (1C Bus) and H288, H301 and M1 (1D Bus) fire-induced damage.

NOTE: REC Non-Critical Header is secured in this fire area to address potential containment over-pressure.

This is a separation issue for Inventory and Pressure Control.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

CBD-07

Description

Loss of both A and B Train 125V and 250V Battery Chargers due to cable damage (EE-CHG-125-1B and EE-CHG-250-1B).

Fire damage to cable MTX10 causes a loss of Main Supply Power to the B Train 125V Battery Charger. The NSCA require at least one train of DC power be available post-fire, and both A and B Train Battery Chargers are affected. The B Train batteries are the credited train post-fire. Battery Charger repair is required commencing within 1.5 hours and completed within 4.5 hours from the time the Battery Charger is lost.

Fire damage to cable MTX9 causes a loss of Main Supply Power to the B Train 250V Battery Charger. The NSCA require at least one train of DC power be available post-fire, and both A and B Train Battery Chargers are affected. The B Train batteries are the credited train post-fire. Battery Charger repair is required commencing within 1.5 hours and completed within 4.5 hours from the time the Battery Charger is lost.

This is a separation issue for Vital Auxiliaries.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

CBD-08

Description

Loss of RPV water inventory to Reactor Building Sumps and Radwaste (CRD-SOV-SO31A and CRD-SOV-SO31B).

Cable damage may result in the spurious opening of CRD-SOV-SO31A and CRD-SOV-SO31B. This opening of the CRD Scram Discharge Volume Vent and Drain Valve SOVs would result in loss of RPV Inventory to the Reactor Building Sumps and then Radwaste. The NSCA require a means of isolating the RPV post-fire to ensure sufficient water inventory. Based on CNS-PSA-007, the flow rate from CRD seal leakage is assumed to be 450 to 600 gpm initially, reducing to 73 gpm after 70 minutes, and down to 40 gpm after 4 hours, based on all CRD seals leaking at the same time at the maximum amount prior to repair.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

CBD-09

Description

Loss of Decay Heat Removal, due to flow diversion or loss of a pump in the RHR system or loss of cooling from the SW system due to cable damage (RHR-MOV-MO20-ASD, RHR-MOV-MO26B-ASD, RHR-MOV-MO57-ASD, EE-CB-RHRP1D, and SW-MOV-MO89B-ASD).

SPC Train B is the credited train for Decay Heat Removal. The complete loss of RHR flow, or the diversion of flow, would challenge this NSPC.

Based on cable failure, valve RHR-MOV-MO20 may spuriously open, resulting in a 20-inch flow diversion through the bypass line, resulting in a loss or reduction of flow during the SPC Train B mode of operation.

The normal method of operation of RHR-MOV-MO26B during ASD would be to remove control power fuses (3C), and operate from the MCC starter or local handwheel. Control cable damage may spuriously operate this valve prior to removing the control power fuses. The feeder cable is undamaged. RHR-MOV-MO26B presents a 10-inch diversion path to establishing SPC Train B.

The normal method of operation of RHR-MOV-MO57 during ASD would be to remove control power fuses and operate from the MCC starter or local handwheel. Control cable damage may spuriously operate this valve prior to removing the control power fuses. The feeder cable is undamaged. RHR-MOV-MO57 presents a 4-inch diversion path to establishing SPC Train B.

Due to multiple damaged cables, the RHR Pump breaker may fail to close automatically or remotely. This would result in a loss of Decay Heat Removal for SPC Train B.

Based on cable damage to RHRSW Booster Pump breakers, neither breaker can be closed removing remote operability of SW-MOV-MO89B. The valve is normally closed, and is required to be open to establish SW flow through the RHR Heat Exchanger.

This is a separation issue for Decay Heat Removal.

Table B-3 Fire Area Transition

Fire Area		Description
CB-D		Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room
	Disposition	A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria. Risk: Acceptable with Recovery Action. See LAR Attachment G, Table G-1 for Recovery Action details. Modification: None
CBD-10		
	Description	Preventing the loss of containment over-pressure in support of RHR Pump operation for SPC operation (RW-AOV-AO82 and RW-AOV-AO94, REC-MOV-712MV, REC-MOV-713MV, PC-AOV-245AV, and PC-AOV-246AV) The NSPC require a means to provide Decay Heat Removal post-fire. SPC Train B is credited for Decay Heat Removal. The loss of containment over-pressure could result in the loss of Net Positive Suction Head (NPSH) for the credited RHR Pump during the SPC mode of operation. Cable damage to the SOVs associated with RW-AOV-AO82, RW-AOV-AO83, RW-AOV-AO94, and RW-AOV-AO95 may result in spurious opening. These valves isolate the drywell equipment and floor drain sump discharge paths, and their opening would result in a loss of containment over-pressure. REC-MOV-712MV and REC-MOV-713MV isolate the REC Critical and Non-Critical Headers. The valves are normally open, and are required to be closed to secure flow to the Drywell Coolers. Based on cable damage and Control Room abandonment, remote operation of these valves is not possible. Cable damage to M355 (PC-AOV-245AV) and M358 (PC-AOV-246AV) may cause the valves to fail open. These valves are normally closed and desired closed. Spurious opening of these valves results in Containment and Suppression Chamber vent/purge lines opening, resulting in loss of containment over-pressure. This is a separation issue for Decay Heat Removal.

Table B-3 Fire Area Transition

Fire Area		Description
CB-D		Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

CBD-11

Description

Prevent loss of RPV water inventory and/or high pressure in the low pressure portion of the RWCU system from the Control Room due to cable fire damage (RWCU-MOV-MO15).

The NSPC requires a means of isolating the RPV and low pressure systems from the RPV post-fire. Cable damage (MR122 and PC86) may cause spurious opening of RWCU-MOV-MO15, and the inability to close RWCU-MOV-MO15 would result in loss of RPV inventory to the RWCU system, or cause high pressure in the low pressure portion of RWCU piping downstream.

This is a separation issue for Inventory and Pressure Control.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

CBD-12

Description

Inability to control power or ensure positive reactor shutdown due to fire damage (RPS-ELECT).

The NSCA require a means of ensuring the reactor is shutdown post-fire. Cable damage may result in the inability of a manual scram to occur from the Control Room.

EE 10-064 shows that a fire in the Control Room will not totally disable the ability to close the MSIVs from the Control Room. Control Room operation to close the MSIVs will be successful if performed.

This is a separation issue for Reactivity Control and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

CBD-13

Description

Preventing an RPV overfill condition with spurious RCIC startup and loss of control due to cable damage (RCIC-MOV-MO15).

The NSCA require a means of maintaining RPV water level post-fire. RCIC normally maintains level during transient conditions, but with the cables damaged due to fire in this area for RCIC-MOV-MO15, RCIC-MOV-MO16, and RCIC-MOV-131MV, system status is not assured, and may result in a startup of the RCIC system without control, and cause an overfill condition. RPV water level can still be maintained by the HPCI system if required.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

CBD-14

Description

Preventing loss of remote valve control for essential systems from the ASD Room (EE-MCC-R-1A).

The NSCA require a mean of RPV pressure and inventory control post-fire. The feeder cable from MCC-K (MK134) is affected by the fire and MCC-R is the power supply to HPCI-MOV-MO15-ASD, MS-MOV-MO74, REC-MOV-695MV, RHR-MOV-MO20, RHR-MOV-MO57-ASD, and RWCU-MOV-MO15. The normal position of HPCI-MOV-MO15 is open and the desired position is open, with HPCI as the credited train of RPV inventory control. HPCI-MOV-MO15 may spuriously close prior to shifting to the ASD Room. Shift of power will provide both indication of valve position, and allow for control power from the ASD panel.

This is a separation issue for Inventory and Pressure Control.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

CBD-15

Description

Loss of power to critical equipment for plant operation (SW Pump B and SW Pump D).

The NSCA require a means of powering the critical pieces of equipment SW Pump B feeder cable H521, and SW Pump D feeder cable H531. Failure of these cables will result in loss of the credited Service Water function.

This is a separation issue for Vital Auxiliaries.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
10A	Detection	Ionization	R	Y	N	N	Y	N	N
10A	Suppression	Auto Total Flooding Halon 1301	F	N/A	N	N	Y	N	N
10B	Detection	Heat	R	N	N	N	N	Y	N
10B	Detection	Ionization	R	N	N	N	N	Y	N
8A	Detection	Incipient (in Panels 9-32 and 9-33)	R	N	N	N	N	Y	N
8A	Detection	Ionization	R	N	N	N	Y	Y	N
9A	Detection	Heat Actuated Devices	R	Y	N	N	Y	Y	N
9A	Detection	Ionization	R	Y	N	N	Y	Y	N
9A	Feature	Flame Impingement Shield	N/A	N/A	N	N	N	Y	N
9A	Feature	Flame Impingement Shield	N/A	N/A	N	N	N	Y	N
9A	Suppression	Preaction Sprinkler System	F	N/A	N	N	Y	Y	N
9B	Detection	Ionization	R	N	N	N	Y	Y	N
9B	Feature	Flame Impingement Shield	N/A	N/A	N	N	N	Y	N
9B	Suppression	Automatic Wet-Pipe	F	N/A	N	N	Y	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	Description
CB-D	Control Room, Computer Room, Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. In the event of normal operation, the pre-action and automatic wet pipe suppression systems will not adversely effect the other equipment operating within the Fire Zone. The possibility of inadvertent actuation of the pre-action suppression system in the Cable Spreading Room is not assumed since the suppression lines are not under constant pressure. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment. In the event of normal operation, the Halon system will not adversely effect the other equipment operating within the Fire Zone, as potential corrosion developing on equipment is a long term issue that would be significantly reduced by extensive post-discharge clean up. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

In Fire Zone 9A, fire insulated board has been installed to around Division II power feeds near the floor. Cable trays C209, C212, C213, C232 and C233 in Fire Zone 9A are provided with solid bottom 1/4" asbestos boards, and the trays below are provided with solid metal covers.

In Fire Zone 9A, flame impingement shielding is being installed for cable trays and conduit to prevent damage from fires involving panels PMIS-MUX-LNK6 and PMIS-MUX-LNK7. See LAR Attachment S, Table S-2, Item S-2.7.

In Fire Zone 9B a flame impingement shield has been installed beneath the Division II conduit bank.

Table B-3 Fire Area Transition

Fire Area	Description
DG-A	Diesel Generator Room 1A

Fire Zone	Description
14A	Diesel Generator Room 1A
14C	DG1 Day Tank Room

Regulatory Basis

4.2.3.2 - Deterministic Approach

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train B will be operated to maintain Suppression Pool temperature.	None
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - Emergency Condensate Storage Tank level [from Control Room] - RHR and SW flow indications [from Control Room] - HPCI flow, pressures, and turbine speed indication [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection is provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. The HPCI system will be used to control RPV pressure and to maintain RPV level.	None
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	**Description**	
DG-A	Diesel Generator Room 1A	
Vital Auxiliaries	Mechanical: -REC will be supplied by Train B to provide the cooling supply to the ECCS. -SW Train B will be operated to provide the cooling supply to the REC system and RHR Train B Heat Exchanger. Electrical (AC/DC): - Offsite Startup Transformer aligned to 4160V Bus 1G - 125/250 VDC Train B is available HVAC: - HPCI/CS Train B - Quad area cooling - AC Switchgear Room 1G - Essential Control Building HVAC system - DC Switchgear Room 1B - Essential Control Building HVAC system - Battery Room 1B - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1B - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	**Description**
DG-A	Diesel Generator Room 1A

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire

Table B-3 Fire Area Transition

Fire Area	Description
DG-A	Diesel Generator Room 1A

hazards of the areas, and adequate separation has been provided. In general, minor variations to the configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.

Basis

The bases that justify this conclusion are summarized as follows:

- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade.
- Ventilation systems can typically be used for smoke and heat removal.
- Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
- The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202.
- The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105.
- The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201.
- The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104.
- The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104.
- The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6.
- The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited for the acceptability of door R7.
- The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for

Table B-3 Fire Area Transition

Fire Area	Description
DG-A	Diesel Generator Room 1A

the acceptability of doors R101 and R102.
- The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102.
- The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115.
- Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.

Variances from Deterministic Requirements (VFDR)

None

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
14A	Detection	Heat	R	N	N	N	Y	Y	N
14A	Detection	Ionization	R	Y*	N	N	Y	Y*	N
14A	Suppression	Automatic Total Flooding Carbon Dioxide	F	N/A	N	N	Y	Y*	N
14C	Detection	Heat	R	Y	N	N	N	N	N
14C	Suppression	Automatic Total Flooding Carbon Dioxide	F	N/A	N	N	N	N	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	Description
DG-A	Diesel Generator Room 1A

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment.

In the event of normal operation, the carbon dioxide system will not adversely effect the other equipment operating within the Fire Zone as potential condensation developing on electrical components is a long term issue that would be eliminated by ventilating the compartment after actuation. Per drawing FH-16282, all carbon dioxide nozzles are located greater than five feet away from any equipment included in the PRA analysis, and therefore, the possibility of thermal shock due to carbon dioxide actuation has been deemed to be a non-issue in this Fire Area. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

*Note: The total flooding carbon dioxide system in the Diesel Generator Room is automatic, however, the Risk Significance field in the "Required Fire Protection Systems and Features" Table for the total flooding carbon dioxide system credits a manual capability of the suppression system only.

Table B-3 Fire Area Transition

Fire Area	Description
DG-B	Diesel Generator Room 1B

Fire Zone	Description
14B	Diesel Generator Room 1B
14D	DG2 Day Tank Room

Regulatory Basis

4.2.3.2 - Deterministic Approach

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A will be operated to maintain Suppression Pool temperature.	None
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, SW, RHR flow indications, and REC pressure indication [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train A to maintain RPV level.	None
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
DG-B	Diesel Generator Room 1B	
Vital Auxiliaries	Mechanical: -REC will be supplied by Train A to provide the cooling supply to the ECCS. -SW Train A will be operated to provide the cooling supply to the REC system and the RHR Train A Heat Exchanger. Electrical (AC/DC): - Offsite Startup Transformer is aligned to 4160V Bus 1F - 125/250 VDC Train A is available HVAC: - RCIC/CS Train A - Quad area cooling - AC Switchgear Room 1F - Essential Control Building HVAC system - DC Switchgear Room 1A - Essential Control Building HVAC system - Battery Rooms 1A - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1A - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	Description
DG-B	Diesel Generator Room 1B

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire

Table B-3 Fire Area Transition

Fire Area	Description
DG-B	Diesel Generator Room 1B

hazards of the areas, and adequate separation has been provided. In general, minor variations to the configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.

Basis

The bases that justify this conclusion are summarized as follows:

• Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
• Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade.
• Ventilation systems can typically be used for smoke and heat removal.
• Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
• The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202.
• The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105.
• The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201.
• The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307.
• The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104.
• The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103.
• The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104.
• The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6.
• The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited for the acceptability of door R7.
• The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for

Table B-3 Fire Area Transition

Fire Area	**Description**
DG-B	Diesel Generator Room 1B

the acceptability of doors R101 and R102.
- The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102.
- The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115.
- Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.

Variances from Deterministic Requirements (VFDR)

None

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
14B	Detection	Heat	R	N	N	N	Y	Y	N
14B	Detection	Ionization	R	Y*	N	N	Y	Y*	N
14B	Suppression	Automatic Total Flooding Carbon Dioxide	F	N/A	N	N	Y	Y*	N
14D	Detection	Heat	R	Y	N	N	N	N	N
14D	Suppression	Automatic Total Flooding Carbon Dioxide	F	N/A	N	N	N	N	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	Description
DG-B	Diesel Generator Room 1B

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment.

In the event of normal operation, the carbon dioxide system will not adversely effect the other equipment operating within the Fire Zone as potential condensation developing on electrical components is a long term issue that would be eliminated by ventilating the compartment after actuation. Per drawing FH-16282, all carbon dioxide nozzles are located greater than five feet away from any equipment included in the PRA analysis and therefore, the possibility of thermal shock due to carbon dioxide actuation has been deemed to be a non-issue in this Fire Area. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

*Note: The total flooding carbon dioxide system in the Diesel Generator Room is automatic, however, the Risk Significance field in the "Required Fire Protection Systems and Features" Table for the total flooding carbon dioxide system credits a manual capability of the suppression system only.

Table B-3 Fire Area Transition

Fire Area	Description
IS-A	Intake Structure

Fire Zone	Description
20A	Service Water Pump Area
20B	Circulating Water Pump and Traveling Screen Area

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A or B will be operated to maintain Suppression Pool temperature.	ISA-01 ISA-02 ISA-03
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, SW, RHR flow indications, and REC pressure indication [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of either Core Spray Train A or Train B to maintain RPV level.	ISA-01 ISA-02 ISA-03
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
IS-A	Intake Structure	
Vital Auxiliaries	Mechanical: -REC Train A or B will be operated to provide the cooling supply to ECCS. -SW Train A or B will be operated to provide the cooling supply to REC and the RHR Train A or B Heat Exchanger based on available SW pumps. Electrical (AC/DC): - Offsite Emergency Transformer is aligned to 4160V Bus 1F and 1G - 125/250 VDC Train A and B is available HVAC: - HPCI/RCIC/CS Trains A and B - Quad area cooling - AC Switchgear Rooms - Essential Control Building HVAC system - DC Switchgear Rooms - Essential Control Building HVAC system - Battery Rooms - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Rooms - Essential Control Building HVAC system	ISA-01 ISA-02 ISA-03

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Existing Engineering Equivalency Evaluations (EEEE)

None

Table B-3 Fire Area Transition

Fire Area	**Description**
IS-A	Intake Structure

Variances from Deterministic Requirements (VFDR)

ISA-01

Description

Preventing flow diversion from reducing the effectiveness of SW for cooling plants loads with only one SW pump operating (SW-MOV-36MV and SW-MOV-37MV).

SW Train A and B are required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. CS is credited for Inventory Control, SPC is credited for Decay Heat Removal, REC provides cooling for RHR Pumps and the HPCI, RCIC, RHR and CS Pump Rooms (Quads), and SW provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The diversion could challenge these NSPCs during single SW pump operation.

Fire in the area could result in physical damage to component cabling and result in the spurious opening without the ability to remotely close either or both valves. These valves support the SW system supply side cross connection and non-vital load isolation.

This is a separation issue for Inventory and Pressure Control, Vital Auxiliaries, and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
IS-A	Intake Structure

ISA-02

Description

Preventing a full or partial loss of SW for supporting Decay Heat Removal and Inventory and Pressure Control (SW-STNR-A and SW-STNR-B).

SW Train A and B are required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. CS is credited for Inventory Control, SPC is credited for Decay Heat Removal, REC provides cooling for RHR Pumps and the HPCI, RCIC, RHR, and CS Pump Rooms (Quads), and SW provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The complete loss or diversion of SW could challenge these NSPCs.

Fire in the area could result in physical damage to component cabling. Based on size of the structure and physical location within the structure, at least one Train would be available.

All SW Pumps and Strainers are located in Fire Zone 20A and at least one pump and strainer is required to achieve the NSPC function.

Quad cooling via REC is required to support CS. Quad heat up due to fire location in Fire Area IS-A is not credible. However, even if cooling is not required to the Quad, cooling to the RHR Heat Exchanger is still required for supporting SPC mode of operation. SW Train A and B may be credited to support various functions for NSPC depending on location of fire.

Cable damage to MTX1 (SW Strainer B) or MLX29 (SW Strainer A) results in a loss of the automatic features of the SW Train B and A Strainer. The loss of the automatic features could result in the clogging of the strainer resulting in a complete loss of SW flow.

This is a separation issue for Inventory and Pressure Control, Vital Auxiliaries, and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
IS-A	Intake Structure

ISA-03

Description

Preventing a full or partial loss of SW for supporting Decay Heat Removal and Inventory and Pressure Control. (SW Pump A and SW Pump D).

SW Trains A and B are required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. CS is credited for Inventory Control, SPC is credited for Decay Heat Removal, REC provides cooling for RHR Pumps and the HPCI, RCIC, RHR, and CS Pump Rooms (Quads), and SW provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The complete loss or diversion of SW could challenge these NSPC. Fire in the area could result in physical damage to component cabling. Based on size of the structure and physical location within the structure at least one Train would be available.

All SW Pumps and Strainers are located in Fire Zone 20A and at least one pump and strainer is required to achieve the NSPC function. Quad cooling via REC is required to support CS. Quad heat up due to fire location in Fire Area IS-A is not credible. However, even if cooling not required to the Quad, cooling to the RHR Heat Exchanger is still required for supporting SPC mode of operation. SW Train A and B may be credited to support various functions for NSPC depending on location of fire.

Cable damage to H401 (SW Pump A) or H531 (SW Pump D) results in a loss of power to the SW Pumps A or D. The loss of power to both would result in a complete loss of SW flow.

This is an issue for Inventory and Pressure Control and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
IS-A	Intake Structure

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
20A	Detection	Thermal Heat	R	Y	N	N	N	Y	N
20A	Detection	Ionization	R	Y	N	N	N	Y	N
20A	Detection	Flame	R	N	N	N	N	N	N
20A	Suppression	Total Flooding Halon 1301	F	N/A	N	N	N	Y	N
20B	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. In the event of normal operation, the Halon system will not adversely effect the other equipment operating within the Fire Zone as potential corrosion developing on equipment is a long term issue that would be significantly reduced by extensive post-discharge clean up. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-A	RCIC and Core Spray Pump Room

Fire Zone	Description
1A	RCIC and Core Spray Pump Room

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train B will be operated to maintain Suppression Pool temperature.	RBA-03
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, SW, RHR flow indications, and REC pressure indication [from Control Room]	RBA-01
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train B to maintain RPV level.	None
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-A	RCIC and Core Spray Pump Room	
Vital Auxiliaries	Mechanical: -REC will be supplied by Train B to provide the cooling supply to the ECCS. -SW Train B will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160V Bus 1G - 125/250 VDC Train B is available HVAC: - CS Train B - Quad area cooling - AC Switchgear Room 1G - Essential Control Building HVAC system - DC Switchgear Room 1B - Essential Control Building HVAC system - Battery Rooms 1B - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1B - Essential Control Building HVAC system	RBA-02

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Existing Engineering Equivalency Evaluations (EEEE)

None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-A	RCIC and Core Spray Pump Room

Variances from Deterministic Requirements (VFDR)

RBA-01

Description

Ability to monitor temperatures (Torus water) for SPC mode of operation of Decay Heat Removal (PC-TR-24).

The indication cable from Junction Box 307 to the Foxboro rack is damaged for all temperature indicators (1A-1H and 2A-2F), resulting in a loss of Torus water temperature indication in the Control Room. Loss of indication would result in the loss of ability to monitor Suppression Pool temperature for monitoring containment pressure as well as NPSH for the RHR pumps which take suction off the Suppression Pool.

This is a separation issue related to Process Monitoring.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-A	RCIC and Core Spray Pump Room

RBA-02

Description

Preventing a loss of SW cooling flow to the credited REC Heat Exchanger (SW-AOV-TCV451B).

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC.

The NSPC require a means of maintaining and controlling RPV water inventory post-fire. CS Train B is the credited train. SW-AOV-TCV451B controls the outlet flow from REC Heat Exchanger B. REC Train B is the credited train for supporting RHR Pump cooling, the credited RHR Pump for SPC, and Quad cooling for the RHR Pump and CS Pump Rooms. Cable damage to M923 could result in SW-AOV-TCV451B remaining energized, essentially locking the valve in the closed position, securing flow through the Heat Exchanger. Cable M923 is routed in a dedicated conduit in this area. If the valve fails to open, this causes a no-flow condition for SW through the credited REC Heat Exchanger, removing the REC ability to cool the credited RHR Pump for SPC and Quad cooling for the RHR and CS Pumps.

This is a separation issue for Vital Auxiliaries.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-A	RCIC and Core Spray Pump Room

RBA-03

Description

Preventing a loss of SW cooling flow to credited RHR Heat Exchanger or damage to RHRSW Booster Pumps due to run out (SW-MOV-MO89B).

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. SPC Train B is the credited train of Decay Heat Removal. The valve is normally closed and cable damage to M923 may cause the valve to spuriously open, or not allow it to open electrically. If the valve fully opens with the RHRSW Booster Pump operating, this may cause the pump to run out, damaging the pump. If the valve fails to open, this causes a no-flow condition for SW through the credited RHR Heat Exchanger, removing the ability to cool the credited SPC Train B for Decay Heat Removal.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-A	RCIC and Core Spray Pump Room

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
1A	Detection	Heat	R	N	N	N	N	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-B	Reactor Building South East Quad

Fire Zone	Description
1B	Core Spray Pump Room
1G	Control Rod Drive Pump Area

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A or B will be operated to maintain Suppression Pool temperature.	RBB-01
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, SW, RHR flow indications, and REC pressure indication [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train A to maintain RPV level.	None
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-B	Reactor Building South East Quad	
Vital Auxiliaries	Mechanical: -REC Train A or B will be operated to provide the cooling supply to the ECCS. -SW Train A or B will be operated to provide the cooling supply to the REC system and RHR Train A or B Heat Exchanger based on available SW Pump. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160V Bus 1F or 1G - 125/250 VDC Train A and B is available HVAC: - RCIC/CS Trains A and B - Quad area cooling - AC Switchgear Rooms - Essential Control Building HVAC system - DC Switchgear Rooms - Essential Control Building HVAC system - Battery Rooms - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Rooms - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-B	Reactor Building South East Quad

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire

Table B-3 Fire Area Transition

Fire Area	Description
RB-B	Reactor Building South East Quad

hazards of the areas, and adequate separation has been provided. In general, minor variations to the configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.

Basis

The bases that justify this conclusion are summarized as follows:

- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade.
- Ventilation systems can typically be used for smoke and heat removal.
- Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
- The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202.
- The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105.
- The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201.
- The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104.
- The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104.
- The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6.
- The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited for the acceptability of door R7.
- The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for

Table B-3 Fire Area Transition

Fire Area	Description
RB-B	Reactor Building South East Quad

the acceptability of doors R101 and R102.
• The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102.
• The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115.
• Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.

Variances from Deterministic Requirements (VFDR)

RBB-01

Description

Preventing loss of containment over-pressure in support of RHR Pump operation for SPC operation (PC-AOV-245AV).

The NSPC require a means of Decay Heat Removal (SPC Train A) post-fire. Both PC-AOV-245AV and PC-MOV-230MV cables are affected by the fire and may spuriously open. These valves isolate the Vent and Purge path from the Suppression Chamber and would result in a loss of containment over-pressure.

SPC Train A is credited for Decay Heat Removal. The loss of containment over-pressure could result in the loss of NPSH for the credited RHR Pump during the SPC mode of operation.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-B	Reactor Building South East Quad

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
1B	Detection	Heat	R	N	N	N	Y	Y	N
1G	Detection	Heat	R	N	N	N	Y	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

Fire Zone	Description
1C	RHR Pump Room 1A and 1C
2A-2	CRD Units - North
2A-3	903' South Corridor
2B	RHR Heat Exchanger Room A

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train B will be operated to maintain Suppression Pool temperature.	RBCF-05 RBCF-06 RBCF-07 RBCF-09 RBCF-10
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indications [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train B to maintain RPV level.	RBCF-01 RBCF-02 RBCF-03 RBCF-04 RBCF-08 RBCF-09
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A	
Vital Auxiliaries	Mechanical: - REC will be supplied by SW Train B to provide the cooling supply to the ECCS. - SW Train B will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160G Bus - 125/250 VDC Train B is available HVAC: - DG/CS Train B - Quad area cooling - DG 2 HVAC system - AC Switchgear Room 1G - Essential Control Building HVAC system - DC Switchgear Room 1B - Essential Control Building HVAC system - Battery Room 1B - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1B - Essential Control Building HVAC system	RBCF-11

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6,

Table B-3 Fire Area Transition

Fire Area		Description
RB-CF		Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A
		R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire hazards of the areas, and adequate separation has been provided. In general, minor variations to the configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.
	Basis	The bases that justify this conclusion are summarized as follows: • Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas. • Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade. • Ventilation systems can typically be used for smoke and heat removal. • Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations. • The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202. • The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105. • The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201. • The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307. • The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104. • The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103. • The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104. • The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6. • The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A
	for the acceptability of door R7. • The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for the acceptability of doors R101 and R102. • The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102. • The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115. • Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

Variances from Deterministic Requirements (VFDR)

RBCF-01

Description

Loss of RPV water inventory to Reactor Building Sumps and Radwaste (CRD-SOV-SO31A and CRD-SOV-SO31B).

The NSCA require a means of isolating the RPV post-fire to ensure sufficient water inventory. Cable damage may result in the spurious opening of CRD-SOV-SO31A and CRD-SOV-SO31B. This opening of the CRD Scram Discharge Volume Vent and Drain Valve SOVs would result in loss of RPV Inventory to the Reactor Building Sumps and then to Radwaste. Based on CNS-PSA-007, the CRD seal leakage flow rate is assumed to be 450 to 600 gpm initially, reducing to 73 gpm after 70 minutes, and down to 40 gpm after 4 hours, based on all CRD seals leaking at the same time at the maximum amount prior to repair.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

RBCF-02

Description

Preventing a RR Pump Seal LOCA due to inability to secure the RR Pump from the Control Room with the REC Non-Critical Header secured (RR Pump Breaker for 4160C-1CS).

Breaker F/FDR to the 4160V Bus from the Startup Transformer: This is a normally available, required open breaker that provides motive power to the RR Pump A. The RR Pumps are required to trip to prevent a potential RR Pump Seal LOCA when REC is not available to provide cooling. This would challenge the NSPC for Inventory Control. Remote operation of the breaker from the Control Room is lost due to cable, H254 (1C Bus) having fire-induced damage.

NOTE: REC Non-Critical Header is secured in this fire area to address potential containment over-pressure.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

RBCF-03

Description

Preventing an RPV Overfill due to the inability to close the inboard or outboard steam supply valves to the HPCI Turbine (HPCI-MOV-MO14-Passive and HPCI-MOV-MO16).

HPCI-MOV-MO14, MO15, and MO16 are all impacted by fire in this area.

Spurious opening of the HPCI-MOV-MO16 (HPCI Steam Admission Valve) due to cable damage on HP185 would result in rapid RPV overfill condition. This VFDR concern is limited to fires occurring at RB 903' Elevation. HPCI is not credited in this area for NSCA success. Core Spray Train B is available automatically.

Cable damage to DC190 and DC191 can cause HPCI-MOV-MO16 to not be able to be closed from the Control Room. This failure would result in the rapid RPV overfill condition if HPCI-MOV-MOV14 spuriously opens. This VFDR concern is limited to fires occurring at Reactor Building 859' and 881' Elevations. HPCI is not credited in this area for NSCA success. Core Spray Train B is available automatically.

This is a separation issue in isolating the RPV for Inventory and Pressure Control.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

RBCF-04

Description

Ability to close SRVs automatically/remotely from the Control Room could be lost due to cable fire damage (Pilot Valves SPV71A, SPV71B, SPV71C, and SPV71D).

The NSPC require the ability to isolate the RPV to maintain water inventory above the active fuel. Spurious operations could result in the opening of up to four ADS valves. The NSPC could be challanged if the affected ADS valves are not returned to their fail-safe closed position within 18 minutes for single spurious operation.

NOTE: Cables causing spurious operations are only impacted during Fire Zones 2A-2 and 2A-3 full zone burn out.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

RBCF-05

Description

Preventing loss of containment over-pressure in support of RHR Pump operation for SPC operation (PC-MOV-231MV, RW-AOV-AO82, and RW-AOV-AO94).

NSPC require a means of Decay Heat Removal post-fire. SPC Train B is credited for Decay Heat Removal. The loss of containment over-pressure could result in the loss of NPSH for the credited RHR pump during SPC mode of operation.

Both PC-AOV-AO246 and PC-MOV-231MV cables are affected by the fire and may spuriously open. These valves isolate the Vent and Purge path from the Drywell, and would result in a loss of containment over-pressure.

RW-AOV-AO82, RW-AOV-AO83, RW-AOV-AO94, and RW-AOV-AO95 cables are all affected by the fire, and may spuriously open. These valves isolate the Drywell Equipment and Floor Drain Sump discharge paths and would result in a loss of containment over-pressure.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

RBCF-06

Description

Preventing damage to credited RHR pump / loss of Decay Heat Removal due to spurious closure of the minimum flow recirculation line for RHR Train B (RHR-MOV-MO16B).

RHR-MOV-MO16B could spuriously close based on equipment and cable failures. The NSPC require a means of Decay Heat Removal post-fire. RHR Train B is credited to support SPC mode of RHR following a fire in this area. Failures associated with RHR-Logic-B may result in spurious starting of RHR Pump 1D. Spurious closure of RHR-MOV-MO16 concurrent with a running RHR Pump (1D) would result in potential pump damage due to overheating, and loss of SPC Train B.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

RBCF-07

Description

Preventing flow diversion during SPC mode of RHR for Decay Heat Removal (RHR-MOV-MO20-Passive and RHR-MOV-MO27B).

The NSPC require a means of Decay Heat Removal post-fire. SPC Train B is the credited train of Decay Heat Removal. RHR-MOV-MO20 is normally closed, and is desired closed during SPC mode of operation. Based on cable failure, the valve may spuriously open, resulting in a 20-inch flow diversion through the bypass line. This results in a loss or reduction of flow during the SPC Train B mode of operation.

Spurious opening of RHR-MOV-MO25B, and inability to close normally open RHR-MOV-MO27B (20-inch flow diversion path) would result in loss of SPC Train B. An IN 92-18 concern (MBR2) may prevent ability to close RHR-MOV-MO25B and isolate the flow diversion.

This is a separation issue for Decay Heat Removal.

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

RBCF-08

Description

Prevent loss of RPV water inventory and, or high temperature and pressure in the low pressure portion of the RWCU system from the Control Room due to cable fire damage (RWCU-MOV-MO18).

The NSPC require a means of isolating the RPV and low pressure systems from the RCPB post-fire. The inability to close RWCU-MOV-MO18 or RWCU-MOV-MO15 may result in loss of RPV water inventory to the RWCU system or cause high temperatures and pressures in the low pressure portion of piping downstream.

NOTE: Fire modeling indicates a fixed ignition source could potentially disable RWCU-MOV-MO18 due to cable failure and also affect normal power to MCC R, resulting in inability to close RWCU-MOV-MO15 from the Control Room. For all other fire scenarios RWCU-MOV-MO18 will be available to close from the Control Room.

This is a separation issue for Inventory and Pressure Control.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Modification.

Modification: Item S-1.1 of LAR Attachment S, Table S-1.

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

RBCF-09

Description

Preventing a full or partial loss of SW for supporting Decay Heat Removal and Inventory and Pressure Control from the Control Room with SW supplying REC (REC-MOV-694MV, REC-MOV-695MV, REC-MOV-697MV, REC-MOV-698MV, REC-MOV-711MV-Passive, SW-MOV-37MV, SW-MOV-886MV-Passive, SW-MOV-887MV, SW-MOV-888MV-Passive, and SW-MOV-889MV).

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. CS Train B is credited for Inventory Control, SPC Train B is credited for Decay Heat Removal, REC Train B provides cooling for RHR Pump 1D and the RHR and CS Pump Rooms (Quads), and SW Train B provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The complete loss or diversion of SW to REC could challenge these NSPC.

Based on fire location on the 903' Elevation, heat up of the Quads is not expected due to the fire location.

Based on cable failures, the REC and SW systems need to be manually aligned for SW Train B to supply REC Train B (Critical Header only), and keep pressure on the REC Train A Header.

Cable damage (MY344) for SW-MOV-37MV could remove the ability to close the valve if already open. This would divert flow to non-critical loads and the non-credited SW Train with only a single SW Pump available. This reduction in flow may not allow sufficient cooling to REC and RHR Heat Exchangers. The valve splits the SW Train B from SW Train A and the non-essential loads.

This is a separation issue for Inventory and Pressure Control and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

RBCF-10

Description

Preventing a loss of SW cooling flow to credited RHR Heat Exchanger or damage to RHRSW Booster Pumps due to run-out (SW-MOV-MO89B).

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for the NSPC.

SPC Train B is the credited train of Decay Heat Removal. SW-MOV-MO89B is normally closed, and with damage to RHR-Logic B, control from the Control Room or automatic operation would not be available. The failure of this valve to open results in a no-flow condition for SW through the credited RHR Heat Exchanger. This removes the ability to cool the credited SPC Train B for Decay Heat Removal.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

RBCF-11

Description

Loss of G Bus powered Switchgear Cooling Fans due to cable damage (EE-CB-4160F-1FS and EE-CB-4160F-SS1F).

The Essential Control Building HVAC is required to support the credited train of electrical power. Division II fans (SF and EF) are unavailable due to potential cable damage (MRB7). Division I fans are unavailable due to cable failures opening EE-CB-4160F-SS1F and EE-CB-4160F-1FS.

This is a separation issue related to Vital Auxiliaries, Essential Control Building HVAC.

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
1C	Detection	Heat	R	N	N	N	Y	Y	N
2A-2	Detection	Ionization	R	N	N	N	N	Y	N
2A-3	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
2B	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	Description
RB-CF	Reactor Building Northwest Quad, Reactor Building 903 North Area and South Corridor, RHR Heat Exchanger Room A

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

Fire Zone	Description
1D	RHR Pump Room 1B and 1D
1E	HPCI Pump Room
2A-3	903' South Corridor
2C	CRD Units - South
2D	RHR Heat Exchanger Room B

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A will be operated to maintain Suppression Pool temperature	RBDI-06 RBDI-07 RBDI-08
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room with OMA] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indications [from Control Room]	RBDI-05
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train A to maintain RPV level.	RBDI-01 RBDI-02 RBDI-03 RBDI-04 RBDI-06
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B	
Vital Auxiliaries	Mechanical: - REC will be supplied by SW Train A to provide the cooling supply to the ECCS. - SW Train A will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): -Offsite Emergency Transformer aligned to 4160F Bus -125/250 VDC Train A is available HVAC: - RCIC/CS Train A - Quad area cooling - DG 1 HVAC system - AC Switchgear Room 1G - Essential Control Building HVAC system - DC Switchgear Room 1A - Essential Control Building HVAC system - Battery Room 1A - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1A - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire

Table B-3 Fire Area Transition

Fire Area	Description
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

hazards of the areas, and adequate separation has been provided. In general, minor variations to the configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.

Basis

The bases that justify this conclusion are summarized as follows:

- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade.
- Ventilation systems can typically be used for smoke and heat removal.
- Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
- The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202.
- The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105.
- The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201.
- The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104.
- The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104.
- The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6.
- The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited for the acceptability of door R7.
- The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for

Table B-3 Fire Area Transition

Fire Area	Description
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B
	the acceptability of doors R101 and R102. • The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102. • The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115. • Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

Variances from Deterministic Requirements (VFDR)

RBDI-01

Description

Loss of RPV water inventory to Reactor Building Sumps and Radwaste (CRD-SOV-SO31A and CRD-SOV-SO31B).

Cable damage may result in the spurious opening of CRD-SOV-SO31A and CRD-SOV-SO31B. This opening of the CRD Scram Discharge Volume Vent and Drain Valve SOVs would result in loss of RPV Inventory to the Reactor Building Sumps and then Radwaste. The NSCA require a means of isolating the RPV post-fire to ensure sufficient water inventory. Based on CNS-PSA-007, the CRD seal leakage flow rate is assumed to be 450 to 600 gpm initially, reducing to 73 gpm after 70 minutes, and down to 40 gpm after 4 hours, based on all CRD seals leaking at the same time at the maximum amount prior to repair.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

RBDI-02

Description

Preventing a RR Pump Seal LOCA due to inability to secure the RR Pumps from the Control Room with the REC Non-Critical header secured (RR Pump Breakers for 4160C-1CS and 4160D-1DS).

Breakers F/FDR to the 4160V Bus from the Startup Transformer: These are normally available, required open breakers that provides motive power to the RR Pumps. The RR Pumps are required to trip to prevent a potential seal LOCA when REC is not available to provide cooling. This would challenge the NSPC for Inventory Control. REC is secured in this fire area to address a potential loss of containment over-pressure. Remote operation of the breaker(s) from the Control Room is lost due to cables H254 (1C bus) and H294 (1D bus) fire-induced damage.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Table B-3 Fire Area Transition

Fire Area		Description
RB-DI		Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B
	Disposition	A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria. Risk: Acceptable with Recovery Action. See LAR Attachment G, Table G-1 for Recovery Action details. Modification: None
RBDI-03		
	Description	Preventing an RPV Overfill due to the inability to close the inboard or outboard steam supply valves to the HPCI Turbine (HPCI-MOV-MO14-Passive and HPCI-MOV-MO15). HPCI-MOV-MO14, HPCI-MOV-MO15, and HPCI-MOV-MO16 are all impacted in this area. Spurious opening of HPCI-MOV-MO15 (HPCI Steam Admission Valve) due to cable damage on HP185, would result in a rapid RPV overfill condition. This VFDR concern is limited to fires occurring at RB 903' Elevation. HPCI is not credited in this area for NSCA success. Core Spray Train A is available. Cable damage can cause HPCI-MOV-MO15 to not be able to be closed from the Control Room. This failure would result in the rapid RPV overfill condition if HPCI-MOV-MO14 spuriously opens. This VFDR concern is limited to fires occurring at the Reactor Building 859' and 881' Elevations. HPCI is not credited in this area for NSCA success. Core Spray Train A is available. This is a separation issue in isolating the RPV for Inventory and Pressure Control.
	Disposition	A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria. Risk: Acceptable Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

RBDI-04

Description

Ability to close SRV's automatically / remotely from the Control Room is lost due to cable fire damage (Pilot Valves SPV71A, SPV71B, SPV71C, SPV71D, SPV71E, SPV71F, SPV71G, and SPV71H).

The NSPC require the ability to isolate the RPV to maintain water inventory above the active fuel. Spurious operations could result in the opening of up to four ADS valves. SPV71A, SPV71B, SPV71C, and SPV71D utilize penetration X-100-G and SPV71E, SPV71F, SPV71G, and SPV71H utilize penetration X-100-A. Based on physical separation of these penetrators, at least one should survive limiting the total number of valves affected at any one time to 4 of the 8. The NSPC could be challanged if the affected ADS valves are not returned to their fail-safe closed position within 18 minutes for single spurious operation.

NOTE: SPV71A, SPV71B, SPV71C, and SPV71D spurious cables are only impacted by a full zone burn out of Fire Zone 2A-3 in this area. SPV71E, SPV71F, SPV71G, and SPV71H are impacted in Fire Zone 2C scenarios and within full zone burnout in this area.

This is a separation issue related to Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

RBDI-05

Description

Loss of RPV water level and pressure along with Torus level due to the fire affects on sensing line piping (NBI-LT-52A, NBI-LT-52C, NBI-LT-59A, NBI-LT-59C, NBI-LT-91A, NBI-LT-91C, NBI-PT-53A, and NBI-PT-53C).

The NSPC require a means of monitoring plant conditions post-fire. While instruments are not located within this fire area, their sensing lines are affected by the fire, making indication unreliable. Unaffected instruments need to be isolated from the common cold reference leg continuous backfill supply.

This is a separation issue for Process Monitoring.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

RBDI-06

Description

Preventing a full or partial loss of SW for supporting Decay Heat Removal and RPV Inventory Control from the Control Room with SW supplying REC (REC-MOV-714MV-Passive, SW-MOV-887MV, and SW-MOV-889MV).

SW Train A is required to support various functions (SPC, RCIC and CS Quad cooling, REC, DGs, etc) for the NSPC. CS Train A is credited for RPV Inventory Control, SPC Train A is credited for Decay Heat Removal, REC-A provides cooling for RHR Pump 1A and the RHR and CS Pump Rooms (Quads), and SW Train A provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The complete loss or diversion of SW to REC could challenge these NSPC.

Current requirements indicate 1 hour to establish room cooling.

NOTE: Based on fire location on the 903' Elevation, additional heat up of the Quad due to contribution of the fire is not expected due to the fire location.

Based on cable failures, the REC and SW systems need to be aligned for SW Train A to supply the REC Train A Critical Header only, and keep pressure on the REC Train B Critical Header.

This is a separation issue for Inventory and Pressure Control and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

RBDI-07

Description

Preventing a loss of Decay Heat Removal for credited SPC Train A by loss of SW cooling flow to the credited RHR Heat Exchanger or Diversion of RHR flow in SPC mode of operation (SW-MOV-MO89A and RHR-MOV-MO27A).

The NSPC require a means of Decay Heat Removal post-fire. SPC Train A is the credited source of Decay Heat Removal for this area. SW-MOV-MO89A is normally closed, and damage to RHR-Logic A automatic operations or control from the Control Room would not be available. This valve failing to open results in a no-flow condition for SW through the credited RHR Heat Exchanger. This removes the ability to cool the credited SPC Train A for Decay Heat Removal.

RHR-MOV-MO27A is a normally open valve, and the injection line is isolated by RHR-MOV-MO25A. RHR-MOV-MO25A fails open in this area, and based on cable damage, neither RHR-MOV-MO25A nor RHR-MOV-MO27A can be closed remotely to isolate the injection line. The spurious opening of RHR-MOV-MO25A, along with inability to close either valve, results in a 20-inch diversion to the SPC Train A flow path. An IN 92-18 concern may prevent ability to close RHR-MOV-MO25A and isolate the flow diversion.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

RBDI-08

Description

Preventing loss of containment over-pressure in support of RHR pump operation for SPC operation (PC-AOV-245AV).

The NSPC require a means of Decay Heat Removal post-fire. SPC Train A is credited for Decay Heat Removal. The loss of containment over-pressure could result in the loss of NPSH for the credited RHR pump during SPC mode of operation.

Both PC-AOV-245AV and PC-MOV-230MV cables are affected by the fire, and may spuriously open. These valves isolate the Vent and Purge path from the Suppression Chamber, and would result in a loss of containment over-pressure.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
1D	Detection	Heat	R	N	N	N	N	Y	N
1E	Detection	Heat	R	N	N	N	N	Y	N
2A-3	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
2C	Detection	Ionization	R	N	N	N	N	Y	N
2C	Detection	Heat	R	Y	N	N	N	N	N
2C	Suppression	Preaction Sprinkler System	P	N/A	N	N	N	N	N
2D	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	Description
RB-DI	Reactor Building Southwest Quad, HPCI Room, Reactor Building 903 South, RHR Heat Exchanger Room B

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. In the event of normal operation, the pre-action system will not adversely effect the other equipment operating within the Fire Zone. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-E	Suppression Pool Area

Fire Zone	Description
1F	Suppression Pool Area

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A will be operated to maintain Suppression Pool temperature	RBE-01 RBE-03
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level [from Control Room] - Suppression Pool temperature [locally] - CS, RHR, and SW flow indications [from Control Room]	RBE-02
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train A to maintain RPV level.	RBE-03
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-E	Suppression Pool Area	
Vital Auxiliaries	Mechanical: - REC will be supplied by Train A to provide the cooling supply to the ECCS. - SW Train A will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160F Bus - 125/250 VDC Train A is available HVAC: - DG /CS Train A - Quad area cooling - DG 1 HVAC system - AC Switchgear Room 1G - Essential Control Building HVAC system - DC Switchgear Room 1A - Essential Control Building HVAC system - Battery Room 1A - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1A - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-E	Suppression Pool Area

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire

Table B-3 Fire Area Transition

Fire Area	Description
RB-E	Suppression Pool Area

hazards of the areas, and adequate separation has been provided. In general, minor variations to the configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.

Basis

The bases that justify this conclusion are summarized as follows:

- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade.
- Ventilation systems can typically be used for smoke and heat removal.
- Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
- The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202.
- The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105.
- The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201.
- The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104.
- The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104.
- The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6.
- The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited for the acceptability of door R7.
- The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for

Table B-3 Fire Area Transition

Fire Area	Description
RB-E	Suppression Pool Area
	the acceptability of doors R101 and R102. • The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102. • The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115. • Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.

Table B-3 Fire Area Transition

Fire Area	Description
RB-E	Suppression Pool Area

Variances from Deterministic Requirements (VFDR)

RBE-01

Description

Preventing loss of containment over-pressure in support of RHR pump operation for SPC operation (PC-AOV-245AV, RW-AOV-AO82, and RW-AOV-AO94).

The NSPC requires a means of Decay Heat Removal post-fire. SPC Train A is credited for Decay Heat Removal. The loss of containment over-pressure could result in the loss of NPSH for the credited RHR pump during the SPC mode of operation.

RW-AOV-AO82, RW-AOV-AO83, RW-AOV-AO94, and RW-AOV-AO95 cables are all affected by the fire and may spuriously open. These valves isolate the Drywell Equipment and Floor Drain Sump discharge path and would result in a loss of containment over-pressure.

Both PC-AOV-245AV and PC-MOV-230MV cables are affected by the fire and may spuriously open. These valves isolate the Vent and Purge path from the Suppression Chamber, and would result in a loss of containment over-pressure.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-E	Suppression Pool Area

RBE-02

Description

Preventing the loss of Torus level and temperature due to the fire affects on sensing line piping and temperature element signal cable damage (PC-LR-1A and PC-TR-24).

The NSCA require a means of monitoring Torus level. PCDPT-3A1 and PC-DPT-3B2 are affected, due to sensing lines located in the Torus fire area. The PCDPT-3A1 sensing line is located in the Northeast quadrant of the Torus. The PC-DPT-3B2 sensing line is located in the Southeast quadrant of the Torus. A separation of approximately 75 feet exists between the redundant sensing lines.

NOTE: Fire scenarios in RB-E consist of a non-ventilated electrical panel and transient fires. Based on NEDC 09-101, "EPM Report R1906-711-RB - Detailed Fire Modeling Report Fire Compartment RB," there are no secondary combustibles that would be involved in the fires postulated in this fire compartment. Therefore, the zone of influence of the fire scenarios in RB-E are not large enough to result in damage to targets located 75 feet apart.

The NSCA require a means of monitoring Torus temperature. PC-TR-24 provides Torus temperature indication for Primary Containment.

This is a separation issue for Process Monitoring.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-E	Suppression Pool Area

RBE-03

Description

Preventing a full or partial loss of SW for the Reactor Building in support of Decay Heat Removal (RHR Heat Exchanger) and Inventory and Pressure Control (REC Heat Exchanger) from the Control Room or automatically due to cable and equipment damage (SW-AOV-TCV451A and SW-MOV-89A).

SW Train A is required to support various functions (SPC, RCIC and CS Quad cooling, REC, DGs, etc) for NSPC. CS Train A is credited for Inventory and Pressure Control, SPC Train A is credited for Decay Heat Removal, REC Train A provides cooling for RHR Pump 1A and the RHR and CS Pump Rooms (Quads) and SW Train A provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The complete loss of SW to the Reactor Building could challenge these NSPC.

SW-MOV-MO89A is normally closed, and cable damage (M920 and M921) to this valve may cause spurious operation to open, with automatic operation or control from the Control Room would not be available. This valve failing to open results in a no-flow condition for SW through the credited RHR Heat Exchanger. This removes the ability to cool the credited SPC Train A for Decay Heat Removal.

CS Train A is the credited train. SW-AOV-TCV451A controls the outlet flow from the REC Heat Exchanger A. REC Train A is the credited train for supporting RHR Pump cooling, and the credited RHR Pump for SPC and Quad cooling for the RHR Pump and CS Pump Rooms. Cable damage to M920, M921, and REC-TIC- 451A could result in SW-AOV-TCV451A remaining energized, essentially locking the valve in the closed position, securing flow through the Heat Exchanger. Cable M920, M921, and M923 are routed in a dedicated conduit in this area. If the valve fails to open, this causes a no-flow condition for SW through the credited REC Heat Exchanger removing the REC ability to cool the credited RHR Pump for SPC and Quad cooling for the RHR Pump and CS Pump Rooms.

This is a separation issue for Inventory and Pressure Control and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-E	Suppression Pool Area

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
1F	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. The Suppression Pool area of the Reactor Building on the 859' and 881' Elevations are not subject to adverse effects to equipment by water intrusion from fire suppression systems. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

Fire Zone	Description
2A-1	Reactor Building 903' Northeast Corner

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train B will be operated to maintain Suppression Pool temperature.	RBFN-01 RBFN-08 RBFN-09
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - Emergency Condensate Storage Tank level [from Alternate Shutdown Panel]	RBFN-12 RBFN-13
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection is provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. The HPCI system will be used to control RPV pressure and to maintain RPV level.	RBFN-01 RBFN-02 RBFN-04 RBFN-05 RBFN-06 RBFN-07 RBFN-10 RBFN-11
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-FN	Reactor Building 903' Northeast Corner	
Vital Auxiliaries	Mechanical: - REC will be supplied by SW Train B to provide the cooling supply to the ECCS in the alternate shutdown lineup. - SW Train B will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers in the alternate shutdown lineup. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160G Bus - 125/250 VDC Train B is available HVAC: - HPCI/CS Train B - Quad area cooling - AC Switchgear Room 1G - Essential Control Building HVAC system - DC Switchgear Room 1B - Essential Control Building HVAC system - Battery Room 1B - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1B - Essential Control Building HVAC system	RBFN-01 RBFN-03 RBFN-13

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 05-034 - Evaluation of the Reclassification of Door R104 Under the FHA

Purpose The purpose of this evaluation is to permanently reclassify door R104 from a 3-hour fire door to a 1-hour fire door required to separate Fire Area RB-FN/Fire Zone 2A-1 (Reactor Building 903'-6" Northeast Corner) and Fire Area TB-C/Fire Zone 2E (Steam Tunnel).

Conclusion It is acceptable to revise the fire barrier between Fire Area RB-FN/Fire Zone 2A-1, and Fire Area TB-C/Fire Zone 2E from a 3-hour rated barrier to a 1-hour fire rated barrier.

Basis The bases that justify this conclusion are summarized as follows:

- Ignition sources near door R104 include, but are not limited to: MCC-K, RCIC Rack, and Condensate Pumps CP-R-A1 and A2. Each of these ignition sources is spatially separated from door R104 by approximately 5 feet or more. In the case of MCC-K, as much as approximately 15 feet. A postulated fire at each of these ignition sources would propagate upward into the cable raceways. Horizontal propagation will not occur due to the scarcity and discontinuity of combustibles in the zone. Without horizontal propagation of the postulated fire, no challenge can be made to door R104.
- The combustible loading calculation, NEDC 93-161, shows that the primary contributor of combustibles in Reactor Building Elev. 903'-6" are located 20 feet above the floor in the overhead cable raceways. These cables are fire retardant and are equivalent to IEEE-383 rated cable. Fire in the cable raceways would not present a fire severity of greater than 1 hour to door R104 due to a spatial separation of 10 feet or more.
- This area is also equipped with fire and smoke detection as well as fire suppression. The smoke and fire detections systems would ensure immediate fire brigade response. The fire suppression system alone will mitigate any postulated fire at the floor level, well before the fire reaches a severity of greater than 1 hour.

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire hazards of the areas, and adequate separation has been provided. In general, minor variations to the configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.

Basis

The bases that justify this conclusion are summarized as follows:

- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade.
- Ventilation systems can typically be used for smoke and heat removal.
- Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
- The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202.
- The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105.
- The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201.
- The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104.
- The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104.
- The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6.
- The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited for the acceptability of door R7.
- The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for the acceptability of doors R101 and R102.
- The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner
	Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102. • The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115. • Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-FN	Reactor Building 903' Northeast Corner

Variances from Deterministic Requirements (VFDR)

RBFN-01

Description

Preventing a full or partial loss of SW for supporting Decay Heat Removal and RPV Inventory and Pressure Control with SW supplying REC (EE-CB-4160GSWP1D, REC-MOV-695MV, REC-MOV-694MV, REC-MOV-698MV, SW-AOV-TCV451B, SW-FI-132B, SW-MOV-37MV, SW-MOV-651MV, SW-MOV-887MV, and SW-MOV-889MV).

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. HPCI is credited for RPV Inventory Control, SPC Train B is credited for Decay Heat Removal, REC Train B provides cooling for RHR Pump 1B and the RHR Room (Quads) and SW Train B provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The complete loss or diversion of SW to REC could challenge these NSPC.

Based on cable failures, the REC and SW systems are aligned to SW Train B to supply the REC Train B Critical Header. This is a separation issue for Vital Auxiliaries for which:

•Based on cable failures, EE-CB-4160G-SWP1D may trip open.
•REC-MOV-694MV is normally open and presents a flow diversion during REC operation supplied from SW. Cable damage removes the ability to remotely close the valve from the Control Room.
•REC-MOV-698MV is normally open and presents a flow diversion from the REC Train B Critical Header. Cable damage removes the ability to remotely close the valve from the Control Room.
•REC-MOV-695MV cable failure removes the ability to remotely close the valve from the Control Room.
•Cable damage may cause SW-AOV-TCV451B to fail closed, securing the SW Train B discharge back to the river and removing cooling flow to the HPCI and RHR Pumps.
•Cable damage may cause SW-FI-132B to result in SW flow indication not being available in the Control Room. However, indication remains available at the ASD Panel.
•SW-MOV-37MV may need to be closed to isolate the diversion of SW flow from the critical SW and REC loads to the non-critical SW loads with only a single operating SW Pump.
•SW-MOV-651MV needs to be closed to prevent flow diversion from the SW system to the REC Heat Exchanger, which is bypassed in ASD lineup.
•SW-MOV-887MV needs to be open to supply SW to the REC Critical Header.
•SW-MOV-889MV needs to be open to supply SW to the REC Critical Header.

This is a separation issue for Vital Auxiliaries, Inventory and Pressure Control, and Decay Heat Removal.

Table B-3 Fire Area Transition

Fire Area		Description
RB-FN		Reactor Building 903' Northeast Corner
	Disposition	A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria. Risk: Acceptable Modification: None
RBFN-02		
	Description	Ability to close MSIVs automatically / remotely from the Control Room is lost due to cable fire damage. MSIVs (MS-AOV-AO80A, MS-AOV-AO80B, MS-AOV-AO80C, and MS-AOV-AO80D) may not go closed from the Control Room due to cable damage. The NSPC requires a means of isolating the RPV to ensure sufficient water inventory for Decay Heat Removal. This is a separation issue related to Inventory and Pressure Control, RPV Isolation.
	Disposition	A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria. Risk: Acceptable Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

RBFN-03

Description

Establish vital auxiliaries and control by powering the credited 4160G Bus from the Emergency Transformer (EE-CB-4160F-1FS, UNDERVOLTAGE-BUS-G, EE-CB-4160G-1GB, EE-CB-4160G-1GE, EE-CB-4160G-1GS, EE-CB-4160G-SS1G, EE-CB-4160G-RSWP1B-OCT, and EE-CB-4160G-SWP1B-OCT).

The NSCA require at least one source of AC power available. Offsite power from the Emergency Transformer is unavailable using the normal transfer mode due to fire damage to supply breakers feeding the 4160G Bus.

There are multiple breaker and cables affected on the 4160F Bus. Multiple cables that support EE-CB-4160F-1FS are also damaged due to fire and may close the breaker.

Due to cable damage on the UV circuit for the 4160G Bus, in addition to cable damage listed below, all breakers need to be operated.

• The EE-CB-4160G-1GB breaker is normally closed, and should open as part of the transfer process due to loss of Startup Transformer, but cable damage to H561 may not allow it to open. The DG and Emergency Transformer are not rated to normally carry both the vital and non-vital buses. Additionally, the DG2/1GS breakers are interlocked with the BG/GB breakers, and will not close if the breaker is not opened.

•Cable damage (H572, H573) due to fire could close EE-CB-4160G-1GE or blow its control power fuses. The breaker is normally closed and needs to be open to allow for control of the DG, and not allow damage to the DG or the bus if the DG came onto the bus out-of-phase.

•Based on cable failure (H551, H552, H553, H555) due to fire damage, the 4160G-1GS Breaker may trip open or blow control power fuses and not close.

•Cable damage to H542 and H543 may cause EE-CB-4160G-SS1G to trip open. Opening this breaker causes a loss of power to the 480V G Bus. The breaker is normally closed and is desired closed to support NSPC requirements for Process Monitoring and Decay Heat Removal.

•EE-CB-4160G-RSWP1B-OCT may not trip due to potential fire damage to the control and load cables, resulting in loss of the credited 4160G Bus.

Table B-3 Fire Area Transition

Fire Area | **Description**

RB-FN | Reactor Building 903' Northeast Corner

•EE-CB-4160G-RSWP1D-OCT may not trip due to potential fire damage to the control and load cables, resulting in loss of the credited 4160G Bus.

•EE-CB-4160G-SWP1B-OCT may not trip due to potential fire damage to the control and load cables, resulting in loss of the credited 4160G Bus.

Therefore, the above OCT circuits are needed to ensure that the credited critical train of power (4160G) can remain free of damage based on cable protection or modification to re-route impacted cables to the load or control circuitry.

This is a separation issue for Vital Auxiliaries.

Disposition A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

RBFN-04

Description Ability to close SRV's automatically / remotely from the Control Room is lost due to cable fire damage (Pilot Valves SPV71A, SPV71B, SPV71C, SPV71D, SPV71E, SPV71F, SPV71G, and SPV71H).

The NSCA require the ability to isolate the RPV to maintain water inventory above the top of active fuel. Spurious operations could result in the opening of up to eight (8) ADS valves. The NSPC could be challanged if the affected ADS valves are not returned to their fail-safe closed position within 18 minutes for single spurious operation.

This is a separation issue related to Inventory and Pressure Control, RPV Isolation.

Disposition A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

RBFN-05

Description

Preventing a loss of Inventory and Pressure Control by losing remote operation of the HPCI system (HPCI-ECCS, HPCI-FAN-GSE, HPCI-FIC-108, HPCI-MOV-MO14, HPCI-MOV-MO15, HPCI-MOV-MO15-PASSIVE, HPCI-MOV-MO16-PASSIVE, HPCI-MOV-MO17, HPCI-MOV-MO19, HPCI-MOV-MO20, HPCI-MOV-MO21, HPCI-MOV-MO24, HPCI-MOV-MO25, HPCI-MOV-MO58, HPCI-P-ALOP, HPCI-P-CP, HPCI-PI-109, HPCI-PI-111, HPCI-PI-112, HPCI-PI-116, HPCI-SI-2792, and HPCI-TU-TURB).

Based on cable damage to valves, valve control, indication, and the HPCI Turbine, the system may fail to initiate automatically from the Control Room or not be able to be controlled if it does initiate. The ASD portion of the system is unaffected by a fire in this area.

This is a separation issue related to Inventory and Pressure Control.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

RBFN-06

Description

Preventing loss of remote valve control for essential systems from the ASD Room (EE-MCC-R-1A).

The NSPC require a mean of RPV inventory and pressure control post-fire. Feeder cable from MCC K (MK134) is affected by the fire, and MCC-R is the power supply to HPCI-MOV-MO15-ASD, MS-MOV-MO74, REC-MOV-695MV, RHR-MOV-MO20, RHR-MOV-MO57-ASD, and RWCU-MOV-MO15. The normal position for HPCI-MOV-MO15 is open and the desired position is open, with HPCI as the credited train of inventory control. HPCI-MOV-MO15 may spuriously close prior to shifting to the ASD Room.

This is a separation issue for Inventory and Pressure Control.

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

RBFN-07

Description

Loss of RPV water inventory to Reactor Building Sumps and Radwaste (CRD-SOV-SO31A and CRD-SOV-SO31B).

Cable damage may result in the spurious opening of CRD-SOV-SO31A and CRD-SOV-SO31B. This opening of the CRD Scram Discharge Volume Vent and Drain Valve SOVs would result in loss of RPV Inventory to the Reactor Building Sumps and then Radwaste. The NSCA require a means of isolating the RPV post-fire to ensure sufficient water inventory. Based on CNS-PSA-007, the CRD seal leakage flow rate is assumed to be 450 to 600 gpm initially, reducing to 73 gpm after 70 minutes, and down to 40 gpm after 4 hours, based on all CRD seals leaking at the same time at the maximum amount prior to repair.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

RBFN-08

Description

Loss of Decay Heat Removal, due to flow diversion, loss of flow path, loss of pump in the RHR system or loss of cooling from the SW system due to cable damage (RHR-FI-133B, RHR-MOV-MO12B-PASSIVE, RHR-MOV-MO13D-PASSIVE, RHR-MOV-MO16B, RHR-MOV-MO20-PASSIVE, RHR-MOV-MO26B, RHR-MOV-27B, RHR-MOV-MO34B, RHR-MOV-MO38B, RHR-MOV-39B, RHR-MOV-MO57-PASSIVE, RHR-MOV-MO65B, RHR-MOV-MO66B, EE-CB-4160G-RHRP1D, and SW-MOV-MO89B). SPC Train B is the credited train for Decay Heat Removal. The complete loss of RHR flow, or the diversion of flow, would challenge this NSPC.

Cable damage to the Control Room indication RHR-FI-133B requires monitoring flow at the unaffected ASD Room indication.

Cable damage to RHR-MOV-MO12B may cause the valve to spuriously close securing discharge from the RHR Heat Exchanger. Control is shifted to the ASD Panel.

Based on cable damage, RHR-MOV-MO13D may spuriously close, therefore, not allowing for a suction path from the Suppression Pool to establish SPC flow. Control is shifted to the ASD Panel.

RHR-MOV-MO16B is normally open to ensure a discharge flow path to prevent damage to the RHR Pumps. Based on cable damage, the valve will not be able to close from the Control Room, therefore, resulting in a 4-inch flow diversion from SPC mode of operation.

Based on cable failure, RHR-MOV-MO20 may spuriously open, resulting in a 20-inch flow diversion through the bypass line, resulting in a loss or reduction of flow during SPC Train B mode of operation.

Control cable damage may spuriously operate RHR-MOV-MO26B. The feeder cable is undamaged. RHR-MOV-MO26B represents a 10-inch diversion path to establishing SPC Train B.

Both RHR-MOV-MO27B and RHR-MOV-MO25B have cable damage which would allow 24-inch flow diversion from SPC flow.

Based on cable failures to RHR-MOV-34B, RHR-MOV-MO38B, and RHR-MOV-MO39B, a return path to the Suppression Pool for SPC Train B may not be available.

RHR-MOV-MO57 damage to cable MR33 could result in spurious opening of the valve without a means to

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

remotely close the valve from the Control Room, resulting in a 4-inch flow diversion.

RHR-MOV-MO65B cable damage may result in spurious closure isolating flow to the RHR Heat Exchanger in SPC Train B mode.

RHR-MOV-MO66B, while normally open, is desired to be closed for SPC Train B operation. Cable damage will not allow for closure of the valve from the Control Room, resulting in the bypass of the RHR Heat Exchanger.

Due to multiple damaged cables, the RHR Pump breaker may fail to close automatically or remotely. This would result in a loss of Decay Heat Removal for SPC Train B.

Based on cable damage to the RHRSW Booster Pump breakers, they cannot be closed. This removes electrical control of RHR-MOV-MO89B and secures SW flow to the RHR Heat Exchanger.

This is a separation issue for Decay Heat Removal.

Disposition A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

RBFN-09

Description

Preventing loss of containment over-pressure in support of RHR pump operation for SPC operation (RW-AOV-AO82 and RW-AOV-AO94, REC-MOV-712MV, REC-MOV-713MV, PC-AOV-245AV, and PC-AOV-246AV).

The NSPC require a means of Decay Heat Removal post-fire. SPC Train B is credited for Decay Heat Removal. The loss of containment over-pressure could result in the loss of NPSH for the credited RHR pump during SPC mode of operation.

Cable damage to the SOVs associated with RW-AOV-AO82, RW-AOV-AO83, RW-AOV-AO93, and RW-AOV-AO94 may result in spurious opening. These valves isolate the Drywell Equipment and Floor Drain Sump discharge paths, and their opening would result in a loss of containment over-pressure.

REC-MOV-712MV and REC-MOV-713MV isolate the REC Critical and Non-Critical Headers. The valves are normally open, and are required to close to secure flow to the Drywell coolers. Based on cable damage and Control Room abandonment, remote operation of these valves is not possible.

Cable damage to M355 (PC-AOV-245AV) and M358 (PC-AOV-246AV) may cause the valves to fail open. These valves are normally closed and desired closed. Spurious opening of these valves results in Drywell and Suppression Chamber vent and purge lines opening, resulting in loss of containment over-pressure.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

RBFN-10

Description

Prevent loss of RPV inventory and high pressure in the low pressure portion of the RWCU system from the Control Room due to cable fire damage (RWCU-MOV-MO15).

The NSPC require a means of isolating the RPV and low pressure systems from the RPV post-fire. Cable damage (MR122) may cause spurious opening and the inability to close RWCU-MOV-MO15, resulting in loss of RPV Inventory to the RWCU system, or cause high pressure in the low pressure portion of RWCU piping downstream.

This is a separation issue for Inventory and Pressure Control.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

RBFN-11

Description

Preventing an RPV overfill condition with spurious RCIC startup and loss of control due to cable damage (RCIC-MOV-MO15).

The NSPC require a means of maintaining water level in the RPV post-fire. HPCI maintains level in this area, but with the cables damaged due to fire in this area for RCIC-MOV-MO15, RCIC-MOV-MO16, and RCIC-MOV-131MV, system status is not assured, and may result in a startup of the RCIC system without control and cause an RPV overfill condition.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-FN	Reactor Building 903' Northeast Corner

RBFN-12

Description

Loss of RPV water level and pressure along with Suppression Chamber level and temperature indications in the Control Room (NBI-LT-59B, NBI-LT-91B, NBI-PT-53B, PC-LT-10, and PC-TR-25).

The NSPC require a means of monitoring plant conditions post-fire. While instrument cables to the Control Room are impacted by the fire, these indications are available at the ASD Panel.

This is a separation issue for Process Monitoring.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

RBFN-13

Description

Loss of power to critical equipment for plant operation (EE-MCC-TX and SW Pump D).

The NSCA require a means of powering critical pieces of equipment. EE-MCC-TX feeder cables SG-27A and 27B, along with SW Pump D feeder cable H521. Failure of these cables will result in loss of credited process monitoring instrumentation and the SW function.

This is a separation issue for Process Monitoring and Vital Auxiliaries.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area | **Description**

RB-FN | Reactor Building 903' Northeast Corner

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
2A-1	Detection	Ionization	R	N	N	N	Y	Y	N
2A-1	Detection	Heat	R	N	N	N	Y	Y	N
2A-1	Suppression	Automatic Wet-Pipe	P	N/A	N	N	Y	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. The CRD solenoid-operated valves and the Scram Discharge Volume level instruments will be impacted by actuation of the suppression system. Motor Control Center EE-MCC-K is provided with horizontal spray shields to prevent direct spray on the MCC and the conduit ends are sealed to prevent water from entering the MCCs. The conduit ends entering starter racks EE-STR-125 RCIC, EE-STR (RHR-MOV-MO67), EE-STR-125 RCIC (RCIC-MOV-MO131), and disconnect switches EE-DSC-125 RCIC and EE-SW-125 RCIC and Distribution Panels EE-PNL-AA3, EE-PNL-BB3, EE-PNL-CA, and EE-PNL-CB are provided with seals to prevent water from entering the equipment. In the event of normal operation, the wet-pipe sprinkler system will not adversely effect the other equipment operating within the Fire Zone. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-J	Critical Switchgear Room 1F

Fire Zone	Description
3A	Critical Switchgear Room 1F

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train B will be operated to maintain Suppression Pool temperature.	RBJ-03
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - Emergency Condensate Storage Tank level [from Control Room] - RHR and SW flow indications [from Control Room] - HPCI flow, pressures, and turbine speed indication [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection is provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. The HPCI system will be used to control RPV pressure and to maintain RPV level.	RBJ-01
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-J	Critical Switchgear Room 1F	
Vital Auxiliaries	Mechanical: - REC will be supplied by Train B to provide the cooling supply to the ECCS. - SW Train B will be operated to provide the cooling supply to the REC system, RHR Heat Exchangers and DG 2. Electrical (AC/DC): - DG 2 aligned to 4160G Bus - 125/250 VDC Train B is available HVAC: - DG 2 HVAC system - AC Switchgear Room 1G - Essential Control Building HVAC system - DC Switchgear Room 1B - Essential Control Building HVAC system - Battery Room 1B - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1B - Essential Control Building HVAC system	RBJ-02 RBJ-03

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-J	Critical Switchgear Room 1F

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-031 - Evaluation of Critical Switchgear Rooms 1F and 1G Fire Barrier Separation

Purpose The purpose of this evaluation is to document the acceptability of the fire barrier separation that has been provided for Critical Switchgear Rooms 1F and 1G. A fire rating cannot be assigned to the barriers due to miscellaneous door discrepancies, ventilation ducts that do not contain fire dampers at the barriers, and unsealed bus duct penetrations.

Conclusion Based on the expected fire hazards and the existing fire protection features associated with the Critical Switchgear Rooms, the installation of additional fire seals within the bus duct enclosure, fire dampers in the ventilation ductwork, or penetration seal material in the conduits, will not significantly improve fire protection required to ensure redundant safe shutdown capability. The existing fire barrier configurations are adequate for the fire hazards in the areas, and to prevent propagation of fire between adjacent areas.

Basis The bases that justify this conclusion are summarized as follows:

- The wall separating the two Switchgear Room Fire Zones is approximately 12 inches of reinforced concrete.
- On the Switchgear Room side of the wall, the ducts are coated with a 1.5-hour rated fire retardant material.
- The outlet of the supply duct and the inlet of the exhaust duct in the Critical Switchgear Room are provided with 1.5-hour rated fire dampers.
- The exhaust ductwork in the Critical Switchgear Room has a smoke detection system interlock that is arranged to shutdown the operating supply fan.
- Plant procedures effectively reduce the possibility of a fire involving transient ignition sources and combustibles.
- Equivalent fire severity in both Switchgear Rooms is less than 1-minute and consists of an assumed transient combustible load and small quantities of miscellaneous plastic.
- The Corridor side of wall near the ventilation openings is essentially void of any significant combustible loading with the exception of miscellaneous combustibles stored in closed metal lockers.
- A smoke detection system has been provided in both Critical Switchgear Rooms which will result in an alarm to the constantly attended Control Room upon activation.
- Manual fire suppression means has been provided for this area in the form of fire hoses and extinguishers.

Table B-3 Fire Area Transition

Fire Area	Description
RB-J	Critical Switchgear Room 1F

Variances from Deterministic Requirements (VFDR)

RBJ-01

Description

Preventing a RR Pump Seal LOCA due to inability to secure the RR Pump from the Control Room with the REC Non-Critical Header secured (RR Pump Breaker for 4160C-1CS).

Breaker F/FDR to the 4160V Bus from the Startup Transformer: This is a normally available, required open breaker that provides motive power to RR Pump A. The RR Pump is required to trip to prevent potential RR Pump Seal LOCA when REC is not available to provide cooling. This would challenge the NSPC for Inventory Control. REC is secured in this fire area to address potential containment over-pressure. Remote operation of the breaker from the Control Room is lost due to cables H251 and H254 (1C bus) fire-induced damage.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-J	Critical Switchgear Room 1F

RBJ-02

Description

Loss of Critical Switchgear cooling due to cable damage to damper AD-1408 (HV-FAN-SF-SWGR-1G and HV-FAN-EF-SWGR-1G).

Fire damage to cables will not preclude operation of the EF-SWGR-1G and SF-SWGR-1G fans from the 1G AC Switchgear Room. Cable damage will affect the operation of ventilation damper AD-1408 by energizing its solenoid and keeping the damper open. The NSCA model requires that either train of Switchgear Room cooling fans be available to ensure Switchgear remains available post-fire.

Based on damage to damper AD-1408, fans HV-FAN-SF-SWGR-1G and HV-FAN-EF-SWGR-1G will not provide adequate ventilation.

This is a separation issue for Vital Auxiliaries, Electrical Power Distribution.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

RBJ-03

Description

Preventing a loss of SW for supporting Decay Heat Removal (SW-MOV-MO89B).

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for the NSPC. SPC Train B is credited for Decay Heat Removal; REC Train B provides cooling for RHR Pump 1D and the HPCI Room. The complete loss of SW to the RHR Heat Exchanger could challenge this NSPC.

Cable damage to both RHRSW Booster Pump breakers would result in the inability to close either breaker, causing SW-MOV-MO89B to fail closed without the ability to open from the Control Room. This would result in the loss of SW flow to the RHR Heat Exchanger for SPC Train B.

This is a separation issue for Vital Auxiliaries and Decay Heat Removal.

Table B-3 Fire Area Transition

Fire Area	Description
RB-J	Critical Switchgear Room 1F

Disposition A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
3A	Detection	Ionization	R	N	N	N	Y	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. There are no fixed suppression systems in the area. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-K	Critical Switchgear Room 1G

Fire Zone	Description
3B	Critical Switchgear Room 1G

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A will be operated to maintain Suppression Pool temperature.	RBK-03 RBK-05
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indications [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train A to maintain RPV level.	RBK-01 RBK-03
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-K	Critical Switchgear Room 1G	
Vital Auxiliaries	Mechanical: - REC will be supplied by SW Train A to provide the cooling supply to the ECCS. - SW Train A will be operated to provide the cooling supply to the REC system, RHR Heat Exchangers, and DG 1. Electrical (AC/DC): - DG 1 aligned to 4160F Bus - 125/250 VDC Train A is available HVAC: - DG/CS Train A - Quad area cooling - DG 1 HVAC system - AC Switchgear Room 1F - Essential Control Building HVAC system - DC Switchgear Room 1A - Essential Control Building HVAC system - Battery Room 1A - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1A - Essential Control Building HVAC system	RBK-02 RBK-03 RBK-04

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-K	Critical Switchgear Room 1G

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-031 - Evaluation of Critical Switchgear Rooms 1F and 1G Fire Barrier Separation

Purpose The purpose of this evaluation is to document the acceptability of the fire barrier separation that has been provided for Critical Switchgear Rooms 1F and 1G. A fire rating cannot be assigned to the barriers due to miscellaneous door discrepancies, ventilation ducts that do not contain fire dampers at the barriers, and unsealed bus duct penetrations.

Conclusion Based on the expected fire hazards and the existing fire protection features associated with the Critical Switchgear Rooms, the installation of additional fire seals within the bus duct enclosure, fire dampers in the ventilation ductwork, or penetration seal material in the conduits, will not significantly improve fire protection required to ensure redundant safe shutdown capability. The existing fire barrier configurations are adequate for the fire hazards in the areas, and to prevent propagation of fire between adjacent areas.

Basis The bases that justify this conclusion are summarized as follows:

- The wall separating the two Switchgear Room Fire Zones is approximately 12 inches of reinforced concrete.
- On the Switchgear Room side of the wall, the ducts are coated with a 1.5-hour rated fire retardant material.
- The outlet of the supply duct and the inlet of the exhaust duct in the Critical Switchgear Room are provided with 1.5-hour rated fire dampers.
- The exhaust ductwork in the Critical Switchgear Room has a smoke detection system interlock that is arranged to shutdown the operating supply fan.
- Plant procedures effectively reduce the possibility of a fire involving transient ignition sources and combustibles.
- Equivalent fire severity in both Switchgear Rooms is less than 1-minute and consists of an assumed transient combustible load and small quantities of miscellaneous plastic.
- The Corridor side of wall near the ventilation openings is essentially void of any significant combustible loading with the exception of miscellaneous combustibles stored in closed metal lockers.
- A smoke detection system has been provided in both Critical Switchgear Rooms which will result in an alarm to the constantly attended Control Room upon activation.
- Manual fire suppression means has been provided for this area in the form of fire hoses and extinguishers.

Table B-3 Fire Area Transition

Fire Area	Description
RB-K	Critical Switchgear Room 1G

Variances from Deterministic Requirements (VFDR)

RBK-01

Description

Preventing a RR Pump Seal LOCA due to inability to secure the RR Pumps from the Control Room with the REC Non-Critical header secured (RR Pump Breakers for 4160C-1CS and 4160D-1DS).

Breakers F/FDR to the 4160V Buses from the Startup Transformer: These are normally available, required open breakers that provide motive power to the RR Pumps. The RR Pumps are required to trip to prevent potential RR Pump Seal LOCA when REC is not available to provide cooling. This would challenge the NSPC for Inventory Control. REC is secured in this fire area to address the potential containment over-pressure. Remote operation of the breaker(s) from the Control Room is lost due to cables H251 and H254 (1C bus) and H291 and H294 (1D bus) fire-induced damage.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-K	Critical Switchgear Room 1G

RBK-02

Description

Establish vital auxiliaries by powering the credited 4160F Bus from DG1 (EE-CB-4160F-1FA and EE-CB-4160F-1FS).

The NSCA require at least one source of AC power be available. Offsite power from the Emergency Transformer is unavailable due to fire damage. Cable damage (H432 and H433) could result in loss of breaker control, or spurious actuation of normally closed EE-CB-4160F-1FA. Breaker 1FA may not open as part of the transfer process due to loss of the Startup Transformer. Cable damage to H443 in the 1FS breaker may cause it to spuriously close, placing the 4160F Bus on the damaged bus duct from the Emergency Transformer.

The FA and FS breaker cables discussed above are different than the cables that relay actual breaker position to the DG breaker to close. Therefore, the potential for out-of-phase paralleling is not a concern based on the following: The DG will start normally as part of the electrical power transfer process. With cable damage to the 1FA and 1FS breakers, either the 1FA breaker may stay closed or the 1FS breaker may close on its own. The 1FS breaker closing is a result of the normal transfer process, or cable damage, even though the Emergency Transformer cabling is damaged by fire. Either of the above situations will result in the DG output breaker not closing on its own. While this ensures the DG and offsite power will not be paralleled out-of-phase, there may be no source of AC power to the credited 4160F Bus.

This is a separation issue for Vital Auxiliaries, Electrical Power Distribution.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Modification

Modification: Item S-2.1 of LAR Attachment S, Table S-2.

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-K	Critical Switchgear Room 1G

RBK-03

Description

4160F UV Circuit cable damage affects automatic/remote operation from the Control Room of Critical Pumps for Safe Shutdown (EE-CB-4160F-CSP1A, EE-CB-4160F-RHRP1A, and EE-CB-4160F-RSWP1A).

Fire in the area could result in damage to the 4KV Bus 1F UV circuit, which will cause a trip signal to the CS Pump 1A, RHR Pump 1A, RHRSW Booster Pump 1A and SW Pump 1A breakers. The SW Pump 1A breaker is covered separately, VFDR RBK-04, due to the time requirements to ensure the DG remains available for powering the 4160F Bus.

The NSCA requires the ability to maintain RPV level. CS Train A is credited for Inventory Control in this area. RHR Train A is credited to support SPC mode of RHR following a fire in this area. The NSPC require a means of Decay Heat Removal post-fire be available. SPC Train A is credited for Decay Heat Removal. Inability to keep the RHR Pump 1A breaker closed would result in loss of RHR flow to the RHR Heat Exchanger resulting in loss of SPC Train A. Inability to close the RHRSW Booster Pump 1A breaker results in SW-MOV-MO89A closing and staying closed, securing SW flow to the RHR Heat Exchanger A (see VFDR RBK-05). SW provides cooling to REC, DGs, and RHR systems.

REC is required to provide both Quad cooling in support of Pump operation and cooling for the RHR Pump and Pump Room. Additional Quad heat up due to fire in this area is not credible.

This is a separation issue for Vital Auxiliaries, Inventory and Pressure Control, and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Modification and Recovery Action carried forward as Defense-in-Depth for EE-CB-4160F-RSWP1A.

Modification: Item S-2.1 of LAR Attachment S, Table S-2.

Modification: Item S-2.1 of LAR Attachment S, Table S-2.

Table B-3 Fire Area Transition

Fire Area	Description
RB-K	Critical Switchgear Room 1G

RBK-04

Description

DG1 is the credited source of power. Damage to SW Pump 1A (4160F UV circuit) results in the requirement for tripping the DG.

Cable damage to the 4KV Bus 1F UV circuit will cause a trip signal to the SW Pump 1A breaker. Loss of the SW Pump would result in a loss of SW cooling to the RHR Heat Exchangers, DGs, and REC system cooling to critical loads.

This is a separation issue for Vital Auxiliaries, Inventory and Pressure Control, and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Modification

Modification: Item S-2.1 of LAR Attachment S, Table S-2.

RBK-05

Description

Preventing a loss of SW for supporting Decay Heat Removal (SW-MOV-MO89A).

SW Train A is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for the NSPC. SPC Train A is credited for Decay Heat Removal, REC Train A provides cooling for RHR Pump 1A and the CS Quad. The complete loss of SW to the RHR Heat Exchanger could challenge the NSPC. Cable damage to both RHRSW Booster Pump breakers would result in the inability to close either breaker, causing SW-MOV-MO89A to fail closed without the ability to open from the Control Room. This would result in the loss of SW flow to the RHR Heat Exchanger for SPC Train A.

Note: EE-CB-4160F-SS1F is the feeder breaker, and while it has an overcurrent trip it is not part of the UV circuit, and therefore, this breaker will not open when power is restored to the critical MCCs on the 480F Bus (MCC-Q is one of these that will remain closed – there are several 480 Bus breakers that will trip open on the loss-of-power to the 480V bus). Therefore, once power is restored to the bus (to get the pumps running), SW-MOV-MO89A will be operable from its starter.

This is a separation issue for Decay Heat Removal.

Table B-3 Fire Area Transition

Fire Area	Description
RB-K	Critical Switchgear Room 1G

Disposition A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
3B	Detection	Ionization	R	N	N	N	Y	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. There are no fixed suppression systems in the area. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

Fire Zone	Description
2B	RHR Heat Exchanger Room A
3C	REC Heat Exchanger and Pump Area
3D	Reactor Recirculation Motor Generator Set Lube Oil Cooler Area
3E-2	RWCU Pump Area and Corridor

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train B will be operated to maintain Suppression Pool temperature.	RBM-05 RBM-07
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level [from Control Room] - RPV pressure [from NBI-PI-61 at Rack 25-51] - Suppression Pool level and temperature [from Control Room] - RHR and SW flow indications [from Control Room] - Emergency Condensate Storage Tank level [from Control Room] - RCIC system flow, pressures, and turbine speed indication [from Control Room]	RBM-04
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection is provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. The RCIC system will be used to control RPV pressure and to maintain RPV level.	RBM-01 RBM-03 RBM-06 RBM-08
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A	
Vital Auxiliaries	Mechanical: - REC will be supplied by SW Train B to provide the cooling supply to the ECCS. - SW Train B will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160G Bus - 125/250 VDC Train B is available HVAC: - DG/CS Train A - Quad area cooling - AC Switchgear Room 1G - Essential Control Building HVAC system - DC Switchgear Room 1B - Essential Control Building HVAC system - Battery Rooms 1B - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1B - Essential Control Building HVAC system	RBM-02 RBM-06 RBM-07

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Existing Engineering Equivalency Evaluations (EEEE)

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

Variances from Deterministic Requirements (VFDR)

RBM-01

Description

Preventing a RR Pump Seal LOCA due to inability to secure the RR Pumps from the Control Room with the REC Non-Critical Header secured (RR Pump Breakers for 4160C-1CS and 4160D-1DS).

Breakers F/FDR to the 4160V Buses from the Startup Transformer: These are normally available, required open breakers that provide motive power to the RR Pumps. The RR Pumps are required to trip to prevent a potential RR Pump Seal LOCA when REC is not available to provide cooling. This would challenge the NSPC for Inventory Control.

REC is secured in this fire area to address potential containment over-pressure. Remote operation of the breaker(s) from the Control Room is lost due to cables H251 and H254 (1C bus), and H291 and H294 (1D bus) fire-induced damage.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

RBM-02

Description

Establish vital auxiliaries by powering the credited 4160F and G Bus from Offsite Power (EE-CB-4160F-SS1F, EE-CB-4160F-1FS, EE-CB-4160G-1GS, EE-CB-4160FRHRP1B, and EE-CB-4160G-SS1G)

The NSCA require at least one source of AC power be available. Both 4160F and G UV circuits have cable damage and DC control power cables are damaged.

EE-CB-4160F-SS1F breaker cable damage (H422 and H423) may cause it to spuriously trip, causing a loss-of-power to the 480F Bus.

Cable damage (DC control power and UV Circuit) could result in the breaker (EE-CB-4160F-1FS) not closing automatically during the normal transfer process due to loss of the Startup Transformer.

Cable damage (H551, H552, H553, H555 and UV Circuit) could result in the breaker (EE-CB-4160G-1GS) not closing automatically during the normal transfer process due to loss of the Startup Transformer.

Fire in the area could result in damage to the 4KV Bus 1F UV circuit which will cause a trip signal to EE-CB-4160F-RHRP1B. The NSCA require the ability to remove Decay Heat post-fire. RHR Train B is credited to support the SPC mode of RHR following a fire in this area. Inability to keep RHR Pump 1B breaker closed would result in the loss of RHR flow to the RHR Heat Exchanger, resulting in loss of SPC Train B.

Cable damage (H542, H543, DC control power, and UV circuit) could result in the breaker (EECB-4140G-SS1G) tripping, securing power to the 480G Bus.

This is a separation issue for Vital Auxiliaries.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

RBM-03

Description

Ability to close SRV's automatically/remotely from the Control Room is lost due to cable fire damage (Pilot Valves SPV71D and SPV71F).

The NSPC require the ability to isolate the RPV to maintain water inventory above the active fuel. Spurious operations could result in the opening of up to two ADS valves. Affected ADS valves need to be returned to their fail-safe closed position within 18 minutes for single spurious operation.

SRV's D and F are impacted as a result of fire in this area. Cables RP517 and RP518 are impacted for SPV71D, and RP519 and RP520 for SPV71F. In Fire Area RB-M, RP517 and 518 are impacted in the following scenarios: 3C-TS03, 3D-TS02, 3C zone failure and 3D zone failure. In Fire Area RB-M, RP519 and 520 are impacted in the following scenarios: 3C-TS02, 3C-TS12, and 3C zone failure. Therefore, two SRVs would only be impacted at the same time in RB-M for a full zone burnout of Fire Zone 3C.

This is a separation issue for isolating the RPV for Inventory and Pressure Control.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

RBM-04

Description

Preventing the loss of RPV level and pressure indication due to the fire affects on sensing line piping and pressure transmitter signal cable damage (NBI-LI-91A and RFC-PI-90A).

The NSCA require a means of monitoring RPV water level. NBI-LT-91A, NBI-LT-91B, and NBI-LT- 91C are affected due to sensing lines located in the Fire Area. The NBI-LT-91A and NBI-LT-91C sensing line is located in the west side of the Reactor Building. The NBI-LT-91B sensing line is located in the east side of the Reactor Building. A separation of approximately 80 feet exists between the redundant sensing lines.

The NSCA require a means of RPV pressure indication, and all three channels are affected in this area. RFC-PI-90A provides the Control Room with RPV pressure indication.

Note: Fire scenarios in Fire Area RB-M consist of REC Pump motor fires, a ventilated electrical panel fire, non-ventilated electrical panel and MCC fires, and transient fires. The REC Pump motors are not oil lubricated, and there are no secondary combustibles that would be involved in the pump fire or the ventilated electrical panel fire. Therefore, the zone of influence of the fire scenarios in Fire Area RB-M is not large enough to result in damage to targets located 80 feet apart.

This is a separation issue for Process Monitoring.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

RBM-05

Description

Preventing loss of containment over-pressure in support of RHR pump operation for SPC operation (PC-MOV-231MV).

The NSPC require a means of Decay Heat Removal post-fire be available. SPC Train B is credited for Decay Heat Removal. The loss of containment over-pressure could result in the loss of NPSH for the credited RHR pump during the SPC mode of operation.

Both PC-AOV-246AV and PC-MOV-231MV cables could be affected by the fire and may spuriously open. These valves isolate the Vent and Purge path from the Drywell, and would result in a loss of containment over-pressure.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

RBM-06

Description

Preventing a full or partial loss of SW for supporting Decay Heat Removal and Inventory and Pressure Control from the Control Room with SW supplying REC (REC-MOV-694MV, REC-MOV- 695MV, REC-MOV-697MV, REC-MOV-698MV, REC-MOV-711MV-Passive, REC-MOV-714MV-Passive, SW-MOV-651MV, SW-MOV-886MV-Passive, SW-MOV-887MV, and EE-CB-4160G-SWP1B).

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. RCIC is credited for Inventory and Pressure Control, SPC Train B is credited for Decay Heat Removal, REC Train B provides cooling for RHR Pump 1D and the RHR and CS Pump Rooms (Quads) and SW Train B provides the cooling for REC and RHR Heat Exchangers (DGs not credited). The complete loss or diversion of SW to REC could challenge these NSPCs.

Current requirements indicate for RCIC that REC flow is required within 4 hours of the start of RCIC Turbine. Based on fire location, heat up of the Quad is not expected.

NOTE: Cross-ties between REC Train A and REC Train B are opened to support RCIC Room cooling with the credited REC Train B.

Cable damage to 4160G UV Circuit may cause SW Pump 1B breaker to spuriously trip. The breaker tripping will cause a complete loss of SW flow.

This is a separation issue for Inventory and Pressure Control and Vital Auxiliaries.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

RBM-07

Description

Preventing a complete or partial loss of Suppression Pool Cooling due to flow diversion or loss of SW cooling (RHR-MOV-MO20-Passive, RHR-MOV-MO27B, and SW-MOV-MO89B).

SW Train B is required to support various functions (SPC, HPCI and CS Quad cooling, REC, DGs, etc) for NSPC. SPC Train B is credited for Decay Heat Removal, REC Train B provides cooling for RHR Pump 1B, and REC Train A cools the RCIC Quad. The flow diversion of RHR and/or the complete loss of SW to the RHR Heat Exchanger could challenge this NSPC.

RHR-MOV-MO20 is normally closed, and desired closed during SPC mode of operation. Based on cable failure (MR22), the valve may spuriously open, resulting in a 20-inch flow diversion through the bypass line. This diversion would result in a loss or reduction of flow during SPC Train B mode of operation.

RHR-MOV-MO27B is normally open, with RHR-MOV-MO25B normally closed during the SPC mode of operation. Based on cable failure, RHR-MOV-MO25B may spuriously open, resulting in a 20-inch flow diversion without the ability to close either RHR-MOV-MO25B or RHR-MOV-MO27B from the Control Room. This diversion would result in a loss or reduction of flow during SPC Train B mode of operation.

Cable damage to both RHRSW Booster Pump breakers results in the inability to close either breaker, causing SW-MOV-MO89B to fail closed without the ability to open from the Control Room. This would result in the loss of SW flow to the RHR Heat Exchanger for SPC Train B.

This is a separation issue for Vital Auxiliaries and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

RBM-08

Description

Loss of RCIC operation due to cable damage providing erroneous RPV low water level and RPV high water level signals, resulting in RCIC not starting (RCIC ECCS, RCIC-MOV-MO131, and RCIC-MOV-MO18).

Cable damage could cause a spurious RPV high and low water level signal to the RCIC ECCS logic. These erroneous signals will not allow RCIC to start up automatically as required. Isolating the damaged cables will allow for RCIC operation.

This is a separation issue for Inventory and Pressure Control.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
2B	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
3C	Detection	Ionization	R	N	N	N	N	Y	N
3C	Feature	Radiant Energy Shield	N/A	N/A	N	N	N	Y	N
3D	Detection	Ionization	R	N	N	N	N	Y	N
3D	Suppression	Automatic Wet-Pipe	P	N/A	N	N	N	N	N
3E-2	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	Description
RB-M	Reactor Building North / East Side, RHR Heat Exchanger Room A

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. In the event of normal operation of the wet-pipe sprinkler system in Fire Zone 3D, there is a limited potential for effects from the system, as water discharge would be contained within the diked area around the RR MG Set Heat Exchangers, which has sealed penetrations. A limited amount of overspray will occur to adjacent areas next to the heat exchangers. This area has no direct flow path to the 903'-6" Elevation below, and there are no identified direct paths from Fire Zone 4D above. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

In Fire Zone 3C, radiant shielding is being installed for the conduit bank located along the west wall of the Critical Switchgear Rooms to prevent damage from transient fires. See Attachment S, Table S-2, Item S-2.5.

Cable tray risers west of the elevator, in Fire Zone 3C, are being provided with radiant shielding to prevent damage from transient fires. See Attachment S, Table S-2, Item S-2.6.

Table B-3 Fire Area Transition

Fire Area	Description
RB-N	Reactor Building South West Corner, RHR Heat Exchanger Room B and RWCU Heat Exchanger Room

Fire Zone	Description
2D	RHR Heat Exchanger Room B
3E-1	RWCU Regenerative Heat Exchanger Areas
3E-2	RWCU Pump Area and Corridor

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A will be operated to maintain Suppression Pool temperature.	RBN-01 RBN-02
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indications [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train A to maintain RPV level.	RBN-01
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-N	Reactor Building South West Corner, RHR Heat Exchanger Room B and RWCU Heat Exchanger Room	
Vital Auxiliaries	Mechanical: - REC will be supplied by Train A to provide the cooling supply to the ECCS. - SW Train A will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160F Bus - 125/250 VDC Train A is available HVAC: - DG/CS Train A - Quad area cooling - DG 1 HVAC system - AC Switchgear Room 1F - Essential Control Building HVAC system - DC Switchgear Room 1A - Essential Control Building HVAC system - Battery Room 1A - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Room 1A - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Existing Engineering Equivalency Evaluations (EEEE)

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-N	Reactor Building South West Corner, RHR Heat Exchanger Room B and RWCU Heat Exchanger Room

Variances from Deterministic Requirements (VFDR)

RBN-01

Description

Preventing loss of containment over-pressure in support of RHR pump operation for SPC operation while not allowing flow diversion from credited REC Train A for cooling to CS Quad and RHR pump (REC-MOV-712MV and REC-MOV-714MV-Passive).

The NSPC require a means of RPV Inventory and Pressure Control and Decay Heat Removal post-fire. CS Train A is credited for Inventory and Pressure Control and SPC Train A is credited for Decay Heat Removal. The flow diversion through REC-MOV-712MV and REC-MOV-714MV would result in a reduction of cooling flow to the credited RHR Pump and the CS Pump Quads. The loss of containment over-pressure could result in the loss of NPSH for the credited RHR pump during the SPC mode of operation.

REC-MOV-712MV isolates the REC Non-Critical Header from the outlet of the REC Train A Heat Exchanger. REC is required to provide Quad cooling in support of Core Spray pump operation for NSCA. CS Train A is the credited success path following a fire in this area. Current requirements indicate 1 hour is needed to establish Quad cooling. REC-MOV-712VN represents a 12-inch flow diversion from the REC Critical Header to ensure adequate Quad cooling.

REC-MOV-714MV isolates the REC Train B Heat Exchanger. REC is required to provide Quad cooling in support of CS Pump operation for NSCA. CS Train A is the credited success path following a fire in this area. Current requirements indicate 1 hour is needed to establish Quad cooling. REC-MOV-714MV represents a 6-inch flow diversion from the REC Train A Critical Header to ensure adequate Quad cooling.

This is a separation issue for Inventory and Pressure Control and Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-N	Reactor Building South West Corner, RHR Heat Exchanger Room B and RWCU Heat Exchanger Room

RBN-02

Description

Preventing a complete or partial loss of Suppression Pool Cooling due to flow diversion (RHR-MOV-MO27A).

SPC Train A is credited for Decay Heat Removal post-fire in support of the NSPC. RHR-MOV-MO27A is normally open with RHR-MOV-MO25A normally closed during the SPC mode of operation. Based on cable failure, RHR-MOV-MO25A may spuriously open, resulting in a 20-inch flow diversion, without the ability to close either RHR-MOV-MO25A or RHR-MOV-MO27A from the Control Room due to cable failures. This diversion would result in a loss or reduction of flow during SPC Train A mode of operation.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-N	Reactor Building South West Corner, RHR Heat Exchanger Room B and RWCU Heat Exchanger Room

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
2D	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
3E-1	Detection	Ionization	R	N	N	N	N	N	Y
3E-2	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-P	Reactor Building 958 Accessible Areas

Fire Zone	Description
4A	Reactor Building Elevator and Accessway Area
4B	Reactor Building HVAC Area
4C	Fuel Pool Heat Exchanger, CRD Repair Room, and Raw Water Cleanup Areas
4D	Reactor Recirculation Motor Generator Set Oil Pump Area

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A or B will be operated to maintain Suppression Pool temperature.	RBP-01
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indications [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of either Core Spray Train A or Train B to maintain RPV level.	RBP-02
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-P	Reactor Building 958 Accessible Areas	
Vital Auxiliaries	Mechanical: - REC will be supplied by Train A or B to provide the cooling supply to the ECCS. - SW Train A or B will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160F or 4160G Bus - 125/250 VDC Trains A and B are available [from Control Room] HVAC: - RCIC/CS Trains A and B - Quad area cooling - AC Switchgear Rooms - Essential Control Building HVAC system - DC Switchgear Rooms - Essential Control Building HVAC system - Battery Rooms - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Rooms - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-P	Reactor Building 958 Accessible Areas

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title

EE 12-013 - Evaluation of the SLC Pump Tank and Accessway (Fire Zone 5A) and Refueling Floor (Fire Zone 6) Fire Barrier Separation

Purpose

This evaluation is written to justify several unrated and open penetrations in the Fire Area RB-T floor boundaries to the adjacent fire area, including an open equipment hatchway, an open stairwell, and fire doors having a fire resistance rating less than 3 hours. The evaluation documents that the separation between Fire Area RB-T and adjacent fire areas is adequate to demonstrate compliance NFPA 805 section 4.2.3.2.

Conclusion

The existing fire area boundary configurations and automatic and manual fire protection features provide a level of protection that are adequate to prevent damage to safe shutdown systems in adjacent fire areas. The performance-based analysis has assessed the adequacy of the fire barrier forming the fire boundary to withstand the fire effects of the hazards in the area.

Basis

The bases that justify this conclusion are summarized as follows:

- The combustible loading classification for Fire Zone 5A is "LOW" and is mainly attributed to cable insulation, lube oil, miscellaneous plastic, fiberglass, and PVC.
- The SLC equipment is located within a concrete curbed area which will prevent any spill of oil down to Fire Zone 4A via floor openings.
- The combustible loading classification for Fire Zone 4A is "LOW" and is mainly attributed to cable insulation, miscellaneous fiberglass, and plastic.
- Fixed ignition sources in Fire Zone 4A are limited to small electrical panels, instrument racks, and MCCs. The MCCs are well-sealed, robustly secured panels and are not located adjacent to the open Stair S-3.
- The lack of significant combustible materials in either fire zone and the lack of intervening combustibles in the open Stair S-3 significantly reduce the chance of fire propagation between zones.
- The combustible loading classification for Fire Zone 5B is "MEDIUM" and is mainly attributed to cable insulation, lube oil, charcoal, miscellaneous plastic, and fiberglass.
- The combustible loading classification for Fire Zone 6 is "LOW" and is mainly attributed to lube oil.
- All fire zones are provided with automatic detection system coverage. In the event of a fire, detection system actuation will result in rapid fire brigade response and subsequent manual extinguishment utilizing hose stations and portable extinguishers strategically located in adjacent fire zones.
- Transient combustibles are administratively controlled by CNS procedure 0.7.1, "Control of Combustibles," effectively reducing the possibility of a fire involving transient materials.
- Fire Zone 5B is provided with I-beams at the floor to contain the oil to the area around the MG Sets. The location of the I-beams prevents oil entering the area near the fire doors.

Table B-3 Fire Area Transition

Fire Area	Description
RB-P	Reactor Building 958 Accessible Areas
	• The Stud Tensioner in Fire Zone 6 is only used when manned, therefore, a fire involving the oil would be immediately detected. Additionally, the equipment hatchway is provided with steel curbing to prevent oil leakage down to lower elevations. • A carbon dioxide hose reel is provided along the west wall near MG Set A in Fire Zone 5B. A portable foam hose cart is located near the southwest stairwell in Fire Zone 5B. • Fixed automatic preaction sprinkler system coverage is provided for the Reactor MG Sets in Fire Zone 5B.

Table B-3 Fire Area Transition

Fire Area	Description
RB-P	Reactor Building 958 Accessible Areas

Variances from Deterministic Requirements (VFDR)

RBP-01

Description

Preventing the loss of Decay Heat Removal function during Suppression Pool Cooling operations due to loss of NPSH.

Both valves (PC-AOV-246AV and PC-MOV-231MV) in the Drywell vent path could be affected by fire in this area. The spurious opening of PC-MOV-231MV resulting from cable damage to any single cable or combination of MRA13, MRA14, and MRA16 may challenge the NSPC for Decay Heat Removal. When this containment vent/purge line valve opens and stays open, it results in the loss of containment over-pressure, and therefore, the possible loss of NPSH for the RHR pump in supporting SPC Trains A and B. Normally open PC-V-510 and normally closed PC-MOV-306MV are in series and provide for equalizing around PC-MOV-231MV. Neither valve is modeled, based on 2-inch bypass line using a conservative assumption, since both PC-MOV-306MV and PC-MOV-231MV are located physically in close proximity, fire damage to PC-MOV-231MV will cause PC-MOV-306MV to fail as-is, which would leave a 2-inch vent path around PC-MOV-231MV.

This is a separation issue for Decay Heat Removal.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-P	Reactor Building 958 Accessible Areas

RBP-02

Description

Preventing a RR Pump Seal LOCA due to inability to secure the RR Pumps from the Control Room with the REC Non-Critical header secured (RR Pump Breakers for 4160C-1CS and 4160D-1DS).

Breakers F/FDR to the 4160V Buses from the Startup Transformer: These are normally available, required open breakers that provide motive power to the RR Pumps. The RR Pumps are required to trip to prevent a potential seal LOCA when REC is not available to provide cooling. This would challenge the NSPC for Inventory and Pressure Control. REC is secured in this fire area to address potential containment over-pressure. Remote operation of the breaker(s) from the Control Room is lost due to cables H251 (1C bus) and H291 (1D bus) fire-induced damage.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
RB-P	Reactor Building 958 Accessible Areas

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
4A	Detection	Ionization	R	N	N	N	Y	Y	N
4B	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
4C	Detection	Ionization	R	N	N	N	N	Y	N
4D	Detection	Ionization	R	N	N	N	N	Y	N
4D	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. In the event of normal operation of the wet-pipe sprinkler system in Fire Zone 4D, water would be dispersed at approximately the 8 ft level and below throughout the zone. The zone is provided with containment dikes and sealed pipe shafts and an enclosed stairwell. Additionally, the door to the stairwell is equipped with a dike to prevent flow of suppression water and oil into the stairwell. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
RB-T	Reactor Building East Side and Refueling Floor

Fire Zone	Description
5A	SLC Pump Tank and Accessway
6	Refueling Floor

Regulatory Basis

4.2.3.2 - Deterministic Approach

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A or B will be operated to maintain Suppression Pool temperature.	None
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indications [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of either Core Spray Train A or Train B to maintain RPV level.	None
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-T	Reactor Building East Side and Refueling Floor	
Vital Auxiliaries	Mechanical: - REC will be supplied by Train A or B to provide the cooling supply to the ECCS. - SW Train A or B will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160F or 4160G Bus - 125/250 VDC Trains A and B are available [from Control Room] HVAC: - RCIC/CS Trains A and B - Quad area cooling - AC Switchgear Rooms - Essential Control Building HVAC system - DC Switchgear Rooms - Essential Control Building HVAC system - Battery Rooms - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Rooms - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-T	Reactor Building East Side and Refueling Floor

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 12-013 - Evaluation of the SLC Pump Tank and Accessway (Fire Zone 5A) and Refueling Floor (Fire Zone 6) Fire Barrier Separation

Purpose This evaluation is written to justify several unrated and open penetrations in the Fire Area RB-T floor boundaries to the adjacent fire area, including an open equipment hatchway, an open stairwell, and fire doors having a fire resistance rating less than 3 hours. The evaluation documents that the separation between Fire Area RB-T and adjacent fire areas is adequate to demonstrate compliance NFPA 805 section 4.2.3.2.

Conclusion The existing fire area boundary configurations and automatic and manual fire protection features provide a level of protection that are adequate to prevent damage to safe shutdown systems in adjacent fire areas. The performance-based analysis has assessed the adequacy of the fire barrier forming the fire boundary to withstand the fire effects of the hazards in the area.

Basis The bases that justify this conclusion are summarized as follows:

- The combustible loading classification for Fire Zone 5A is "LOW" and is mainly attributed to cable insulation, lube oil, miscellaneous plastic, fiberglass, and PVC.
- The SLC equipment is located within a concrete curbed area which will prevent any spill of oil down to Fire Zone 4A via floor openings.
- The combustible loading classification for Fire Zone 4A is "LOW" and is mainly attributed to cable insulation, miscellaneous fiberglass, and plastic.
- Fixed ignition sources in Fire Zone 4A are limited to small electrical panels, instrument racks, and MCCs. The MCCs are well-sealed, robustly secured panels and are not located adjacent to the open Stair S-3.
- The lack of significant combustible materials in either fire zone and the lack of intervening combustibles in the open Stair S-3 significantly reduce the chance of fire propagation between zones.
- The combustible loading classification for Fire Zone 5B is "MEDIUM" and is mainly attributed to cable insulation, lube oil, charcoal, miscellaneous plastic, and fiberglass.
- The combustible loading classification for Fire Zone 6 is "LOW" and is mainly attributed to lube oil.
- All fire zones are provided with automatic detection system coverage. In the event of a fire, detection system actuation will result in rapid fire brigade response and subsequent manual extinguishment utilizing hose stations and portable extinguishers strategically located in adjacent fire zones.
- Transient combustibles are administratively controlled by CNS procedure 0.7.1, "Control of Combustibles," effectively reducing the possibility of a fire involving transient materials.
- Fire Zone 5B is provided with I-beams at the floor to contain the oil to the area around the MG Sets. The location of the I-beams prevents oil entering the area near the fire doors.

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-T	Reactor Building East Side and Refueling Floor

- The Stud Tensioner in Fire Zone 6 is only used when manned, therefore, a fire involving the oil would be immediately detected. Additionally, the equipment hatchway is provided with steel curbing to prevent oil leakage down to lower elevations.
- A carbon dioxide hose reel is provided along the west wall near MG Set A in Fire Zone 5B. A portable foam hose cart is located near the southwest stairwell in Fire Zone 5B.
- Fixed automatic preaction sprinkler system coverage is provided for the Reactor MG Sets in Fire Zone 5B.

Variances from Deterministic Requirements (VFDR)

None

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
5A	Detection	Ionization	R	N	N	N	Y	N	N
6	Detection	Heat	R	N	N	N	Y	N	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-T	Reactor Building East Side and Refueling Floor

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. The standpipe system riser in this area, if inadvertently ruptured, could result in water down the pipe chase in which it is located. A minimal amount of water would spill in this area and the standpipe system is equipped with a water flow alarm. The containment H2/O2 analyzer panels (PC AN H2/O2) are more than 50 feet from the standpipe. Therefore, pipe rupture will not affect these panels. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-V	Reactor Recirculation Motor Generator Set Area

Fire Zone	**Description**
5B	Reactor Recirculation Motor Generator Set Area

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	**Method of Accomplishment**	**Comments / VFDR**
Decay Heat Removal	SPC Train A or B will be operated to maintain Suppression Pool temperature	None
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indications [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of either Core Spray Train A or Train B to maintain RPV level.	RBV-01
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
RB-V	Reactor Recirculation Motor Generator Set Area	
Vital Auxiliaries	Mechanical: - REC will be supplied by Train A or B to provide the cooling supply to the ECCS. - SW Train A or B will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160F or 4160G Bus - 125/250 VDC Trains A and B are available [from Control Room] HVAC: - RCIC/CS Trains A and B - Quad area cooling - AC Switchgear Rooms - Essential Control Building HVAC system - DC Switchgear Rooms - Essential Control Building HVAC system - Battery Rooms - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Rooms - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-V	Reactor Recirculation Motor Generator Set Area

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 12-013 - Evaluation of the SLC Pump Tank and Accessway (Fire Zone 5A) and Refueling Floor (Fire Zone 6) Fire Barrier Separation

Purpose This evaluation is written to justify several unrated and open penetrations in the Fire Area RB-T floor boundaries to the adjacent fire area, including an open equipment hatchway, an open stairwell, and fire doors having a fire resistance rating less than 3 hours. The evaluation documents that the separation between Fire Area RB-T and adjacent fire areas is adequate to demonstrate compliance NFPA 805 section 4.2.3.2.

Conclusion The existing fire area boundary configurations and automatic and manual fire protection features provide a level of protection that are adequate to prevent damage to safe shutdown systems in adjacent fire areas. The performance-based analysis has assessed the adequacy of the fire barrier forming the fire boundary to withstand the fire effects of the hazards in the area.

Basis The bases that justify this conclusion are summarized as follows:

• The combustible loading classification for Fire Zone 5A is "LOW" and is mainly attributed to cable insulation, lube oil, miscellaneous plastic, fiberglass, and PVC.
• The SLC equipment is located within a concrete curbed area which will prevent any spill of oil down to Fire Zone 4A via floor openings.
• The combustible loading classification for Fire Zone 4A is "LOW" and is mainly attributed to cable insulation, miscellaneous fiberglass, and plastic.
• Fixed ignition sources in Fire Zone 4A are limited to small electrical panels, instrument racks, and MCCs. The MCCs are well-sealed, robustly secured panels and are not located adjacent to the open Stair S-3.
• The lack of significant combustible materials in either fire zone and the lack of intervening combustibles in the open Stair S-3 significantly reduce the chance of fire propagation between zones.
• The combustible loading classification for Fire Zone 5B is "MEDIUM" and is mainly attributed to cable insulation, lube oil, charcoal, miscellaneous plastic, and fiberglass.
• The combustible loading classification for Fire Zone 6 is "LOW" and is mainly attributed to lube oil.
• All fire zones are provided with automatic detection system coverage. In the event of a fire, detection system actuation will result in rapid fire brigade response and subsequent manual extinguishment utilizing hose stations and portable extinguishers strategically located in adjacent fire zones.
• Transient combustibles are administratively controlled by CNS procedure 0.7.1, "Control of Combustibles," effectively reducing the possibility of a fire involving transient materials.
• Fire Zone 5B is provided with I-beams at the floor to contain the oil to the area around the MG Sets. The location of the I-beams prevents oil entering the area near the fire doors.

Table B-3 Fire Area Transition

Fire Area	Description
RB-V	Reactor Recirculation Motor Generator Set Area

• The Stud Tensioner in Fire Zone 6 is only used when manned, therefore, a fire involving the oil would be immediately detected. Additionally, the equipment hatchway is provided with steel curbing to prevent oil leakage down to lower elevations.
• A carbon dioxide hose reel is provided along the west wall near MG Set A in Fire Zone 5B. A portable foam hose cart is located near the southwest stairwell in Fire Zone 5B.
• Fixed automatic preaction sprinkler system coverage is provided for the Reactor MG Sets in Fire Zone 5B.

Variances from Deterministic Requirements (VFDR)

RBV-01

Description

Preventing a RR Pump Seal LOCA due to inability to secure the RR Pumps from the Control Room with the REC Non-Critical Header secured (RR Pump Breakers for 4160C-1CS and 4160D-1DS).

Breakers F/FDR to the 4160V Buses from the Startup Transformer: These are normally available, required open breakers provide motive power to the RR Pumps. The RR Pumps are required to trip to prevent a potential seal LOCA when REC is not available to provide cooling. This would challenge the NSCA for Inventory and Pressure Control. REC is secured in this fire area to address potential containment over-pressure. Remote operation of the breaker(s) from the Control Room is lost due to cables H251 and H725 (1C bus), and H291 and H726 (1D bus) fire-induced damage.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	**Description**
RB-V	Reactor Recirculation Motor Generator Set Area

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
5B	Detection	Ionization	R	N	N	N	Y	Y	N
5B	Detection	Heat	R	N	N	N	N	N	N
5B	Detection	Flame	R	N	N	N	N	N	N
5B	Detection	Heat Actuated Devices	R	Y	N	N	Y	Y	N
5B	Suppression	Preaction Sprinkler System	P	N/A	N	N	Y	Y	N
5B	Suppression (manual)	Deluge Water Spray	P	N/A	N	N	N	N	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	Description
RB-V	Reactor Recirculation Motor Generator Set Area

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. In the event of normal operation of the preaction sprinkler system in the Fire Area there is a limited potential for effects from the system. A preaction system is not pressurized with water throughout its piping system and can only discharge water through a sprinkler head that has operated by fusing the heat sensitive link or by inadvertent rupture of the pipe or head. The preaction system has a predischarge alarm by heat actuating devices (HADS) and a water flow alarm that would notify the Control Room of both occurrences. The suppression system for the SBGT Room was modified (DC-86-012) to a manual actuation system. The SBGT filter water spray systems are located in the filter housing assemblies, and the nozzles would discharge within the entire enclosure upon manual actuation. The enclosure is watertight and resultant water would be channeled into a floor drain within the room. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

Fire Zone	Description
11A	Turbine Lube Oil Storage Tank Room
11B	Turbine Building Basement - South
11C	H2 Seal Oil Unit Area
11D	Condenser Pit Area
11E	Reactor Feed Pumps Area
11F	Turbine Building Controlled Corridor 882' Elevation
11G	Steam Jet Air Ejector Room
11H	Mechanical Vacuum Pumps Room
11J	Condensate, Condensate Booster and TEC Pumps Area
11K	Turbine Oil Conditioner Room
11L	Pipe Chase
12A	ISO Phase Bus Duct Area
12B	Turbine Building Controlled Corridor 903' Elevation
12C	Condenser and Heater Bay Areas
12D	Turbine Building Floor - North
12E	Turbine Oil Reservoir Area
12F	Turbine Building Document Storage Vault
13A	Turbine Operating Floor
13B	Non-Critical Switchgear Room
13C	Electrical Shop
13D	Instrument Shop, Instrument Records and Chart Rooms
15	Heating Boiler Room
16	Turbine Building Exhaust Fan Room
17	Water Treatment Building
18A	Machine Shop
18B	Machine Shop Clean Tool Room
18C	Machine Shop Oil Storage Room
18D	Machine Shop Paint Storage Room

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility
18E	Machine Shop Lunch Room and Records Storage Room
19A	Office Building Controlled Corridor 903' Elevation
19B	Office Building Occupancies and Controlled Corridors
19C	Office Building Penthouse
21A	Radwaste Building Basement
21B	Radwaste Building First Floor
21C	Radwaste Building Second Floor
21D	Radwaste Building Third Floor
22A	Augmented Radwaste Building Basement
22B	Augmented Radwaste Building First Floor
22C	Augmented Radwaste Building Second Floor
24	Multi-Purpose Facility

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A will be operated to maintain Suppression Pool temperature	None
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indications [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of Core Spray Train A to maintain RPV level.	TBA-01
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility	
Vital Auxiliaries	Mechanical: - REC will be supplied by SW Train A to provide the cooling supply to the ECCS. - SW Train A will be operated to provide the cooling supply to the REC system, RHR Heat Exchangers, and DG 1. Electrical (AC/DC): - Diesel Generator 1 aligned to 4160F Bus - 125/250 VDC Train A is available [from Control Room] HVAC: - RCIC/CS Train A - Quad area cooling - DG 1 HVAC system - AC Switchgear Rooms - Essential Control Building HVAC system - DC Switchgear Rooms - Essential Control Building HVAC system - Battery Rooms - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Rooms - Essential Control Building HVAC system	TBA-02 TBA-03 TBA-04 TBA-05

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-031 - Evaluation of Critical Switchgear Rooms 1F and 1G Fire Barrier Separation

Purpose The purpose of this evaluation is to document the acceptability of the fire barrier separation that has been provided for Critical Switchgear Rooms 1F and 1G. A fire rating cannot be assigned to the barriers due to miscellaneous door discrepancies, ventilation ducts that do not contain fire dampers at the barriers, and unsealed bus duct penetrations.

Conclusion Based on the expected fire hazards and the existing fire protection features associated with the Critical Switchgear Rooms, the installation of additional fire seals within the bus duct enclosure, fire dampers in the ventilation ductwork, or penetration seal material in the conduits, will not significantly improve fire protection required to ensure redundant safe shutdown capability. The existing fire barrier configurations are adequate for the fire hazards in the areas, and to prevent propagation of fire between adjacent areas.

Basis The bases that justify this conclusion are summarized as follows:

- The wall separating the two Switchgear Room Fire Zones is approximately 12 inches of reinforced concrete.
- On the Switchgear Room side of the wall, the ducts are coated with a 1.5-hour rated fire retardant material.
- The outlet of the supply duct and the inlet of the exhaust duct in the Critical Switchgear Room are provided with 1.5-hour rated fire dampers.
- The exhaust ductwork in the Critical Switchgear Room has a smoke detection system interlock that is arranged to shutdown the operating supply fan.
- Plant procedures effectively reduce the possibility of a fire involving transient ignition sources and combustibles.
- Equivalent fire severity in both Switchgear Rooms is less than 1-minute and consists of an assumed transient combustible load and small quantities of miscellaneous plastic.
- The Corridor side of wall near the ventilation openings is essentially void of any significant combustible loading with the exception of miscellaneous combustibles stored in closed metal lockers.
- A smoke detection system has been provided in both Critical Switchgear Rooms which will result in an alarm to the constantly attended Control Room upon activation.
- Manual fire suppression means has been provided for this area in the form of fire hoses and extinguishers.

Table B-3 Fire Area Transition

Fire Area	**Description**
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire hazards of the areas, and adequate separation has been provided. In general, minor variations to the

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.

Basis

The bases that justify this conclusion are summarized as follows:

- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade.
- Ventilation systems can typically be used for smoke and heat removal.
- Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
- The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202.
- The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105.
- The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201.
- The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104.
- The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104.
- The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6.
- The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited for the acceptability of door R7.
- The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for

Table B-3 Fire Area Transition

Fire Area		Description
TB-A		Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility
		the acceptability of doors R101 and R102. • The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102. • The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115. • Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.
EEEE Title		EE 09-036 - Evaluation of Cable Expansion Room Penetration Seals
	Purpose	This evaluation documents the adequacy of the non-fire rated expansion joints in the Cable Expansion Room Appendix R fire barriers. In addition, conduit penetrations exist in the floor/ceiling assembly penetrating to the Office Building Corridor below that are not sealed with grout to the depth specified in the design details.
	Conclusion	Based on the fire protection features provided in the Cable Expansion Room, including automatic suppression and detection system coverage, the ability to achieve safe shutdown independent of the area, and the lack of safe shutdown equipment/cables and combustible materials in the fire zone adjacent to the Cable Expansion Room, the configurations, as provided, are considered adequate.
	Basis	The bases that justify this conclusion are summarized as follows: • The expansion joints are provided with 14 gauge metal covers. The metal covers will provide some degree of fire protection to impede the spread of fire, smoke, and hot gases to the adjacent fire zones. • An automatic sprinkler system and smoke detectors are provided in the Cable Expansion Room. In the event of a fire in the area, detection system actuation will result in alarm in the Control Room, fire brigade response, and subsequent manual extinguishment utilizing hose stations and portable extinguishers. • The presence of automatic detection and suppression ensures that a fire will be limited by automatic or manual suppression, such that breaching of the barriers via the expansion joints or conduit penetrations will be limited. • Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of a fire.

Table B-3 Fire Area Transition

Fire Area	**Description**
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

EEEE Title EE 09-047 - Doors Required for NFPA 805 Building Separation

Purpose

This evaluation was written to justify fire door assemblies provided in fire-rated barriers that may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. This evaluation documents the adequacy of the configurations that have been provided for fire doors B100, B101, B114A, and D301, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

Conclusion

The fire door configurations have been evaluated (i.e., fire doors B100, B101, B114A, and D301) to provide a level of protection commensurate with the fire hazards of the areas, and adequate separation has been determined to be provided. In general, minor variations to the configurations such as conduit penetrations through the door frames, slightly larger gaps between doors and frames, and doors rated for less than that required of the barrier have been evaluated as acceptable.

Basis

The bases that justify this conclusion are summarized as follows:

- Fire severities do not exceed the fire-rated integrity of the doors.
- Detection and/or suppression systems are present in certain zones discussed. Actuation of the detection systems will prompt rapid fire brigade response, and subsequent manual extinguishment using fire hoses and portable extinguishers strategically located throughout the plant.
- Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
- Certain doors discussed are normally closed with card reader access that is monitored by Security personnel.

Table B-3 Fire Area Transition

Fire Area		Description
TB-A		Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility
EEEE Title		LBDCR 2004-023 - Evaluation of a FHA Revision to Relocate the Fire Barrier Between Fire Area IV/Fire Zone 8D and Fire Area VII/Fire Zone 24
	Purpose	The purpose of this evaluation is to demonstrate the relocated fire barrier between Fire Area IV/Fire Zone 8D and Fire Area VIII/Fire Zone 24 is adequate. The section of the barrier that is being changed is the wall area adjacent to, and including, door H100. The barrier is being moved to the vestibule walls and ceiling on the north side of H100. Door H100 will no longer be considered part of the fire barrier. Door H114 on the east side of the vestibule will be evaluated to show it is adequate for the fire hazards that are present. Door H114 is located on the east side of the vestibule. The door is a single leaf unrated metal door. The door has been barricaded with a 1/2 in. thick carbon steel plate plug welded to the frame of the door. The west side of door H114 has been barricaded with concrete blocks stacked approximately 6 ft high and 3 ft deep.
	Conclusion	The walls and ceiling of the vestibule can be credited as a 3-hour fire rating. Based on the fire hazard analysis, fire door H114 is adequate for the hazards associated with the area.
	Basis	The bases that justify this conclusion are summarized as follows: • The automatic smoke detection system that alarms in the Control Room will result in prompt fire brigade response and manual fire brigade extinguishment. Based on low combustible loading and detection in the area, the fire zone boundaries are adequate to prevent fire spread to adjacent fire areas and fire zones. Pre-fire plans are available for fire brigade use in responding to fire events in the fires zones. • An automatic wet pipe sprinkler system has been provided in the Swing Charger Room adjacent to the west wall of Fire Zone 8C. • Given a fire in this zone, safe shutdown can be accomplished as verified by the Appendix R Safe Shutdown Analysis Report. • Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas. • The construction features associated with door H114 and the vestibule are similar to door R115. Door R115 was evaluated under EE 09-035. • Based on the construction of the barrier and door H114 with welded steel plate and the fire severity present in Fire Area VIII/Fire Zone 24 side of the barrier, the Fire Area VIII/Fire Zone 24 fire would not be expected to breach the barrier. Also, a fire starting in Fire Area VIII/Fire Zone 24 would be mitigated by the automatic suppression system so that it would not provide a significant challenge to barrier. A fire starting in Fire Area IV/Fire Zone 8D would not be expected to spread to Fire Area VIII/Fire Zone 24 due to lack of combustibles in the area, lack of combustibles inside the vestibule, and the construction features of door H114. • The walls and ceiling of the vestibule can be credited as a 3-hour fire rating. Based on the fire hazard

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

analysis, fire door H114 is adequate for the hazards associated with the area.

EEEE Title EE 86-5 - Evaluation of HVAC Ducts and Fire Door Between the Control Room and Controlled Corridor

Purpose This engineering evaluation is being prepared to document the acceptability of the three ventilation duct penetrations that are not provided with fire dampers located in the Control Room south wall, which provides Appendix R fire barrier separation between the Control Room and the Controlled Corridor.

Conclusion Based on the construction of the assembly, the lack of significant fire hazards and combustible loading and the presence of installed fire protection features including fire detection, the configuration that has been provided is adequate for the fire hazards of the areas.

Basis The bases that justify this conclusion are summarized as follows:

- The Control Room is constantly attended. Any anticipated fire would be quickly discovered and extinguished in its incipient stage.
- The two larger vent ducts are at an elevation approximately 12 feet above the floor, which is above the suspended ceilings in the corridor. It is, therefore, unlikely that a fire would cause direct flame exposure to the duct in the corridor.
- The corridor area is essentially void of any significant combustible loading with the exception of miscellaneous combustibles stored in closed metal lockers. There are essentially no combustibles normally located in the corridor near the duct penetrations. It is extremely unlikely that the penetrations would ever be exposed to a significant fire.
- Fire propagation from the Control Room to the corridor is considered inconsequential from a safe shutdown standpoint since the office corridor does not contain any safe shutdown cables or equipment.
- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- There are no openings, such as ventilation grills, on either side of the wall in the large ducts to allow flame propagation to enter the ductwork.
- The Battery Room exhaust duct is a Schedule 80 pipe. Fire propagation into or through this pipe is not considered a credible event, as it is of significant construction so as to prevent fire spread.

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

EEEE Title EE 97-124 - Evaluation of Steam Tunnel East Wall Fire Barrier

Purpose

The purpose of this evaluation is to document the adequacy of the fire barrier separating the Steam Tunnel and the Heater Bay on the 909'-6" Elevation of the Turbine Building. This evaluation was prepared to address the adequacy of the entire barrier separation that has been provided, and to justify the lack of fire-rated penetration seal configurations in the barrier.

Conclusion

The barrier between Zones 2E and 12C is an acceptable configuration with respect to limiting fire spread potential between adjacent areas. Safe shutdown capability is assured for both Fire Zones, as documented in the Safe Shutdown Analysis Report. Therefore, this barrier configuration is adequate for CNS Fire Protection Program purposes.

Basis

The bases that justify this conclusion are summarized as follows:

- The fire hazards and combustible loading in the vicinity of the barrier on the Heater Bay side consists primarily of potential transient combustibles. There are negligible quantities of fixed combustible materials near the barrier. The lack of significant fire hazards and combustible loading precludes the possibility of a fire developing to such an extent as to breach and propagate into the Steam Tunnel.
- There are negligible fire hazards and combustible loading in the vicinity of the barrier on the Steam Tunnel side. The lack of significant fire hazards and combustible loading precludes the possibility of a fire developing to such an extent as to breach the barrier and propagate into the Heater Bay.
- Access to these areas is controlled during power operations.
- Safe shutdown capability is assured for both Fire Zones, as documented in the Safe Shutdown Analysis Report.
- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

Variances from Deterministic Requirements (VFDR)

TBA-01

Description

Preventing a RR Pump Seal LOCA due to inability to secure the RR Pumps from the Control Room with the REC Non-Critical Header secured (RR Pump Breakers for 4160C-1CS and 4160D-1DS).

Breakers F/FDR to the 4160V Buses from the Startup Transformer: These are normally available, required open breakers that provide motive power to the RR Pumps. The RR Pumps are required to trip to prevent a potential seal LOCA when REC is not available to provide cooling. This would challenge the NSPC for Inventory and Pressure Control. REC is secured in this fire area to address potential containment over-pressure. Remote operation of the breaker(s) from the Control Room is lost due to cables H251 and H725 (1C bus), and H291 and H726 (1D bus) fire-induced damage.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with Recovery Action.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

TBA-02

Description

Establish vital auxiliaries by powering the credited 4160F Bus from DG1 (EE-CB-4160F-1FA, EE-CB-4160F-1FS, and EE-CB-4160DG1-EG1).

The NSCA require at least one source of AC power be available. Offsite power from the Emergency Transformer is unavailable due to fire damage. Cable damage (H432 and H433) could result in loss of breaker control or spurious actuation of normally closed EE-CB-4160F-1FA. Breaker 1FA may not open as part of the transfer process due to loss of the Startup Transformer. Cable damage to H443 in the 1FS Breaker may cause it to spuriously close, placing the 4160F Bus on the damaged Emergency Transformer. Cable damage to DG20 could blow the "close breaker" circuit control fuses if the DG selector switch is in "Auto," not allowing the DG1 breaker to close, therefore, not placing the DG1 on the bus. The DG is not rated to normally carry both the Vital and Non-Vital buses. Additionally the DG1 breaker is interlocked with the AF/FA breakers, and will not close if the breaker is not opened. Therefore, cbreakers 1FS, 1FA, DG1-EG1 need to be operated to ensure that the credited critical train of power (4160F) can be energized from the DG1. Operation of DG1 is covered in TBA-05.

This is a separation issue for Vital Auxiliaries, Electrical Power Distribution.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable - Recovery Action carried forward as Defense-in-Depth with modification.

See LAR Attachment G, Table G-1 for Recovery Action details.

Modification: Item S-2.2 of of LAR Attachment S, Table S-2.

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

TBA-03

Description

Loss of Critical Switchgear cooling due cable damage to Cooling Fan (HV¬FAN-EF-SWGR-1F).

Fire damage to cable M1368 will not preclude operation of the EF-SWGR-1F fan from the 1F AC Switchgear Room. Cable damage will affect the operation of ventilation dampers AD-1405 and/or AD-1407 by energizing their solenoids and keeping the dampers open. The NSCA model requires that either train of the IF AC Switchgear Room Cooling Fans be available to ensure the Switchgear remains available post-fire.

This is a separation issue for Vital Auxiliaries, Electrical Power Distribution.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

TBA-04

Description

4160F UV Circuit cable damage affects automatic/remote operation from the Control Room of Critical Pumps for Safe Shutdown (EE-CB-4160F-CSP1A, EE-CB-4160F-RHRP1A, and EE-CB-4160F-RWSP1A).

Fire in the area could result in damage to the 4KV Bus 1F UV circuit, which will cause a trip signal to the CS Pump 1A, RHR Pump 1A, RHRSW Booster Pump 1A breakers. The NSCA require the ability to maintain RPV water level. CS Train A is the credited train for Inventory and Pressure Control in this area. RHR Train A is credited to support the SPC mode of RHR following a fire in this area. The NSPC require a means of Decay Heat Removal post-fire. SPC Train A is credited for Decay Heat Removal. Inability to keep the RHR Pump 1A breaker closed would result in loss of RHR flow to the RHR Heat Exchanger, resulting in loss of SPC Train A. Inability to close the RHRSW Booster Pump 1A breaker results in SW-MOV-MO89A closing, and staying closed, isolating SW flow to the RHR Heat Exchanger. SW provides cooling to the REC, DGs, and RHR systems. REC is required to provide both Quad cooling in support of CS Pump operation, and cooling for the RHR pump and RHR Quad. Additional Quad heat-up due to fire in the TB is not credible.

This is a separation issue for Vital Auxiliaries, Electrical Power Distribution.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with modification.

Modification: Item S-2.2 of of LAR Attachment S, Table S-2.

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

TBA-05

Description

DG1 is the credited source of power for a fire in Fire Area TB-A. Damage to SW Pump 1A (4160F UV circuit), results in the requirement for tripping the DG, or inability to start the DG and place it on the bus.

Cable damage to the 4KV Bus 1F UV circuit will cause a trip signal to the SW Pump 1A Breaker. Loss of the SW pump would result in a loss of SW cooling to the RHR Heat Exchangers, DGs, and REC system cooling to critical loads.

This is a separation issue for Vital Auxiliaries, Electrical Power Distribution.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable with modification.

Modification: Item S-2.2 of of LAR Attachment S, Table S-2.

Table B-3 Fire Area Transition

Fire Area | **Description**

TB-A Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
11A	Detection	Ionization	R	N	N	N	N	N	N
11A	Detection	Heat Actuated Devices	R	Y	N	N	N	Y	N
11A	Suppression	Automatic Water Spray	F	N/A	N	N	N	Y	N
11B	Detection	Ionization	R	N	N	N	N	Y	N
11B	Detection	Heat Actuated Devices	R	Y	N	N	N	Y	N
11B	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	Y	N
11C	Detection	Ionization	R	N	N	N	N	N	N
11C	Detection	Heat Actuated Devices	R	Y	N	N	N	Y	N
11C	Suppression	Automatic Water Spray	F	N/A	N	N	N	Y	N
11D	Detection	Heat	R	N	N	N	N	N	N
11D	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	Y	N
11E	Detection	Heat Activated Devices	R	Y	N	N	N	Y	N
11E	Detection	Heat	R	N	N	N	N	N	N
11E	Suppression	Automatic Water Spray	F	N/A	N	N	N	Y	N
11F	Detection	Ionization	R	N	N	N	N	Y	N
11F	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	Y	N
11G	Detection	Heat	R	N	N	N	N	N	N
11G	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	N	N
11H	Detection	Heat	R	N	N	N	N	N	N
11H	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	N	N
11J	Detection	Ionization	R	N	N	N	N	N	N
11J	Suppression	Automatic Wet-Pipe	P	N/A	N	N	N	Y	N
11K	Detection	Ionization	R	N	N	N	N	N	N
11K	Detection	Heat Actuated Devices	R	Y	N	N	N	Y	N

Table B-3 Fire Area Transition

Fire Area **Description**

TB-A Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

11K	Suppression	Automatic Water Spray	F	N/A	N	N	N	Y	N
11L	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
12A	Detection	Heat	R	N	N	N	N	N	N
12A	Detection	Heat Activated Devices	R	Y	N	N	N	N	N
12A	Suppression	Automatic Water Spray	P	N/A	N	N	N	N	N
12B	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
12C	Detection	Heat Activated Devices	R	Y	N	N	N	Y	N
12C	Detection	Heat	R	N	N	N	N	N	N
12C	Suppression	Automatic Water Spray	P	N/A	N	N	N	Y	N
12D	Detection	Ionization	R	N	N	N	Y	Y	N
12D	Detection	Heat Actuated Devices	R	Y	N	N	Y	Y	N
12D	Suppression	Automatic Wet-Pipe	P	N/A	N	N	N	Y	N
12E	Detection	Heat Actuated Devices	R	Y	N	N	N	Y	N
12E	Detection	Ionization	R	N	N	N	N	N	N
12E	Suppression	Automatic Water Spray	F	N/A	N	N	N	Y	N
12F	Detection	Ionization	R	N	N	N	Y	N	N
12F	Suppression	Automatic Wet-Pipe	F	N/A	N	N	Y	N	N
13A	Detection	Heat	R	Y	N	N	N	Y	N
13A	Detection	Flame	R	N	N	N	N	Y	N
13A	Suppression	Automatic Carbon Dioxide	P	N/A	N	N	N	Y	N
13A	Suppression	Automatic Water Spray	P	N/A	N	N	N	Y	N
13B	Detection	Ionization	R	N	N	N	N	Y	N
13C	Detection	Ionization	R	N	N	N	N	Y	N
13D	Detection	Ionization	R	N	N	N	Y	N	N
15	Detection	Heat	R	N	N	N	N	N	N
15	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	Y	N
16	Detection	Ionization	R	N	N	N	N	N	N
17	Detection	Ionization	R	N	N	N	N	N	N
18A	Detection	Heat	R	N	N	N	N	N	N

Table B-3 Fire Area Transition

Fire Area	Description
TB-A	Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

18A	Detection	Ionization	R	N	N	N	N	N	N
18A	Suppression	Automatic Wet-Pipe	P	N/A	N	N	N	N	N
18B	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	N	N
18C	Detection	Ionization	R	N	N	N	N	N	N
18C	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	N	N
18D	Detection	Ionization	R	N	N	N	N	N	N
18D	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	N	N
18E	Detection	Ionization	R	N	N	N	N	N	N
18E	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	N	N
19A	Suppression	Automatic Wet-Pipe	P	N/A	N	N	Y	N	N
19B	Detection	Ionization	R	N	N	N	Y	N	N
19B	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	N	N
19C	Detection	Ionization	R	N	N	N	N	N	N
19C	Detection	Heat	R	N	N	N	N	N	N
19C	Suppression	Automatic Wet-Pipe	P	N/A	N	N	N	N	N
19C	Suppression (manual)	Water Spray	P	N/A	N	N	N	N	N
21A	Detection	Ionization	R	N	N	N	N	N	N
21B	Detection	Ionization	R	N	N	N	Y	N	N
21B	Detection	Heat	R	N	N	N	Y	N	N
21C	Detection	Heat	R	N	N	N	N	N	N
21C	Detection	Ionization	R	N	N	N	N	N	N
21C	Suppression	Bottled Halon 1301	P	N/A	N	N	N	N	N
21C	Suppression	Automatic Wet-Pipe	P	N/A	N	N	N	N	N
21D	Detection	Ionization	R	N	N	N	N	N	N
21D	Suppression	Automatic Wet-Pipe	P	N/A	N	N	N	N	N
22A	Detection	Ionization	R	N	N	N	N	N	N
22A	Detection	Heat	R	N	N	N	N	N	N
22B	Detection	Ionization	R	N	N	N	Y	N	N
22B	Detection	Flame	R	N	N	N	Y	N	N

Table B-3 Fire Area Transition

Fire Area TB-A

Description Turbine Building, Non-Critical Switchgear Room, Office Building, Radwaste Building, Augmented Radwaste Building, and Multi-Purpose Facility

22B	Detection	Heat	R	N	N	N	Y	N	N
22C	Detection	Ionization	R	N	N	N	N	N	N
22C	Detection	Heat	R	N	N	N	N	N	N
22C	Detection	Flame	R	N	N	N	N	N	N
22C	Suppression	Automatic Water Spray	P	N/A	N	N	N	N	N
24	Detection	Heat	R	Y	N	N	Y	N	N
24	Suppression	Preaction Sprinkler System	P	N/A	N	N	Y	N	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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 Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. Per MWR 91-1264, electrical cabinets and conduit have been sealed to mitigate the effects of suppression activities. In the event of normal operation, the automatic suppression systems will not adversely effect the other equipment operating within the Fire Zone. The drainage features and equipment pedestals mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
TB-C	Steam Tunnel

Fire Zone	Description
2E	Steam Tunnel

Regulatory Basis

4.2.4.2 - Performance-Based Approach - Fire Risk Evaluation with simplifying deterministic assumptions

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A or B will be operated to maintain Suppression Pool temperature	None
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indications [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of either Core Spray Train A or Train B to maintain RPV level.	TBC-01
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
TB-C	Steam Tunnel	
Vital Auxiliaries	Mechanical: - REC will be supplied by SW Train A or B and operated to provide the cooling supply to the ECCS. - SW Train A or B will be operated to provide the cooling supply to the REC system and RHR Heat Exchangers. Electrical (AC/DC): - Offsite Emergency Transformer aligned to 4160F or 4160G Bus - 125/250 VDC Trains A and B are available [from Control Room] HVAC: - CS Trains A and B - Quad area cooling - DG 1 and 2 HVAC system - AC Switchgear Rooms - Essential Control Building HVAC system - DC Switchgear Rooms - Essential Control Building HVAC system - Battery Rooms - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Rooms - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area	**Description**
TB-C	Steam Tunnel

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 05-034 - Evaluation of the Reclassification of Door R104 Under the FHA

Purpose The purpose of this evaluation is to permanently reclassify door R104 from a 3-hour fire door to a 1-hour fire door required to separate Fire Area RB-FN/Fire Zone 2A-1 (Reactor Building 903'-6" Northeast Corner) and Fire Area TB-C/Fire Zone 2E (Steam Tunnel).

Conclusion It is acceptable to revise the fire barrier between Fire Area RB-FN/Fire Zone 2A-1, and Fire Area TB-C/Fire Zone 2E from a 3-hour rated barrier to a 1-hour fire rated barrier.

Basis The bases that justify this conclusion are summarized as follows:

- Ignition sources near door R104 include, but are not limited to: MCC-K, RCIC Rack, and Condensate Pumps CP-R-A1 and A2. Each of these ignition sources is spatially separated from door R104 by approximately 5 feet or more. In the case of MCC-K, as much as approximately 15 feet. A postulated fire at each of these ignition sources would propagate upward into the cable raceways. Horizontal propagation will not occur due to the scarcity and discontinuity of combustibles in the zone. Without horizontal propagation of the postulated fire, no challenge can be made to door R104.
- The combustible loading calculation, NEDC 93-161, shows that the primary contributor of combustibles in Reactor Building Elev. 903'-6" are located 20 feet above the floor in the overhead cable raceways. These cables are fire retardant and are equivalent to IEEE-383 rated cable. Fire in the cable raceways would not present a fire severity of greater than 1 hour to door R104 due to a spatial separation of 10 feet or more.
- This area is also equipped with fire and smoke detection as well as fire suppression. The smoke and fire detections systems would ensure immediate fire brigade response. The fire suppression system alone will mitigate any postulated fire at the floor level, well before the fire reaches a severity of greater than 1 hour.

Table B-3 Fire Area Transition

Fire Area	Description
TB-C	Steam Tunnel

EEEE Title EE 97-124 - Evaluation of Steam Tunnel East Wall Fire Barrier

Purpose The purpose of this evaluation is to document the adequacy of the fire barrier separating the Steam Tunnel and the Heater Bay on the 909'-6" Elevation of the Turbine Building. This evaluation was prepared to address the adequacy of the entire barrier separation that has been provided, and to justify the lack of fire-rated penetration seal configurations in the barrier.

Conclusion The barrier between Zones 2E and 12C is an acceptable configuration with respect to limiting fire spread potential between adjacent areas. Safe shutdown capability is assured for both Fire Zones, as documented in the Safe Shutdown Analysis Report. Therefore, this barrier configuration is adequate for CNS Fire Protection Program purposes.

Basis The bases that justify this conclusion are summarized as follows:

- The fire hazards and combustible loading in the vicinity of the barrier on the Heater Bay side consists primarily of potential transient combustibles. There are negligible quantities of fixed combustible materials near the barrier. The lack of significant fire hazards and combustible loading precludes the possibility of a fire developing to such an extent as to breach and propagate into the Steam Tunnel.
- There are negligible fire hazards and combustible loading in the vicinity of the barrier on the Steam Tunnel side. The lack of significant fire hazards and combustible loading precludes the possibility of a fire developing to such an extent as to breach the barrier and propagate into the Heater Bay.
- Access to these areas is controlled during power operations.
- Safe shutdown capability is assured for both Fire Zones, as documented in the Safe Shutdown Analysis Report.
- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.

Table B-3 Fire Area Transition

Fire Area	Description
TB-C	Steam Tunnel

Variances from Deterministic Requirements (VFDR)

TBC-01

Description

Ability to close SRV's automatically/remotely from the Control Room could be lost due to cable fire damage (Pilot Valves SPV71E, SPV71F, SPV71G, and SPV71H).

The NSPC require the ability to isolate the RPV to maintain water inventory above the active fuel. Spurious operations could result in the opening of up to four ADS valves. Affected ADS SRVs need to be returned to their fail-safe closed position within 18 minutes for single spurious operation.

This is a separation issue for Inventory and Pressure Control, RPV Isolation.

Disposition

A fire risk evaluation was performed using the guidelines of NFPA 805 Section 4.2.4 that determined that this VFDR meets the risk acceptance criteria.

Risk: Acceptable

Modification: None

Required Fire Protection Systems and Features

None

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. The steam tunnel is not subject to an adverse effects to equipment by water intrusion from fire suppression systems. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

None

Table B-3 Fire Area Transition

Fire Area	Description
YD	Yard, Fire Pumphouse, Off-Gas Building, and Optimum Water Chemistry Building

Fire Zone	Description
23A	Electric Motor Driven Fire Pump Room
23B	Diesel Driven Fire Pump Room
23C	Diesel Oil Tank Room
25	Off-Gas Building
26	Optimum Water Chemistry Building
Yard	Transformer Yard

Regulatory Basis

4.2.3.2 - Deterministic Approach

Performance Goal	Method of Accomplishment	Comments / VFDR
Decay Heat Removal	SPC Train A or B will be operated to maintain Suppression Pool temperature	None
Process Monitoring	The following indications will be used to support the Process Monitoring function: - RPV water level and pressure [from Control Room] - Suppression Pool level and temperature [from Control Room] - CS, RHR, and SW flow indications [from Control Room]	None
Inventory and Pressure Control	RPV isolation will be accomplished by manual isolation of main steam lines, other discharge paths inboard of the MSIVs, and other system pressure boundaries. RPV over-pressure protection will be provided by SRVs. Only the self-activated spring lift mode is credited for over-pressure protection. ADS will be used to reduce RPV pressure for operation of either Core Spray Train A or Train B to maintain RPV level.	None
Reactivity Control	Subcritical conditions will be achieved and maintained by insertion of the control rods caused by de-energizing RPS. The reactor scram will be the result of an automatic RPS trip or from operator initiation of a manual trip.	None

Table B-3 Fire Area Transition

Fire Area	Description	
YD	Yard, Fire Pumphouse, Off-Gas Building, and Optimum Water Chemistry Building	
Vital Auxiliaries	Mechanical: - REC will be supplied by Train A or B and operated to provide the cooling supply to the ECCS. - SW Train A or B will be operated to provide the cooling supply to the REC system, RHR Heat Exchangers and DGs. Electrical (AC/DC): - DG 1 or 2 aligned to 4160F or 4160G Bus - 125/250 VDC Trains A and B are available [from Control Room] HVAC: - CS Trains A and B - Quad area cooling - DG 1 and 2 HVAC system - AC Switchgear Rooms - Essential Control Building HVAC system - DC Switchgear Rooms - Essential Control Building HVAC system - Battery Rooms - Essential Control Building HVAC system - Auxiliary Relay Room and RPS MG Set Rooms - Essential Control Building HVAC system	None

Reference Document / Document Detail

CNS Calculation NEDC 11-019 "Nuclear Safety Capability Assessment"

Licensing Actions

None

Table B-3 Fire Area Transition

Fire Area **Description**

YD Yard, Fire Pumphouse, Off-Gas Building, and Optimum Water Chemistry Building

Existing Engineering Equivalency Evaluations (EEEE)

EEEE Title EE 09-035 - Evaluation of Fire Doors

Purpose

Fire door assemblies provided in fire-rated barriers may vary slightly from listed configurations, and may or may not provide the same fire rating as that required of the fire barrier. The evaluation documents the adequacy of the configurations that have been provided for fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115, and justifies the continued use of the configurations, as they have been determined to provide a level of protection that is adequate for the fire hazards in the areas.

- Door D202 separates the Cable Expansion Room (Fire Area CB-D) from Stair A-9 (Fire Area TB-A)
- Door H105 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Turbine Building Floor North (Fire Area TB-A)
- Doors H200 and H201 separate the Cable Spreading Room (Fire Area CB-D) from Stair A-10 (Fire Area CB-A)
- Door H202 separates the Cable Spreading Room (Fire Area CB-D) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door H306 separates the Seal Water Pump Area and Corridor (Fire Area CB-A) from the Computer Room (Fire Area CB-D)
- Door H307 separates the Computer Room (Fire Area CB-D) from the I and C Shop (Fire Area TB-A)
- Door N103 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Turbine Building Mezzanine North (Fire Area TB-A)
- Door N104 separates the Diesel Generator Room 1A (Fire Area DG-A) from the Diesel Generator Room 1B (Fire Area DG-B)
- Door R6 separates the Reactor Building Southeast Quad (Fire Area RB-CF) from the Suppression Pool Area (Fire Area RB-E)
- Door R7 separates the Reactor Building Northwest Quad (Fire Area RB-B) from the Suppression Pool Area (Fire Area RB-E)
- Doors R101 and R102 separate the Reactor Building 903' Northeast Corner (Fire Area RB-FN) from the Office Building Controlled Corridor (Fire Area TB-A)
- Door R115 separates Reactor Building 903'-6" CRD Units - South (Fire Area RB-DI) from the Exterior Transformer Yard (Fire Area YD)

Conclusion

The fire door configurations (i.e., fire doors D202, H105, H200, H201, H202, H306, H307, N103, N104, R6, R7, R101, R102, and R115) have been determined to provide a level of protection commensurate with the fire

Table B-3 Fire Area Transition

Fire Area	Description
YD	Yard, Fire Pumphouse, Off-Gas Building, and Optimum Water Chemistry Building

hazards of the areas, and adequate separation has been provided. In general, minor variations to the configurations, such as conduit penetrations through the door frames, slightly larger gaps between doors and frames, and doors rated for less than that required of the barrier, have been evaluated as acceptable based on the fire hazards on either side of the barrier.

Basis

The bases that justify this conclusion are summarized as follows:

- Transient combustibles, hot work, and ignition sources are controlled by administrative procedures that effectively reduce the possibility of fires in these areas.
- Fire hoses and portable extinguishers are strategically located throughout the plant for use by the responding fire brigade.
- Ventilation systems can typically be used for smoke and heat removal.
- Due to control of transient combustibles and ignition sources, door proximity to permanent combustibles, and fire area combustible loading, the majority of these doors will not be challenged by a credible fire. This effectively reduces the risk of the identified deviations.
- The automatic sprinkler system and smoke detection provided in the Cable Expansion Room (Fire Area CB-D) are credited for the acceptability of door D202.
- The automatic smoke detection systems provided in the Seal Water Pump Area and Corridor (Fire Area CB-A) and in the Turbine Building Floor North (Fire Area TB-A) are credited for the acceptability of door H105.
- The pre-action sprinkler system and automatic smoke detection provided in the Cable Spreading Room (Fire Area CB-D) are credited for the acceptability of doors H200 and H201.
- The automatic total flooding Halon 1301 suppression system and automatic smoke detection provided in the Computer Room (Fire Area CB-D) are credited for the acceptability of doors H306 and H307.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1A (Fire Area DG-A) and the heat detection installed are credited for the acceptability of doors N103 and N104.
- The smoke and heat actuated devices provided in the Turbine Building Mezzanine (Fire Area TB-A) are credited for the acceptability of doors N103.
- The automatic total flooding carbon dioxide system actuated by smoke detectors provided in Diesel Generator Room 1B (Fire Area DG-B) and the heat detection installed are credited for the acceptability of door N104.
- The automatic heat detection provided in the Reactor Building Northwest Quad (Fire Area RB-CF) is credited for the acceptability of door R6.
- The automatic heat detection provided in the Reactor Building Southeast Quad (Fire Area RB-B) is credited for the acceptability of door R7.
- The automatic suppression system provided in the Office Building Corridor (Fire Area TB-A) is credited for

Table B-3 Fire Area Transition

Fire Area	Description
YD	Yard, Fire Pumphouse, Off-Gas Building, and Optimum Water Chemistry Building

the acceptability of doors R101 and R102.

- The automatic sprinkler system and smoke detection provided in the Reactor Building 903' Northeast Corner (Fire Area RB-FN) is credited for the acceptability of doors R101 and R102.
- The automatic deluge suppression system actuated by heat actuated devices provided for the yard transformers (Fire Area YD) are credited for the acceptability of door R115.
- Actuation of the detection systems will prompt rapid fire brigade response and subsequent manual extinguishment.

Variances from Deterministic Requirements (VFDR)

None

Table B-3 Fire Area Transition

Fire Area **Description**

YD Yard, Fire Pumphouse, Off-Gas Building, and Optimum Water Chemistry Building

Required Fire Protection Systems and Features

Fire Zone	Type of System	Specific Type of System	Local (L) Remote (R) Full (F) Partial (P)	Detection Actuates Suppression?	Required System?				
					S	L	E	R	D
23A	Detection	Heat	R	N	N	N	N	N	N
23A	Detection	Ionization	R	N	N	N	N	N	N
23B	Detection	Ionization	R	N	N	N	N	N	N
23B	Detection	Flame	R	N	N	N	N	N	N
23B	Detection	Heat	R	N	N	N	N	N	N
23B	Suppression	Automatic Wet-Pipe	F	N/A	Y	N	N	N	N
23C	Detection	Heat	R	N	N	N	N	N	N
23C	Suppression	Automatic Wet-Pipe	F	N/A	N	N	N	N	N
25	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
26	None	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Yard	Detection	Heat Actuated Devices	R	Y	N	N	Y	Y	N
Yard	Suppression	Deluge Water Spray	P	N/A	N	N	Y	Y	N

Legend:

Table Field: "Required System?"	
S	- Required for Chapter 4 Separation Criteria
L	- Required for NRC-Approved Exemption
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

Table B-3 Fire Area Transition

Fire Area	Description
YD	Yard, Fire Pumphouse, Off-Gas Building, and Optimum Water Chemistry Building

Fire Suppression Activities Effect on Nuclear Safety Performance Criteria

The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate Fire Zone. For this reason, fire brigade activities are not expected to fail components not immediately involved in the fire scenario. It has been concluded that water impingement on cables is not a concern. In the event of normal operation, the transformer deluge systems will not adversely effect the other equipment operating within the yard. The drainage features mitigate the potential for flooding damage due to fire suppression, such that the standing water would not affect safety-related equipment. Since it is shown that suppression effects will not impact the NSPC, the Fire Area configuration is deemed acceptable.

Fire Area Comments

The automatic wet-pipe sprinkler system provided over the diesel fire pump is required for compliance with Section 3.9.4 of NFPA 805.

ATTACHMENT D

NEI 04-02 Non-Power Operational Modes Transition

8 Pages

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.

FAQ 07-0040, Rev. 4, 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

Cooper Nuclear Station (CNS) Procedure 0.50.5 provides the definition for Higher Risk Evolutions (HRE) used during plant outages:

Outage activities, plant configurations, or conditions during shutdown where the plant is more susceptible to an event causing loss of a key safety function.

Procedure 0.50.5 provides the following as examples of HRE and/or elevated risk time periods:

- Transfer of RPS power supplies with SDC in service
- Performance of Procedures 6.1DG.302 or 6.2DG.302
- Lowered inventory operation
- Plant in electrical backfeed lineup

Procedure 0.50.5 uses time to boil and decay heat removal system availability in defining actions associated with HRE. Procedure 0.50.5 does not specify all conditions considered HRE, but the examples above are consistent with the FAQ 07-0040, Revision 4, guidance. A Shutdown Safety Contingency Plan is required when entering any HRE. Time to boil is a specified condition to be considered in developing a Contingency Plan.

Procedure 0.50.5 directs that risk management actions should be developed and implemented to include the fire risk of outage activities.

Reference Documents

1. CNS Procedure 0.50.5, Revision 23, "Outage Shutdown Safety"
2. CNS Procedure 6.1DG.302, Revision 65, "Undervoltage Logic Functional, Load Shedding, and Sequential Loading Test (DIV 1)"
3. CNS Procedure 6.2DG.302, Revision 56, "Undervoltage Logic Functional, Load Shedding, and Sequential Loading Test (DIV 2)"
4. FAQ 07-0040, Revision 4, "Non-Power Operations Clarifications" (ADAMS Accession No. ML082070249), as approved by the NRC (ADAMS Accession No.: ML082200528)

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.

FAQ 07-0040, Rev. 4, Implementing Guidance F.2

Identify Components and Cables.

The identification of systems and components to be included in this Non-Power Operations (NPO) Review begins with the identification of the Plant Operational States (POS) that need to be considered.

Review

CNS Procedure 0.50.5, Attachment 1, defines the following Key Safety Functions (KSFs):

- Decay Heat Removal
- Fuel Pool Cooling
- Inventory Control
- Containment
- Power Supplies
- Reactivity Control

Based on FAQ 07-0040, Revision 4, the POS considered for equipment and cable selection in CNS Calculation NEDC 11-003 are:

POS 1	This POS starts when the RHR system is put into service. The RPV head is on and the RCS is closed such that an extended loss of the decay heat removal function without operator intervention could result in a RCS re-pressurization above the shutoff head for the RHR pumps.
POS 2	This POS represents the shutdown condition when: (1) the RPV head is removed and RPV water level is less than the minimum level required for movement of irradiated fuel assemblies within the RPV, as defined by Technical Specifications, or (2) a sufficient RCS vent path exists for decay heat removal.
POS 3	This POS represents the shutdown condition when the RPV water level is equal to, or greater than, the minimum level required for movement of irradiated fuel assemblies within the RPV as define by Technical Specifications. This POS occurs during Mode 5.

The evaluation of these POS resulted in the exclusion of Fuel Pool Cooling, Containment, and Reactivity Control KSF from further consideration based on the following justifications:

- Fuel Pool Cooling is required in order to prevent boiling and the resulting loss of inventory, which can cause damage to the stored fuel cells when they are uncovered. However, per USAR Section X-3.6.3, there is sufficient time to establish adequate makeup to the Spent Fuel Pool prior to the onset of bulk boiling. Plant procedures require logging the temperature starting at every four hours in the event that cooling is lost. A number of options are available for replenishing the water to prevent uncovering

the fuel, including the use of fire hoses, or cross-ties to the RHR system. Therefore, it is unnecessary to model the Fuel Pool Cooling system.

- The Containment KSF specifies the Secondary Containment as the equipment/volume of concern during NPO modes. No specific equipment has been modeled to satisfy this KSF since administrative and procedural controls are credited. The requirements for containment closure capability are contained in CNS Procedure 0.50.5, "Outage Shutdown Safety". Maintenance of decay heat removal and inventory KSF will preclude the need for establishing containment during the times when containment may be relaxed. When containment must be intact, the equipment used is covered by the CNS Technical Specifications, plant design, and procedural or administrative controls.
- The NPO analysis excludes the Reactor Protection System since the plant is already in a shutdown state with all control rods inserted (or reactivity controls are in place per the CNS Technical Specifications to ensure the reactor stays shutdown). No additional equipment is required for the reactivity KSF as the control rods are designed such that sufficient negative reactivity is inserted to ensure K_{eff} <0.99 even with the most reactive rod stuck in the fully withdrawn position.

In CNS Calculation NEDC 11-003, the components relied upon to provide the remaining KSF including support functions were identified. The selection of equipment was further broken down and related by KSF success paths. Power supplies and other supporting components such as interlocks were identified, listed, and tied with their component and KSF success paths in the SAFE fire protection safe shutdown analysis software. For those components not already in SAFE, cable selection and routing were performed in accordance with the Nuclear Safety Capability Assessment (NSCA) methodology, CNS Calculation NEDC 11-019. The NSCA methodology identified all required cables associated with a component and this information was added to SAFE. The SAFE software was run and the resulting analysis by fire area identified "pinch points," that is, fire areas where a KSF success path was not available.

Reference Documents

1. CNS Procedure 0.50.5, Revision 23, "Outage Shutdown Safety"
2. FAQ 07-0040, Revision 4, "Non-Power Operations Clarifications" (ADAMS Accession No. ML082070249), as approved by the NRC (ADAMS Accession No.: ML082200528)
3. CNS Calculation NEDC 11-019, Revision 0, "EPM Report: R1906-004-002, Nuclear Safety Capability Assessment"
4. CNS Calculation NEDC 11-003, Revision 0, "Non-Power Operation Modes Transition Review of EPM Report R1906-006-001"

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.

FAQ 07-0040, Rev. 4, Implementing Guidance F.3

Perform Fire Area Assessments (identify pinch points).

Identify locations where:

- Fires may cause damage to the equipment (and cabling) credited above, or
- KSF 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 compartment are expected to damage equipment (and cabling) thereby eliminating a pinch point.

Review

The NPO fire area reviews conservatively assumed that the entire contents of a fire area would be lost. These reviews identified that there are fire areas where a single fire could result in a loss of all credited paths for a given KSF (i.e., a pinch point). The review also identified that there are certain fire areas that are vulnerable to a loss of a KSF if certain system trains or components are taken out of service during a non-power operational mode and a fire were to occur (i.e. fire area where only a single path is recovered). Fire areas where a fire might cause damage to equipment required to support a KSF path are identified in CNS Calculation NEDC 11-003.

Twenty-one (21) generic fire area pinch points (i.e. a fire area which contains at least one KSF with a pinch point) were identified during the performance of the NPO reviews of the twenty-five (25) fire areas analyzed. Details of the pinch points and fire areas containing the pinch points are contained in CNS Calculation NEDC 11-003.

The assessments that were performed as part of the NPO review conservatively assumed that all NPO components or component cables in the fire area may be lost due to a fire. Using the review methodology outlined in CNS Calculation NEDC 11-003 and the recovery approaches that were developed to alleviate the identified "pinch points," which are also presented in the calculation, precluded the need to utilize fire modeling in order to achieve a KSF.

Reference Document

1. CNS Calculation NEDC 11-003, Revision 0, "Non-Power Operation Modes Transition Review of EPM Report R1906-006-001"

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.

FAQ 07-0040, Rev. 4, 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 would cause equipment damage 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

Ensure that the normal fire protection defense-in-depth features are applicable during NPO modes.

During those NPO evolutions that are defined as HRE:

Additional fire protection defense in depth measures will be taken during HRE 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 KSF perform as needed during the various outage evolutions. During outage planning, the NPO compartment 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 defense-in-depth

In addition, for KSF Equipment removed from service during the HRE the impact should be evaluated based on KSF equipment status and the NPO Compartment Assessment to develop needed contingency plans/actions.

Review

To preclude or mitigate the KSF failures, a number of strategies have been developed. These strategies include revisions to plant shutdown and abnormal operating procedures. These procedural revisions will make changes to plant equipment and electrical system line-ups as the plant is brought to cold shutdown conditions. Plant operational procedures will also be revised to include protective strategies, preemptive actions, and recovery actions for those instances where operator actions would be necessary to ensure that a specific KSF can be maintained. (See Implementation Item S-3.4 of Attachment S, Table S-3). The pinch points that were identified are documented in CNS Calculation NEDC 11-003.

CNS Procedure 0.50.5, "Outage Shutdown Safety," requires the effect of outage work scope on the KSFs to be qualitatively assessed. When entering an HRE, a Shutdown Safety Contingency Plan is required. Risk management actions are recommended when any of the following occur:

- Performance of maintenance activities with potential to cause a fire.
- Removal of fire detection or suppression equipment from service.
- Removal or impairment of fire barriers.
- Removal of KSF equipment from service.

Risk management actions include:

- Compensatory measures for the temporary removal of fire barriers.
- Increased fire watches.
- Re-scheduling activities that involve increased fire likelihood in fire areas where the inservice KSF equipment would be relied upon in the event of a fire.

To address concerns associated with equipment being taken out of service during NPO modes, and the potential for a concurrent fire, CNS procedures will be reviewed and revised, as necessary, to provide instructions that will assist in mitigating the effects of a fire if one were to occur. The procedure revisions will provide guidelines for actions to be taken in specific fire areas when components or system trains are taken out of service. (See Implementation Item S-3.4 of Attachment S, Table S-3). For those fire areas where the credited KSF system or equipment may be been taken out of service by a fire in that area, the following guidelines may be included in the outage management procedure:

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

Reference Documents

1. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," dated December 1991
2. CNS Calculation NEDC 11-003, Revision 0, "Non-Power Operation Modes Transition Review of EPM Report R1906-006-001"
3. CNS Procedure 0.50.5, Revision 23, "Outage Shutdown Safety"

ATTACHMENT E

NEI 04-02 Radioactive Release Transition

96 Pages

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
RB-A	1A	RCIC and Core Spray A Pump Room	CNS-FP-211	N03	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sump B receives drainage flow from the northeast Reactor Building Quad and from upper levels of the Reactor Building. The sump is provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							contamination contained in the smoke. Ventilation supply air is provided to the Quad room and flows through wall penetrations to the Suppression Chamber area where it enters exhaust ducts to the main exhaust system. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-B	1B	Core Spray B Pump Room	CNS-FP-212	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sump D receives drainage flow from the southeast Reactor Building Quad and from upper levels of the Reactor Building. The sump is provided with	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the Quad room and flows through wall penetrations to the Suppression Chamber area where it enters exhaust ducts to the main exhaust system.	undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-CF	1C	RHR Pump Room 1A and 1C	CNS-FP-213	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sump A receives drainage flow from the northwest Reactor Building Quad and from upper levels of the Reactor Building. The sump is provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station.	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the Quad room and flows through wall penetrations to the Suppression Chamber area where it enters exhaust ducts to the main exhaust system. In addition, Quad communication with the 903'-6" Elevation is provided via grated openings. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not	HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-DI	1D	RHR Pump Room 1B and 1D	CNS-FP-214	N03	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sump C receives drainage flow from the southwest Reactor Building Quad and from upper levels of the Reactor Building. The sump is provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the Quad room and flows through wall penetrations to the Suppression Chamber area where it enters exhaust ducts to the main exhaust system. In addition, Quad communication with the 903'-6" Elevation is provided via grated openings. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to	in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-DI	1E	HPCI Pump Room	CNS-FP-214	N03	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sump C receives drainage flow from the southwest Reactor Building Quad and from upper levels of the Reactor Building. The sump is provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the HPCI Room and flows through wall penetrations to the RHR Pump 1B and 1D Room (Fire Zone 1D) and then through to the Suppression Chamber area and then to the exhaust plenum. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also	Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-B	1G	Hydraulic Drive Pump Area	CNS-FP-212	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sump D receives drainage flow from the southeast Reactor Building Quad and from upper levels of the Reactor Building. The sump is provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP).	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the Quad room and flows through wall penetrations to the Suppression Chamber area where it enters exhaust ducts to the main exhaust system. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-E	1F	Suppression Pool Area	N/A	N/A	Yes	N/A	N/A	N/A	N/A - Screened Out
RB-FN	2A-1	903'-6" Northeast	CNS-FP-215	N04	No	Floor drains are routed to monitored Liquid	The Reactor Building (RB) Heating and Ventilating	Training materials reinforce use of the pre-fire plans.	Based on the availability of engineered controls for

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
		Corner				Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the Reactor Building Quads and from upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents for fires where NPO openings may exist.	System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the	Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							general 903'-6" Elevation area. Exhaust is through areas fed from the general area. During NPO (Refueling Activities): Per CNS Technical Specifications, there are periods when Secondary Containment and/or the Control Room Emergency Filter System (CREFS) and SGT System are off-line. During these periods of potential radioactive release, procedures are in place to address fire risk associated with maintenance activities and to close the Reactor Building in a short period of time, which will mitigate consequences of a radioactive release. Therefore, contaminated smoke migration from the Reactor Building is anticipated to be insignificant. During NPO (Remaining outage activities): Per CNS Technical Specifications, the Secondary Containment may be relaxed and the CREFS and SGT System may be offline during this period. As such, procedures are in place to monitor transient combustibles entering the Reactor Building. A credible worst case fire was postulated in NEDC 11-148. The calculation concluded that		

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							the maximum exposure dose limits of 10 CFR 20 would not be exceeded, even when conservatively assuming non-operability of the filtration system and 100% instantaneous airborne dose release from the Reactor Building. For any fires inside the Reactor Building, it is reasonably anticipated that the dose impact would be less than a Fuel Handling Accident, which is a design basis event. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-CF	2A-2	CRD Units - North	CNS-FP-215	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the Reactor Building Quads and from upper	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents for fires where NPO openings may exist.	prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the general 903'-6" Elevation area. Exhaust is through areas fed from the general area.	Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							During NPO (Refueling Activities): Per CNS Technical Specifications, there are periods when Secondary Containment and/or the Control Room Emergency Filter System (CREFS) and SGT System are off-line. During these periods of potential radioactive release, procedures are in place to address fire risk associated with maintenance activities and to close the Reactor Building in a short period of time, which will mitigate consequences of a radioactive release. Therefore, contaminated smoke migration from the Reactor Building is anticipated to be insignificant. During NPO (Remaining outage activities): Per CNS Technical Specifications, the Secondary Containment may be relaxed and the CREFS and SGT System may be offline during this period. As such, procedures are in place to monitor transient combustibles entering the Reactor Building. A credible worst case fire was postulated in NEDC 11-148. The calculation concluded that the maximum exposure dose limits of 10 CFR 20 would not be exceeded, even when conservatively assuming non-operability of		

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							the filtration system and 100% instantaneous airborne dose release from the Reactor Building. For any fires inside the Reactor Building, it is reasonably anticipated that the dose impact would be less than a Fuel Handling Accident, which is a design basis event. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-DI	2A-3	903'-6" South Corridor	CNS-FP-215	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the Reactor Building Quads and from upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents for fires where NPO openings may exist.	Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the general 903'-6" Elevation area. Exhaust is through areas fed from the general area. During NPO (Refueling Activities): Per CNS Technical Specifications, there are periods when Secondary Containment	smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls		Training and Pre-Fire Plan Review Results	Conclusions
						Water	Smoke		
							and/or the Control Room Emergency Filter System (CREFS) and SGT System are off-line. During these periods of potential radioactive release, procedures are in place to address fire risk associated with maintenance activities and to close the Reactor Building in a short period of time, which will mitigate consequences of a radioactive release. Therefore, contaminated smoke migration from the Reactor Building is anticipated to be insignificant. During NPO (Remaining outage activities): Per CNS Technical Specifications, the Secondary Containment may be relaxed and the CREFS and SGT System may be offline during this period. As such, procedures are in place to monitor transient combustibles entering the Reactor Building. A credible worst case fire was postulated in NEDC 11-148. The calculation concluded that the maximum exposure dose limits of 10 CFR 20 would not be exceeded, even when conservatively assuming non-operability of the filtration system and 100% instantaneous airborne dose release from the Reactor Building. For any fires inside the Reactor		

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							Building, it is reasonably anticipated that the dose impact would be less than a Fuel Handling Accident, which is a design basis event. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-CF	2B	RHR HX-1A	CNS-FP-215 CNS-FP-218	N04 N10	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the Reactor Building Quads and from upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air to the RHR Heat Exchanger Room is from the 903'-6" and 931'-6" Elevation general areas. Exhaust from the RHR Heat Exchanger Room is to the exhaust plenum. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard	describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-DI	2C	CRD Units - South	CNS-FP-215	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the Reactor Building Quads and from upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents for fires where NPO openings may exist.	the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the 903'-6" Elevation general area. Exhaust is through areas fed from the general area. During NPO (Refueling Activities): Per CNS Technical Specifications, there are periods when Secondary Containment and/or the Control Room Emergency Filter System (CREFS) and SGT System are off-line. During these periods of potential radioactive release, procedures are in place to address fire risk associated with maintenance activities and to close the Reactor Building in a short period of time, which will mitigate consequences of a radioactive release. Therefore, contaminated	will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							smoke migration from the Reactor Building is anticipated to be insignificant. During NPO (Remaining outage activities): Per CNS Technical Specifications, the Secondary Containment may be relaxed and the CREFS and SGT System may be offline during this period. As such, procedures are in place to monitor transient combustibles entering the Reactor Building. A credible worst case fire was postulated in NEDC 11-148. The calculation concluded that the maximum exposure dose limits of 10 CFR 20 would not be exceeded, even when conservatively assuming non-operability of the filtration system and 100% instantaneous airborne dose release from the Reactor Building. For any fires inside the Reactor Building, it is reasonably anticipated that the dose impact would be less than a Fuel Handling Accident, which is a design basis event. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal		

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-DI	2D	RHR HX-1B	CNS-FP-215 CNS-FP-218	N04 N10	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the Reactor Building Quads and from upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 -	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air to the RHR Heat Exchanger Room is from the 903'-6" and 931'-6" Elevation general areas. Exhaust from the RHR Heat Exchanger Room is to the exhaust plenum. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing	Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls		Training and Pre-Fire Plan Review Results	Conclusions
						Water	Smoke		
							contaminated gaseous effluents.		
TB-C	2E	Steam Tunnel	CNS-FP-215	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the Reactor Building Quads and from upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	There is no individual supply or exhaust air for the steam tunnel. Smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	Training materials reinforce the use of pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for safe removal of contaminated smoke and water runoff in these potentially contaminated areas. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.
RB-J	3A	SWGR Room 1F	CNS-FP-216	N03	Yes	N/A	N/A	N/A	N/A - Screened Out
RB-K	3B	SWGR Room 1G	CNS-FP-217	N03	Yes	N/A	N/A	N/A	N/A - Screened Out
RB-M	3C	REC HX and Pump Area	CNS-FP-218	N10	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas,	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air provided to the 931'-6" Elevation general area. Exhaust is through areas fed from the general	provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							area. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-M	3D	Reactor MG Set Lube Oil Cooler Area	CNS-FP-218	N10	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air provided to the 931'-6" Elevation general area. Exhaust is through areas fed from the general area. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable	describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-N	3E-1	Regenerative HX Areas	CNS-FP-218	N10	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						removal.	(SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. This area has a high potential for contamination and the exhaust air is first passed through banks of prefilters and HEPA filters before discharging into the main exhaust system. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also	performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-N	3E-2	RWCU Recirc Pumps and Corridor	CNS-FP-218	N10	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP).	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. This area has a high potential for contamination and the exhaust air is first passed through banks of prefilters and HEPA filters before discharging into the main exhaust system. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-P	4A	RB Elevator and Accessway Area	CNS-FP-219	N10	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially	Based on the availability of engineered controls for both smoke and fire suppression water runoff,

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						receive drainage flow from the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the 958'-3" Elevation general area. Exhaust is through	contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							areas fed from the general area and through to the exhaust plenum. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-P	4B	RB HVAC Area	CNS-FP-219	N10	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the 958'-3" Elevation general area. Exhaust is through areas fed from the general area and through to the exhaust plenum. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be	Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-P	4C	Fuel Pool HX, CRD Repair Room and Raw Water Cleanup Areas	CNS-FP-219	N10	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the 958'-3" Elevation general area. Exhaust is through areas fed from the general area and through to the exhaust plenum. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with	materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-P	4D	Reactor MG Set Oil Pump Area	CNS-FP-219	N10	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							(ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the 958'-3" Elevation general area. Exhaust is through areas fed from the general area and through to the exhaust plenum. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	Attachment S, Table S-3.	
RB-T	5A	SBLC Pump Tank and Accessway	CNS-FP-220	N05	No	Floor drains are routed to monitored Liquid Radwaste System (LRW).	The Reactor Building (RB) Heating and Ventilating System provides for the	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will	Based on the availability of engineered controls for both smoke and fire

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						Sumps A, B, C, and D receive drainage flow from the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the Standby Liquid Control	identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							System area. Exhaust from this area is through areas it feeds and eventually through to the exhaust plenum. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-V	5B	Reactor MG Set Area	CNS-FP-221 CNS-FP-222	N05 N03	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. The two Reactor Recirculation Motor Generator sets have their own common ventilation system. High radiation level at the RB Isolation Ventilation Radiation Monitoring System will isolate this system. During NPO: The ventilation and monitoring systems as described above remain	these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
RB-T	6	Refueling Floor	CNS-FP-223	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Sumps A, B, C, and D receive drainage flow from the upper levels of the Reactor Building. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions	The Reactor Building (RB) Heating and Ventilating System provides for the movement of air from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust from the building. Two systems monitor the air flow for radioactivity: (1) The RB Ventilation Radiation Monitoring System continuously monitors and records the radiation level at a sample point located in the RB ventilation exhaust vent just prior to the release point. The RB Ventilation Radiation Monitoring System alerts the Control Room when the radiation	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls		Training and Pre-Fire Plan Review Results	Conclusions
						Water	Smoke		
						are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	level is abnormal. (2) The RB Isolation Ventilation Monitoring System monitors the flow of air through the RB plenum and will isolate the RB and initiate the Standby Gas Treatment (SGT) System on high radiation levels. Initiation of the SGT System will process the RB air via its banks of filters, HEPA, and carbon absorbers prior to safely discharging it from the Elevated Release Point (ERP). The ventilation of smoke is accomplished by the RB Heating and Ventilating System governed by the level of radioactivity of the contamination contained in the smoke. Ventilation supply air is provided to the refueling floor, 1001'-0" Elevation general area. Exhaust is through areas fed from the general area and through to the exhaust plenum. During NPO: The ventilation and monitoring systems as described above remain present. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal	and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
CB-A	7A	RHR Service Water Booster Pump and Service Air Compressor Areas	CNS-FP-224	N04	No	Fire Zone 7A is not part of the RCA, however, there is a small section in the southwest corner that is taped off as a contaminated area. Based on RP radiation survey, the level of surface contamination, as well as radiation levels for the taped-off area, was very slight. Control Building non-radioactive floor drains discharge to the "L" sump. Sump pumps L1 and L2 are normally lined up to discharge to the radwaste drain collector tank. If Sump Pumps L1 and L2 are lined up to discharge to the river, Radiation Protection (RP) analyzes the contaminated liquid effluents prior to placing in operation. The potential contaminated water runoff resulting from fire fighting activities in the taped-off	Fire Zone 7A is not part of the RCA, however, there is a small section in the southwest corner that is taped off as a contaminated area. Based on RP radiation survey, the level of surface contamination, as well as radiation levels for the taped-off area, was very slight. The potential quantity of contaminated smoke resulting from fire fighting activities in the taped off area is deemed to be very insignificant. The Control Building Heating and Ventilating system or manual smoke removal techniques can be used to remove the smoke. However, prior to any release, the radiological hazards associated with releasing contaminated smoke from the building will be considered, and direct communication between the Fire Brigade and Radiation Protection will be initiated.	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 -	As the fire zone has been deemed to be of negligible consequence, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						area is deemed to be very insignificant. In addition, provisions are in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents for a fire event.	The quantity of contaminated smoke released to the atmosphere is reasonably anticipated to be well below the maximum exposure dose limits of 10 CFR 20.	Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
CB-A	7B	Emergency Condensate Storage TK. Area	CNS-FP-224	N04	No	Fire Zone 7B is not part of the RCA, however, the area has contamination potential due to the presence of the Emergency Condensate Storage Tanks (ECST). Control Building non-radioactive floor drains discharge to the "L" sump. Sump pumps L1 and L2 are normally lined up to discharge to the radwaste drain collector tank. If Sump Pumps L1 and L2 are lined up to discharge to the river, Radiation Protection (RP) analyzes the contaminated liquid effluents prior to placing in operation. The potential contaminated water runoff resulting from fire fighting activities is deemed to be very insignificant based on the contamination level. In addition, provisions are in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid	Fire Zone 7B is not part of the RCA, however, the area has contamination potential due to the presence of the Emergency Condensate Storage Tanks (ECST). The potential quantity of contaminated smoke resulting from fire fighting activities is deemed to be very insignificant based on the low contamination level. The Control Building Heating and Ventilating system or manual smoke removal techniques can be used to remove the smoke. However, prior to any release, the radiological hazards associated with releasing contaminated smoke from the building will be considered and direct communication between the Fire Brigade and Radiation Protection will be initiated. The quantity of contaminated smoke released to the atmosphere is reasonably anticipated to be well below the maximum exposure dose limits of 10 CFR 20.	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive	As the fire zone has been deemed to be of negligible consequence, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						effluents for a fire event.		release requirements of NFPA 805. See Attachment S, Table S-3.	
CB-D	8A	Aux Relay Room	CNS-FP-225	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-C	8B	RPS Room 1B	CNS-FP-226	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-A	8C	RPS Room 1A	CNS-FP-227	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-A	8D	Seal Water Pump Area and Corridor	CNS-FP-227	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-A-1	8E	Battery Room 1A	CNS-FP-227	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-B	8F	Battery Room 1B	CNS-FP-228	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-B	8G	DC SWGR Room 1B	CNS-FP-228	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-A-1	8H	DC SWGR Room 1A	CNS-FP-227	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-D	9A	Cable Spreading Room	CNS-FP-229	N04	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-D	9B	Cable Expansion Room	CNS-FP-234	N03	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-D	10A	Computer Room	CNS-FP-230	N06	Yes	N/A	N/A	N/A	N/A - Screened Out
CB-D	10B	Control Room and SAS Corridor	CNS-FP-230	N06	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	11A	Turbine Lube Oil Storage TK Room	CNS-FP-242	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available,	Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	11B	Turbine Bldg Basement - South	CNS-FP-242	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						effluents prior to safe removal.	Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous	in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							effluents.		
TB-A	11C	H2 Seal Oil Unit Area	CNS-FP-242	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	11D	Condenser Pit Area	CNS-FP-246	N03	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release,	fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	11E	Reactor Feed Pumps Area	CNS-FP-243	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
TB-A	11F	TB Controlled Corridor 882 Elev	CNS-FP-243 CNS-FP-244	N04 N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available,	Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	11G	Steam Jet Air Ejector Room	CNS-FP-243 CNS-FP-244	N04 N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						effluents prior to safe removal.	Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous	in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							effluents.		
TB-A	11H	Mechanical Vacuum Pumps Room	CNS-FP-244 CNS-FP-245	N04 N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	11J	Condensate, Condensate Booster and TEC Pumps Area	CNS-FP-245	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release,	fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	11K	Turbine Oil Conditioner Room	CNS-FP-245	N04	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls		Training and Pre-Fire Plan Review Results	Conclusions
						Water	Smoke		
							control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
TB-A	11L	Pipe Chase	CNS-FP-243	N04	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	12A	ISO Phase Bus Duct Area	CNS-FP-247	N06	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	12B	TB Controlled Corridor 903 Elev	CNS-FP-248	N07	No	Floor drains are routed to monitored Liquid Radwaste System (LRW).	The Turbine Building Heating and Ventilating system supplies 100%	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will	Based on the availability of engineered controls for both smoke and fire

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke	identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	12C	Condenser and Heater Bay Areas	CNS-FP-248	N07	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also	are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	12D	Turbine Bldg Floor-North	CNS-FP-249	N06	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	12E	Turbine Oil Reservoir Area	CNS-FP-249	N06	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	12F	Turbine Bldg Document Storage Vault	CNS-FP-250	N04	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	13A	Turbine Operating Floor	CNS-FP-251	N08	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
TB-A	13B	Non-Critical SWGR Room	CNS-FP-252	N03	Yes	N/A	N/A	N/A	N/A - Screened Out

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Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
TB-A	13C	Electrical Shop	CNS-FP-253	N06	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	13D	Instrument Shop, Instrument Records and Chart Rooms	CNS-FP-253	N06	Yes	N/A	N/A	N/A	N/A - Screened Out
DG-A	14A	Emergency Diesel Generator 1A Room	CNS-FP-236	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
DG-B	14B	Emergency Diesel Generator 1B Room	CNS-FP-237	N03	Yes	N/A	N/A	N/A	N/A - Screened Out
DG-A	14C	DG 1A Diesel Oil Day TK. Room	CNS-FP-236	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
DG-B	14D	DG 1B Diesel Oil Day TK. Room	CNS-FP-237	N03	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	15	Heating Boiler Room	CNS-FP-238	N05	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	16	Turbine Bldg Exhaust Fan Room	CNS-FP-239	N03	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station.	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish	potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

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Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	17	Water Treatment Bldg	CNS-FP-240 CNS-FP-241	N06 N05	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	18A	Machine Shop	CNS-FP-254 CNS-FP-255	N06 N05	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
TB-A	18B	Machine Shop Clean Tool Room	CNS-FP-254	N06	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	18C	Machine Shop Oil Storage Room	CNS-FP-254	N06	Yes	N/A	N/A	N/A	N/A - Screened Out

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
TB-A	18D	Machine Shop Paint Storage Room	CNS-FP-254	N06	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	18E	Machine Shop Lunch Room and Records Storage Room	CNS-FP-255	N05	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Turbine Building radioactive floor drain sumps receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank or to disposal. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	The Turbine Building Heating and Ventilating system supplies 100% filtered outside air to all areas of the Turbine Building. Air movement is from clean areas to areas with progressively greater contamination potential. Exhaust air is discharged to the atmosphere above the roof of the Fan Room via four exhaust fans. The exhaust system does not contain filtration equipment. The fans are interlocked and are provided with alarms that annunciate in the Control Room. Control Room operators control the operation of the exhaust fans. The Turbine Building Ventilation Radiation Monitoring system is installed to provide a continuous record of exhaust air flow and contamination activity, and to alert the Control Room of such abnormal air activity. The monitoring system does not perform any control functions (e.g. exhaust fan shutdown). Due to the configuration of the Turbine Building Heating and Ventilating system (air movement from	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							clean areas to areas with greater contamination) and the ability of the Control Room to monitor potentially contaminated smoke and shut down exhaust fans, contaminated smoke migration from Turbine Building areas is anticipated to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	19A	Office Bldg Controlled Corridor 903 Elev	CNS-FP-231	N06	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	19B	Office Bldg Occupancies and Controlled Corridors	CNS-FP-232 CNS-FP-233 CNS-FP-235	N08 N09 N07	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	19C	Office Building Penthouse	CNS-FP-269	N05	Yes	N/A	N/A	N/A	N/A - Screened Out

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
IS-A	20A	Service Water Pump Area	CNS-FP-256	N04	Yes	N/A	N/A	N/A	N/A - Screened Out
IS-A	20B	Circ Water Pump and Traveling Screen Area	CNS-FP-256	N04	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	21A	Radwaste Bldg Basement	CNS-FP-257	N07	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Radwaste Building floor drain sumps H and K receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	Ventilation air from areas containing radioactive materials and equipment are monitored by the Radwaste/Augmented Radwaste Building Ventilation Radiation Monitoring System and is discharged to a common exhaust vent for both the Radwaste Building and Augmented Radwaste Building HVAC systems. For the potentially contaminated exhaust, two separate full capacity filter assemblies are installed upstream of the exhaust vent. Each filter assembly is isolated automatically with fan operation. Prefilter and HEPA filter differential pressures are indicated outside each filter assembly compartment. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	Attachment S, Table S-3.	
TB-A	21B	Radwaste Bldg First Floor	CNS-FP-258 CNS-FP-259	N03 N03	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Radwaste Building floor drain sumps H and K receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	Ventilation air from areas containing radioactive materials and equipment are monitored by the Radwaste/Augmented Radwaste Building Ventilation Radiation Monitoring System and is discharged to a common exhaust vent for both the Radwaste Building and Augmented Radwaste Building HVAC systems. For the potentially contaminated exhaust, two separate full capacity filter assemblies are installed upstream of the exhaust vent. Each filter assembly is isolated automatically with fan operation. Prefilter and HEPA filter differential pressures are indicated outside each filter assembly compartment. This area contains solids, liquids, and gaseous waste that ranges from highly radioactive to low level activity. For the highly radioactive waste (e.g. RWCU-related), these wastes are addressed with thick concrete shielding,	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							storage drums, etc., where a fire is unlikely. However, the pre-fire plans identify that for fires involving radioactive materials, the Control Room and Radiation Protection personnel be notified. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	release requirements of NFPA 805. See Attachment S, Table S-3.	
TB-A	21C	Radwaste Bldg Second Floor	CNS-FP-260	N06	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Radwaste Building floor drain sumps H and K receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe	Ventilation air from areas containing radioactive materials and equipment are monitored by the Radwaste/Augmented Radwaste Building Ventilation Radiation Monitoring System and is discharged to a common exhaust vent for both the Radwaste Building and Augmented Radwaste Building HVAC systems. For the potentially contaminated exhaust, two	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	separate full capacity filter assemblies are installed upstream of the exhaust vent. Each filter assembly is isolated automatically with fan operation. Prefilter and HEPA filter differential pressures are indicated outside each filter assembly compartment. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
TB-A	21D	Radwaste Bldg Third Floor	CNS-FP-261	N05	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). Radwaste Building floor drain sumps H and K receive drainage flow. The sumps are provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are	Ventilation air from areas containing radioactive materials and equipment are monitored by the Radwaste/Augmented Radwaste Building Ventilation Radiation Monitoring System and is discharged to a common exhaust vent for both the Radwaste Building and Augmented Radwaste Building HVAC systems.	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	For the potentially contaminated exhaust, two separate full capacity filter assemblies are installed upstream of the exhaust vent. Each filter assembly is isolated automatically with fan operation. Prefilter and HEPA filter differential pressures are indicated outside each filter assembly compartment. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
TB-A	22A	Augmented Radwaste Bldg Basement	CNS-FP-262	N03	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). The Augmented Radwaste Building floor drain sump AA receives drainage flow. The sump is provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain	Ventilation air from areas containing radioactive materials and equipment are monitored by the Radwaste/Augmented Radwaste Building Ventilation Radiation Monitoring System and is discharged to a common exhaust vent for both the Radwaste Building and	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	Augmented Radwaste Building HVAC systems. For the potentially contaminated exhaust, a filter assembly is provided for each fan. The filter assembly consists of a roughing filter and HEPA filter arrangement rated for full air capacity. A fire deluge system is part of each filter assembly, and fire dampers are installed in the exhaust duct. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
TB-A	22B	Augmented Radwaste Bldg First Floor	CNS-FP-263	N03	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). The Augmented Radwaste Building floor drain sump AA receives drainage flow. The sump is provided with pumps which transfer the collected drainage from	Ventilation air from areas containing radioactive materials and equipment are monitored by the Radwaste/Augmented Radwaste Building Ventilation Radiation Monitoring System and is discharged to a common	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	exhaust vent for both the Radwaste Building and Augmented Radwaste Building HVAC systems. For the potentially contaminated exhaust, a filter assembly is provided for each fan. The filter assembly consists of a roughing filter and HEPA filter arrangement rated for full air capacity. A fire deluge system is part of each filter assembly, and fire dampers are installed in the exhaust duct. In addition, CNS Procedure 9.RW.6 restricts the size of the largest single storage container (i.e., sea-land container) inside the Protected Area to a maximum contact dose rate of 100 mRem/hr. However, for conservative purposes, NEDC 11-148 uses a sea-land container having a maximum contact dose rate of 200 mRem/hr in the Protected Area, which is in accordance with DOT shipping restrictions. As demonstrated in NEDC 11-148, the radioactive dose limits of 10 CFR 20 will not be exceeded due to the effects of complete combustion of the largest single radioactive source container (assumed DOT restricted 200 mRem/hr contact dose rate) and fire suppression activities inside the Protected Area. If	undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.		
TB-A	22C	Augmented Radwaste Bldg Second Floor	CNS-FP-264	N03	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). The Augmented Radwaste Building floor drain sump AA receives drainage flow. The sump is provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid	Ventilation air from areas containing radioactive materials and equipment are monitored by the Radwaste/Augmented Radwaste Building Ventilation Radiation Monitoring System and is discharged to a common exhaust vent for both the Radwaste Building and Augmented Radwaste Building HVAC systems. For the potentially contaminated exhaust, a filter assembly is provided for each fan. The filter assembly consists of a roughing filter and HEPA filter arrangement rated for full air capacity. A fire deluge system is part of each filter assembly, and fire dampers are installed in	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						effluents prior to safe removal.	the exhaust duct. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
YD	23A	Electric Motor Driven Fire Pump Room	CNS-FP-265	N03	Yes	N/A	N/A	N/A	N/A - Screened Out
YD	23B	Diesel Driven Fire Pump Room	CNS-FP-265	N03	Yes	N/A	N/A	N/A	N/A - Screened Out
YD	23C	Diesel Oil TK. Room	CNS-FP-265	N03	Yes	N/A	N/A	N/A	N/A - Screened Out
TB-A	24	Multi-purpose Facility	CNS-FP-266	N06	No	The MPF has a drainage collection system on the west side of the building. Radiation Protection can monitor liquid contamination in this drainage system. Curbs are not present in this facility. Provisions are also in place to commence communication between	Ventilation air from areas containing radioactive materials and equipment are monitored by the Multi Purpose Facility (MPF) Building Ventilation Monitoring System. The system provides representative samples of the particulate and iodine activity present in the effluent stream at a point just prior to release to the	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls		Training and Pre-Fire Plan Review Results	Conclusions
						Water	Smoke		
						the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	atmosphere. If radiation is detected in the exhaust by either surveillance of the system particulate filters or a portable Continuous Air Monitoring unit alarm, the HVAC system can be manually switched to an exhaust route that discharges building air through a HEPA filter located on the mezzanine. Fan Unit EF 4 is located in the UPS Battery Room and controls the exhaust from this room. A separate exhaust fan and HEPA filter unit are installed to control the exhaust from the decontamination room when the isolating door is closed over the room. Therefore, ventilation and exhaust are monitored and filtered in this facility. In addition, CNS Procedure 9.RW.6 restricts the size of the largest single storage container (i.e., sea-land container) inside the Protected Area to a maximum contact dose rate of 100 mRem/hr. However for conservative purposes, NEDC 11-148 uses a sea-land container having a maximum contact dose rate of 200 mRem/hr in the Protected Area, which is in accordance with DOT shipping restrictions. As demonstrated in NEDC 11-148, the radioactive dose limits of 10 CFR 20 will not	these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							be exceeded due to the effects of complete combustion of the largest single radioactive source container (assumed DOT restricted 200 mRem/hr contact dose rate) and fire suppression activities inside the Protected Area. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents. It is noted that Procedure 9.RW.6 identifies potential MPF vault storage of sea-land containers having a maximum contact dose rate of 1000 mRem/hr. However, NEDC 11-148 has dispositioned a fire in the MPF vaults as not being credible, and therefore this contact dose rate is permitted.		
YD	25	Off-Gas Bldg	CNS-FP-267	N04	No	Floor drains are routed to monitored Liquid	A single unit supplies the heating and ventilating	Training materials reinforce use of the pre-fire plans.	As the fire zone has been deemed to be of negligible

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						Radwaste System (LRW). Elevated Release Sump "Z" receives drainage flow from the SGT System and the Off Gas System. The sump is provided with pumps which normally discharges the collected drainage from the sump to the Waste Collector Tank. A 3-way valve is installed in this discharge line to allow flow to the Radwaste System Floor Drain Collector Tank as an alternate storage. The collected wastes are analyzed, filtered, and treated prior to either dilution and safe disposal to the discharge flume or re-use in the station. A radiation survey of this building shows a dose contribution from the below ground piping to be less than 1 mRem/hr and with insignificant contamination. Therefore, the potential contaminated water runoff resulting from fire fighting activities is deemed to be insignificant. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	requirements for the Off Gas Building. This unit consists of an inlet damper, filters, supply fan, and a duct mounted resistance electric heating coil. Outside air is supplied to the General Area and Fan Room and air is recirculated from the General Area. A radiation survey of this building shows a dose contribution from the below ground piping to be less than 1 mRem/hr and with insignificant contamination. Therefore, the potential quantity of contaminated smoke resulting from fire fighting activities is deemed to be insignificant. If normal ventilation is not available, smoke will be removed using standard industry manual ventilation techniques (i.e. portable smoke ejectors and flexible ducts) to the outside or to an area where normal ventilation will remove smoke. Prior to any release, plant pre-fire plans provide instructions to establish communications with Radiation Protection personnel, and to also provide precautions for containing, monitoring, and safely releasing contaminated gaseous effluents.	Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	consequence, the NPPD approach will meet NFPA 805 radioactive release performance criteria.
YD	26	OWC Building	CNS-FP-343	N05	Yes	N/A	N/A	N/A	N/A - Screened Out

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
YD	Transform. Yard	Transform. Yard	CNS-FP-268	N04	Yes	N/A	N/A	N/A	N/A - Screened Out
YD	Hydrogen Storage	Hydrogen Storage	CNS-FP-344	N02	Yes	N/A	N/A	N/A	N/A - Screened Out
DW	DW	Drywell	CNS-FP-TBD CNS-FP-TBD CNS-FP-TBD	N00	No	Floor drains are routed to monitored Liquid Radwaste System (LRW). The Drywell floor drain sump receives drainage flow. The sump is provided with pumps which transfer the collected drainage from the sump to the Radwaste System Floor Drain Collector Tank. The collected wastes are analyzed, filtered, and treated prior to either safe disposal or re-use in the station. During NPO: Floor drains as described above remain present. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents for fires where NPO openings may exist.	The drywell ventilation system is connected, through suitable valves, to the normal Reactor Building Heating and Ventilating system. Upon detection of high radiation levels, receipt of high drywell pressure, or receipt of low reactor water level, the Reactor Building Heating and Ventilating system will isolate for filtration and exhaust purposes. For these conditions, exhaust air is passed through the SGT System, which contains banks of prefilters and HEPA filters before discharging to the atmosphere via the Elevated Release Point (ERP). During Normal Operation: Drywell air is normally recirculated through internal cooling units and filtration equipment. During NPO: The Drywell purge system can exhaust air via the Reactor Building Heating and Ventilating system exhaust plenum.	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	Based on the availability of engineered controls for both smoke and fire suppression water runoff, and use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.
N/A	345 kV	345 kV	CNS-FP-341	N04	Yes	N/A	N/A	N/A	N/A - Screened Out

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
	Switchyard	Switchyard							
N/A	Const. Projects Fab Shop	Const. Projects Fab Shop	CNS-FP-345	N03	Yes	N/A	N/A	N/A	N/A - Screened Out
N/A	Craft Change Building	Craft Change Building	CNS-FP-363	N01	Yes	N/A	N/A	N/A	N/A - Screened Out
N/A	Dining Facility	Dining Facility	CNS-FP-355	N01	Yes	N/A	N/A	N/A	N/A - Screened Out
East WH	East WH	East WH	CNS-FP-347	N07	No	NEDC 11-148 has dispositioned the East Warehouse to be of minimal consequence. As such, the radioactive dose limits of 10 CFR 20 will not be exceeded due to the effects of fire suppression activities. Provisions are also in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	NEDC 11-148 has dispositioned the East Warehouse to be of minimal consequence. As such, the radioactive dose limits of 10 CFR 20 will not be exceeded due to the effects of fire suppression activities. Smoke will be removed using manual ventilation to the outside. Prior to any release, the radiological hazards associated with releasing contaminated smoke from the building will be considered and direct communication between the Fire Brigade and Radiation Protection will be initiated.	Although NEDC 11-148 has dispositioned the area to be of minimal consequence, it is the intent of the Fire Brigade to safely contain radioactive contamination within the area as a result of fire fighting activities (i.e. practicing the "ALARA" concept). As such, training materials reinforce the use of pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for safe removal of contaminated smoke and water runoff in these potentially contaminated areas. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training	As the East Warehouse has been dispositioned to not be of consequence, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
								materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
N/A	Comm. Building	Comm. Building	CNS-FP-334	N07	Yes	N/A	N/A	N/A	N/A - Screened Out
N/A	Fire House / Office	Fire House / Office	CNS-FP-349	N02	Yes	N/A	N/A	N/A	N/A - Screened Out
N/A	Learning Center	Learning Center	CNS-FP-273 CNS-FP-274	N05 N06	Yes	N/A	N/A	N/A	N/A - Screened Out
LLRW	LLRW	LLRW Pad	CNS-FP-340	N04	No	As described in NEDC 11-148, the LLRW is subject to a sampling program in accordance with CNS Chemistry Procedure 8.ENV.7. This sampling program identifies that the surface water runoff collection is conducted in addition to the CNS Radiological Environmental Monitoring Program. Therefore, in accordance with Procedure 8.ENV.7 for a fire event, provisions are in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents. As this area is located outside of the Protected Area, any contaminated water runoff is treated as an abnormal release. Abnormal releases are documented and reported in accordance with regulatory guidelines, and	CNS procedures restrict the size of the largest single storage container (i.e., sea-land container) outside the Protected Area to a maximum contact dose rate of 2 mRem/hr. As demonstrated in NEDC 11-148, the radioactive dose limits of 10 CFR 20 will not be exceeded due to the effects of complete combustion of the largest single radioactive source container and fire suppression activities at this location outside the Protected Area. In accordance with Procedure 8.ENV.7 for a fire event, provisions are in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents.	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria.	Based on the results of NEDC 11-148 regarding airborne release and the use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						compensatory measures are undertaken in accordance with CNS Procedures 8.ENV.1, 3, and 7 for radiological environmental monitoring.		Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
N/A	Maint. Training Facility	Maint. Training Facility	CNS-FP-277 CNS-FP-278	N06 N04	Yes	N/A	N/A	N/A	N/A - Screened Out
RW Mat. Storag e	RW Mat. Storage	Radwaste Mat. Storage	CNS-FP-346	N02	No	For a fire event, provisions are in place to commence communication between the Fire Brigade and Radiation Protection and to contain/monitor liquid effluents prior to safe removal.	NEDC 11-148 has demonstrated that the airborne release is well below the 10 CFR 20 limit of 100 mRem. As such, the radioactive dose limits of 10 CFR 20 will not be exceeded due to the effects of fire suppression activities. Smoke will be removed using manual ventilation to the outside. Prior to any release, the radiological hazards associated with releasing contaminated smoke from the building will be considered and direct communication between the Fire Brigade and Radiation Protection will be initiated.	Training materials reinforce use of the pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for the containment and safe removal of contaminated smoke and water runoff in these potentially contaminated areas. Training material and pre-fire plan revisions will describe the presence and potential use of monitored HVAC and drainage systems, if such systems are deemed operational and capable of supporting manual removal efforts. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to	Based on the results of NEDC 11-148 regarding airborne release and the use of revised pre-fire plans and training materials, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
								address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	
N/A	Security Building	Security Building	CNS-FP-333	N10	Yes	N/A	N/A	N/A	N/A - Screened Out
N/A	Tech Support Bldg	Tech Support Bldg	CNS-FP-275 CNS-FP-276	N06 N05	Yes	N/A	N/A	N/A	N/A - Screened Out
N/A	West WH	West WH	CNS-FP-335 CNS-FP-336 CNS-FP-337 CNS-FP-338 CNS-FP-339 CNS-FP-358	N05 N05 N05 N05 N04 N01	Yes	N/A	N/A	N/A	N/A - Screened Out
N/A	N/A	Sea-Land Container	CNS-FP-TBD	N02	No	Per discussion with CNS personnel, sea-land containers may be present in the following fire zones or areas: 2C, 6, 12D, 13A, Augmented Radwaste Building first floor, Multi-purpose Facility, Transformer Yard, Low-level Radwaste Storage Pad, Radwaste Material Storage Building, East Warehouse, and general Site/Yard Areas. Provisions are in place to commence communication between the Fire Brigade and Radiation Protection and to contain and monitor liquid effluents prior to safe discharge to the environment. For these fire zones or areas having installed automatic engineering controls described in this	CNS Procedure 9.RW.6 restricts the size of the largest single storage container (i.e., sea-land container) outside of the MPF vault, but inside the Protected Area, to a maximum contact dose rate of 100 mRem/hr. However for conservative purposes, NEDC 11-148 uses a sea-land container having a maximum contact dose rate of 200 mRem/hr in the Protected Area, which is in accordance with DOT shipping restrictions. As demonstrated in NEDC 11-148, the radioactive dose limits of 10 CFR 20 will not be exceeded due to the effects of complete combustion of the largest single radioactive source container (assumed DOT restricted 200 mRem/hr contact dose rate) and fire suppression activities inside	Training materials reinforce the use of pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for safe removal of contaminated smoke and water runoff in these potentially contaminated areas. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	Based on the use of revised pre-fire plans and training materials and the presence of CNS procedures which restrict the size of the largest single radioactive storage container, and consequently the largest anticipated radioactive dose, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls: Water	Engineering Controls: Smoke	Training and Pre-Fire Plan Review Results	Conclusions
						attachment (i.e. fixed drainage systems to sumps), such controls may also be used to safely monitor, filter, and discharge liquid effluents if deemed available/operational by the responding Fire Brigade and Radiation Protection personnel.	the Protected Area. The disposition of the MPF vault is discussed in Fire Zone 24. A sea-land container stored outside the Protected Area (i.e. LLRW Pad) is restricted to a maximum contact dose rate of 2 mRem/hr per NEDC 11-148. As demonstrated in this calculation, the radioactive dose limits of 10 CFR 20 will not be exceeded due to the effects of complete combustion of the largest single radioactive source container (2 mRem/hr contact dose rate) and fire suppression activities outside the Protected Area.		
YD	Yard	Site/Yard	CNS-FP-352	N02	No	For sea-land containers located in yard areas, provisions are in place to commence communication between the Fire Brigade and Radiation Protection and to contain and monitor liquid effluents prior to safe discharge to the environment.	CNS Procedure 9.RW.6 restricts the size of the largest single storage container (i.e., sea-land container) outside of the MPF vault, but inside the Protected Area, to a maximum contact dose rate of 100 mRem/hr. However, for conservative purposes, NEDC 11-148 uses a sea-land container having a maximum contact dose rate of 200 mRem/hr in the Protected Area, which is in accordance with DOT shipping restrictions. As demonstrated in NEDC 11-148, the radioactive dose limits of 10 CFR 20 will not be exceeded due to the effects of complete combustion of the largest	Training materials reinforce the use of pre-fire plans. Pre-fire plan revisions will identify potentially contaminated areas, provide instructions for communication with Radiation Protection, and describe precautions to be undertaken for safe removal of contaminated smoke and water runoff in these potentially contaminated areas. The level of detail provided in the revised training materials and pre-fire plans will meet NFPA 805 radioactive release performance criteria. Implementation S-3.20 - Pre-fire plans and training	Based on the use of revised pre-fire plans and training materials and the presence of CNS procedures which restrict the size of the largest single radioactive storage container, and consequently the largest anticipated radioactive dose, the NPPD approach will meet NFPA 805 radioactive release performance criteria.

Table E-1 - NEI 04-02 Radioactive Release Transition Review

Fire Area	Fire Zone	Fire Zone Description	Pre-Fire Plan	Pre-Fire Plan Rev	Screened Out?	Engineering Controls Water	Engineering Controls Smoke	Training and Pre-Fire Plan Review Results	Conclusions
							single radioactive source container (assumed DOT restricted 200 mRem/hr contact dose rate) and fire suppression activities inside the Protected Area.	materials will be revised to address radioactive release requirements of NFPA 805. See Attachment S, Table S-3.	

ATTACHMENT F

Fire-Induced Multiple Spurious Operations Resolution

5 Pages

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 was used as input to the overall assessment of MSOs at CNS:

- Post-fire safe shutdown analysis as documented in the NSCA (CNS Calculation NEDC 11-019)
- BWROG generic list of MSOs
- Miscellaneous operating experience
- Fire PRA component selection and MSO calculations

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 of 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:

An expert panel was assembled and met in July 2008 with a follow-up meeting in November 2008. The analysis was updated in January 2011 without reconvening the expert panel. The results of the expert panel are documented in CNS Calculation NEDC 09-080, Revision 1. This calculation describes:

- Details of the conduct of the expert panel
- List of panel members including experience and area of expertise
- Expert panel training
- List of MSOs evaluated including the source of the MSO
- List of MSOs for inclusion in the Fire PRA and the NSCA
- List of all MSOs reviewed and the rationale for inclusion or exclusion in the Fire PRA or NSCA

Step 3 – Update the Fire PRA model and NSCA to include the MSOs of concern

This includes the:

- Identification of equipment (NUREG/CR-6850 Task 2)
- Identification of cables that, if damaged by fire, could result in the spurious operation (NUREG/CR-6850 Task 3, Task 9)
- Identify routing of the cables identified above, including associating that routing with fire areas, fire zones and/or Fire PRA physical analysis units, as applicable.

Include the equipment/cables of concern in the Nuclear Safety Capability Assessment (NSCA). Including the equipment and cable information in the NSCA does not necessarily imply that the interaction is possible since separation/protection may exist throughout the plant fire areas such that the interaction is not possible).

Note: Instances may exist where conditions associated with MSOs do not require update of the Fire PRA and NSCA analysis. For example, Fire PRA analysis in NUREG/CR-6850 Task 2, Component Selection, may determine that the particular interaction may not lead to core damage, or pre-existing equipment and cable routing information may determine that the particular MSO interaction is not physically possible. In other instances, the update of the PRA may not be warranted if the contribution is negligible. The rationale for exclusion of identified MSOs from the Fire PRA and NSCA should be documented and the configuration control mechanisms should be reviewed to provide reasonable confidence that the exclusion basis will remain valid.

Results of Step 3:

The CNS Fire PRA addresses spurious operations, including multiple spurious operations, identified in the NSCA. These multiple spurious operations include those identified in the expert panel review, CNS Calculation NEDC 09-080, Revision 1.

The results of the Fire PRA include:

- Correlation of NSCA components and PRA basic events
- Correlation of PRA basic events and NSCA components

- A listing of MSOs considered with documentation of their disposition

The MSO combinations of concern were also evaluated as part of the NSCA, CNS Calculation NEDC 11-019. For cases where the pre-transition MSO combinations did not meet the deterministic compliance, the MSO combinations were added to the scope of the RI-PB change evaluations.

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 combinations of concern were evaluated as part of the NSCA, CNS Calculation NEDC 11-019. When the deterministic requirements of NFPA 805 could not be met, the MSO was evaluated with a performance-based approach or a plant modification was recommended.

Some of the MSO combinations required recovery actions. The recovery actions were evaluated for feasibility and reliability as well as for any additional risk presented by the action.

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:

As part of Step 4 of the process depicted above, MSO combinations were reviewed for their impact on deterministic compliance (i.e., fire area by fire area reviews to determine if the same fire could result in the potential MSO combinations). As part of the process, VFDR were created where the deterministic requirements of NFPA 805 Section 4.2.3 were not met. These VFDR were addressed by demonstrating compliance with the performance-based approach of Section 4.2.4 of NFPA 805.

Note that the spurious operations reviewed as part of the process included components that were part of the original CNS 10 CFR 50 Appendix R post-fire safe shutdown analysis, as well as components and interactions that were added following a plant-specific review of functional failures and as the industry issue evolved. No specific distinction is made in the program documentation whether the interaction is related to a single spurious operation or MSO, since the risk-informed approach using the Fire PRA provides an integrated plant response model. In

addition to the process defined above another review was performed to gain risk insights related to fire-induced MSOs.

Risk Insights

Spurious operations, both single and multiple, have an impact on the overall fire risk and are included in the Fire PRA model. Fire-induced spurious operations can lead to initiating events and can also affect mitigation of initiating events. Given the potential significance of fire-induced MSOs, an expert panel was held at CNS to systematically search for and identify potential MSOs. Logic modifications were made in the Fire PRA to incorporate potential fire-induced MSO related failures not already included.

While difficult to quantify the impact of MSOs (since the PRA results contain single spurious as well as multiple spurious events), the contribution of fire-induced MSOs is considered to be conservative in the CNS Fire PRA due to the industry's knowledge of the conditional probability and duration of fire-induced spurious operations. Nonetheless fire-induced MSOs are included in the Fire PRA model, and their associated risk is included in the quantification of fire scenarios, of total calculated plant fire risk, and evaluation of VFDR. The VFDR are identified in Attachment C, Table B-3 and a summary of the Fire PRA results is provided in Attachment W.

ATTACHMENT G

Recovery Actions Transition

16 Pages

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 (RA) 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) (PCS) and the determination which pre-transition OMAs are taken at primary control station(s) (Activities that occur in the Control Room are not considered pre-transition OMAs). Activities that take place at primary control station(s) or in the Control Room are not recovery actions, by definition.
- Step 2: Determine the population of recovery actions that are required to resolve VFDR (to meet 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 the 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 Revision 5 (ML110070485), the following locations are considered taking place at the primary control station(s):

Auxiliary Shutdown Room

The Alternate Shutdown Room located in the Reactor Building Southeast Corner of the 903'-6" Elevation provides for local control of selected components. The system is comprised of circuit isolation devices, local controls, and indications for the components necessary to safely shutdown the plant in the event of a fire. The alternate shutdown capability is made up of three control panels in the Alternate Shutdown Room that provide an alternate location from which to operate selected components of the Automatic Depressurization (ADS), High Pressure Coolant Injection (HPCI), Residual Heat Removal (RHR) and Reactor Equipment Cooling (REC) systems. The components have been provided with isolation devices and local controls.

The plant parameters which are necessary to shutdown the reactor have been included in the ASD capability. The process monitoring instrumentation provided on the ASD panels are provided with the information necessary to monitor plant performance.

Transfer switches are used to provide an alternate source of power for selected components as well as direct process signals to indicators on the ASD panel.

The design of the CNS Alternate Shutdown Capability was evaluated by the NRC and found to be in accordance with applicable requirements as indicated in NRC Safety Evaluation for Appendix R to 10 CFR Part 50, Items III.G.3 and III.L, dated April 16, 1984 and August 21, 1985.

Table G-1 - Recovery Actions and Activities Occurring at the Primary Control Station(s) identify 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.

Step 2 – Determine the population of recovery actions that are required to resolve VFDRs (to meet 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 risk acceptance criteria or maintain a sufficient level of defense-in-depth.

Results of Step 2:

The final set of recovery actions is provided in Table G-1 - Recovery Actions and Activities Occurring at the Primary Control Station(s).

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. The additional risk is provided in Attachment W.

All of the recovery actions were reviewed for adverse impact and dispositioned in Fire Area specific Fire Risk Evaluation calculations. None of the recovery actions were found to have an adverse impact on the Fire PRA.

Step 4: Evaluate the Feasibility of Recovery Actions

Recovery actions were evaluated against the feasibility criteria provided in the 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:

Each of the feasibility criteria in FAQ 07-0030 were assessed for the recovery actions listed in Table G-1. The results of the assessment are included in CNS Calculation NEDC 10-041, "Recovery Action Feasibility Review." The assessment addresses the post-fire operator actions (recovery actions) that are being credited for the CNS NFPA 805 program in a subset of the fire areas that are transitioning as performance-based in accordance with NFPA 805, Section 4.2.4. These recovery actions are credited to reduce plant fire risk or are credited as defense-in-depth measures where the Fire PRA quantified CDF and LERF is within the acceptance criteria of RG

1.174. The defense-in-depth measures have been conservatively maintained to provide plant operations with written guidance where such actions will enhance Echelon #3 of defense-in-depth, to provide some assurance that one success path of safe shutdown capability can be restored in the event that Echelon #1 and Echelon #2 of defense-in-depth are somehow degraded or rendered ineffective.

CNS Procedures 5.4Fire-S/D and 5.4Post-Fire currently provide written guidance to the plant operators with respect to achieving and maintaining post-fire safe shutdown in accordance with the requirements of existing fire protection licensing basis. Update of these procedures for the credited NFPA 805 recovery actions and fire area analysis results will be completed as part of LAR implementation (see Attachment S Implementation Item S-3.3).

The NFPA 805 recovery action feasibility assessment documented in CNS Calculation NEDC 10-041 is based on a documentation review. The results provide a high level of confidence that all of the credited NFPA 805 recovery actions are feasible. The majority of actions are currently credited and have been assessed under the existing Fire Protection Program, which included field validation. Several new recovery actions identified during the transition process have also been assessed. However, a confirmatory demonstration (field validation walk-through) of the feasibility for the credited NFPA 805 recovery actions will be performed and documented as part of LAR implementation (see Attachment S, Implementation Item S-3.6). This will include field verification of transit times (i.e., travel times to/from recovery action manipulated plant equipment) and execution times (i.e., time required to physically perform the action, such as handwheel a valve open, open a breaker, etc.) that are identified in the calculation as well as existing communication and lighting aspects.

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., Human Reliability Analysis).
- The reliability of recovery actions not modeled specifically in the Fire PRA is 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 in the Fire PRA were addressed using Fire PRA methods, as documented in CNS Calculation NEDC 09-083 "Task 7.12 Fire Human Reliability Analysis".

An implementation item is identified to review and update, if needed, the Fire HRA upon completion of procedure updates, modifications and training (see Attachment S, Implementation Item S-3.7).

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
CB-A	EE-CB-4160G-1GE	BRKR F/ TIE TO D GEN NU 2	Remove control power fuses and operate the 1GE breaker as required.	CBA-01	RA
CB-A	EE-CB-4160G-CSP1B	BRKR F/ CSP B	Remove control power fuses and operate the CSP1B breaker as required.	CBA-04	RA
CB-A	EE-CB-4160G-RHRP1D	BRKR F/ RHR P D	Remove control power fuses and operate the RHRP1D breaker as required.	CBA-04	RA
CB-A	EE-CB-4160G-SWP1B	BRKR F/ SWP B	Remove control power fuses and close the SWP1B breaker as required.	CBA-02	RA
CB-A	SW-MOV-37MV	SW P CROSSTIE	Open breaker 7A at MCC-Y. Close 37MV via handwheel.	CBA-02	RA
CB-A	SW-MOV-MO89B	RHR HX B SW OUTLET	Remove control power fuses for position 6C at MCC-Y and operate MO89B using the starter.	CBA-02	RA
CB-A	SW-STNR-B	SW STNR B	Manually open SW-V-194.	CBA-05	RA
CB-A-1	EE-CB-4160C-1CS	BRKR F/ FDR TO 4160V BUS C FROM SU XFMR	Remove control power fuses and open the 1CS breaker as required.	CBA1-02	RA
CB-B	EE-CB-4160D-1DS	BRKR F/ FDR TO 4160V BUS D FROM SU XFMR	Remove control power fuses and open the 1DS breaker as required.	CBB-03	RA
CB-D	CRD-SOV-SO31A	SDV VENT & DR PILOT V SO-31A	Close IA-V-16. Remove pipe plug and open IA-V-26.	CBD-08	RA
CB-D	CRD-SOV-SO31B	SDV VENT & DR PILOT V SO-31B	Close IA-V-16. Remove pipe plug and open IA-V-26.	CBD-08	RA
CB-D	EE-CB-4160C-1CS	BRKR F/ FDR TO 4160V BUS C FROM SU XFMR	Remove control power fuses and open the 1CS breaker.	CBD-06	RA
CB-D	EE-CB-4160D-1DS	BRKR F/ FDR TO 4160V BUS D FROM SU XFMR	Remove control power fuses and open the 1DS breaker.	CBD-06	RA

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
CB-D	EE-CB-4160F-1FS	FDR BRKR TO 4160V BUS F FROM EMERG XFMR	Remove control power fuses and open the 1FS breaker as required.	CBD-03	RA
CB-D	EE-CB-4160G-1GB	BRKR F/ TIE TO 4160V BUS B	Remove control power fuses and operate the 1GB breaker as required.	CBD-03	RA
CB-D	EE-CB-4160G-1GE	BRKR F/ TIE TO D GEN NU 2	Remove control power fuses and operate the 1GE breaker as required.	CBD-03	RA
CB-D	EE-CB-4160G-1GS	4160V BUS G FDR BRKR FROM EMERG XFMR	Remove control power fuses and operate the 1GS breaker as required.	CBD-03	RA
CB-D	EE-CB-4160G-RHRP1D	BRKR F/ RHR P D	Remove control power fuses and operate the RHRP1D breaker as required.	CBD-09	RA
CB-D	EE-CB-4160G-SS1G	BRKR F/ 480V SUB G	Remove control power fuses and operate the SS1G breaker as required.	CBD-03	RA
CB-D	EE-CB-4160G-SWP1B	BRKR F/ SWP B	Remove control power fuses and operate the SWP1B breaker as required.	CBD-01	RA
CB-D	EE-CHG-125-1B	125VDC STA SERV BAT CHGR 1B	Repair cable.	CBD-07	RA
CB-D	EE-CHG-250-1B	250VDC STA SERV BAT CHGR 1B	Repair cable.	CBD-07	RA
CB-D	EE-MCC-R-1A	MCC R XFER SW	At MCC-S, unlock and place breaker 7B, MCC-R EMER FEEDER to "ON". At MCC-R, press red EMERG button at compartment 1A, "MCC-R Fed from MCC-S".	CBD-14	RA
CB-D	HPCI-MOV-MO15	ST SUPPLY INBOARD ISO	Operate MO15 from ASD HPCI Panel as required.	NA	PCS
CB-D	HPCI-SW-GSCP	ASD GLAND SEAL CDSR COND P C/S	Place HPCI-GLAND SEAL CDSR COND P ISOLATION Switch to "ISOL" and align pump operation to the HPCI ASD Panel.	NA	PCS
CB-D	HPCI-SW-ISAOP	ASD AUX P ISO SW	Place HPCI-AUX Oil PUMP ISOLATION Switch to "ISOL" and align pump operation to the HPCI ASD Panel.	NA	PCS

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
CB-D	HPCI-SW-ISHPCI	ASD HPCI CONTR & IND ISO SW	HPCI ISOLATION switches to "ISOL".	NA	PCS
CB-D	HPCI-SW-ISMO14	ASD MO-14 ISO SW	Place HPCI-MO-14 ISOLATION Switch to "ISOL" and align valve operation to the HPCI ASD Panel.	NA	PCS
CB-D	HPCI-SW-ISMO15	ASD MO-15 ISO SW	Place HPCI-MO-15 ISOLATION Switch to "ISOL" and align valve operation to the HPCI ASD Panel.	NA	PCS
CB-D	HPCI-SW-ISMO16	ASD MO-16 ISO SW	Place HPCI-MO-16 ISOLATION Switch to "ISOL" and align valve operation to the HPCI ASD Panel.	NA	PCS
CB-D	HPCI-SW-ISMO17	ASD MO-17 ISO SW	Place HPCI-MO-17 ISOLATION Switch to "ISOL" and align valve operation to the HPCI ASD Panel.	NA	PCS
CB-D	HPCI-SW-ISMO19	ASD MO-19 ISO SW	Place HPCI-MO-19 ISOLATION Switch to "ISOL" and align valve operation to the HPCI ASD Panel.	NA	PCS
CB-D	HPCI-SW-ISMO20	ASD MO-20 ISO SW	Place HPCI-MO-20 ISOLATION Switch to "ISOL" and align valve operation to the HPCI ASD Panel.	NA	PCS
CB-D	HPCI-SW-ISMO21	ASD MO-21 ISO SW	Place HPCI-MO-21 ISOLATION Switch to "ISOL" and align valve operation to the HPCI ASD Panel.	NA	PCS
CB-D	HPCI-SW-ISMO24	ASD MO-24 ISO SW	Place HPCI-MO-24 ISOLATION Switch to "ISOL" and align valve operation to the HPCI ASD Panel.	NA	PCS
CB-D	HPCI-SW-ISMO25	ASD MO-25 ISO SW	Place HPCI-MO-25 ISOLATION Switch to "ISOL" and align valve operation to the HPCI ASD Panel.	NA	PCS
CB-D	HPCI-SW-ISMO58	ASD MO-58 ISO SW	Place HPCI-MO-58 ISOLATION Switch to "ISOL" and align valve operation to the HPCI ASD Panel.	NA	PCS
CB-D	MS-SOV-SPV71A-PASSIVE	PILOT VALVE FOR MS-RV-71ARV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	CBD-04	RA
CB-D	MS-SOV-SPV71B-PASSIVE	PILOT VALVE FOR MS-RV-71BRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	CBD-04	RA

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
CB-D	MS-SOV-SPV71C-PASSIVE	PILOT VALVE FOR MS-RV-71CRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	CBD-04	RA
CB-D	MS-SOV-SPV71D	PILOT V F/ MSRV-71DRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	CBD-04	RA
CB-D	MS-SOV-SPV71E-PASSIVE	PILOT VALVE FOR MS-RV-71ERV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	CBD-04	RA
CB-D	MS-SOV-SPV71F	PILOT V F/ MS-RV-71FRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	CBD-04	RA
CB-D	MS-SOV-SPV71G-PASSIVE	PILOT VALVE FOR MS-RV-71GRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	CBD-04	RA
CB-D	MS-SOV-SPV71H-PASSIVE	PILOT VALVE FOR MS-RV-71HRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	CBD-04	RA
CB-D	MS-SW-71E	ASD TORUS RV 71E C/S	Place MS-SRV-71E ISOLATION Switch to "ISOL" and align valve operation to the ASD - ADS/REC Panel.	NA	PCS
CB-D	MS-SW-71F	ASD TORUS RV 71F C/S	Place MS-SRV-71F ISOLATION Switch to "ISOL" and align valve operation to the ASD - ADS/REC Panel.	NA	PCS
CB-D	MS-SW-71G	ASD TORUS RV 71G C/S	Place MS-SRV-71G ISOLATION Switch to "ISOL" and align valve operation to the ASD - ADS/REC Panel.	NA	PCS
CB-D	PC-AOV-245AV	SUPPRESSION CHAMBER EXH OUTBOARD ISO	Close IA-V-16. Remove pipe plug and open IA-V-26.	CBD-10	RA
CB-D	PC-AOV-246AV	DW EXH OUTBOARD ISO	Close IA-V-16. Remove pipe plug and open IA-V-26. Close PC-V-410.	CBD-10	RA
CB-D	REC-FIS-24-ASD	FC-R-1G COOLING WATER OUTLET	Monitor flow at HPCI Room inside panel TB221.	CBD-01	RA

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
CB-D	REC-MOV-695MV-ASD	CRITICAL LOOP SUPPLY CROSSTIE	Remove control power fuses for position 8B at MCC-R and close 695MV using the starter.	CBD-01	RA
CB-D	REC-MOV-712MV	REC HX A OUTLET	Open breaker 4C at MCC-Y. Close 712MV via handwheel.	CBD-10	RA
CB-D	REC-MOV-713MV	REC HX B OUTLET	Open breaker 4B at MCC-RB. Close 713MV via handwheel.	CBD-10	RA
CB-D	REC-MOV-714MV-ASD	SOUTH CRITICAL LOOP SUPPLY	Remove control power fuses for position 7C at MCC-Y and close 714MV using starter.	CBD-01	RA
CB-D	REC-SW-1D	ASD REC P 1D C/S	Place RHR-P-1D ISOLATION Switch to "ISOL" and align pump operation to the RHR ASD Panel.	NA	PCS
CB-D	RHR-MOV-MO20-ASD	RHR CROSSHEADER SHUTOFF	Remove control power fuses for position 3A at MCC-R and operate MO20 using the starter.	CBD-09	RA
CB-D	RHR-MOV-MO26B-ASD	DRYWELL SPRAY LOOP B OUTBOARD ISOLATION	Remove control power fuses for position 3C at MCC-Y and operate MO26B using the starter.	CBD-09	RA
CB-D	RHR-MOV-MO57-ASD	RHR DISCHARGE TO RADWASTE INBOARD THROTTLE	Remove control power fuses for position 3B at MCC-R and operate MO57 using the starter.	CBD-09	RA
CB-D	RHR-SW-ISMO13D	ASD MO-13D ISO SW	Place RHR-MO13D ISOLATION Switch to "ISOL" and align valve operation to the RHR ASD Panel.	NA	PCS
CB-D	RHR-SW-ISMO15D	ASD MO-15D ISO SW	Place RHR-MO-15D ISOLATION Switch to "ISOL" and align valve operation to the RHR ASD Panel.	NA	PCS
CB-D	RHR-SW-ISMO16B	ASD MO-16B ISO SW	Place RHR-MO-16B ISOLATION Switch to "ISOL" and align valve operation to the RHR ASD Panel.	NA	PCS
CB-D	RHR-SW-ISMO27B	ASD MO-27B ISO SW	Place RHR-MO-27B ISOLATION Switch to "ISOL" and align valve operation to the RHR ASD Panel.	NA	PCS
CB-D	RHR-SW-ISMO34B	ASD MO-34B ISO SW	Place RHR-MO-34B ISOLATION Switch to "ISOL" and align valve operation to the RHR ASD Panel.	NA	PCS

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
CB-D	RHR-SW-ISMO39B	ASD MO-39B ISO SW	Place RHR-MO-39B ISOLATION Switch to "ISOL" and align valve operation to the RHR ASD Panel.	NA	PCS
CB-D	RHR-SW-ISMO65B	ASD MO-65B ISO SW	Place RHR-MO-65B ISOLATION Switch to "ISOL" and align valve operation to the RHR ASD Panel.	NA	PCS
CB-D	RHR-SW-ISMO66B	ASD MO-66B ISO SW	Place RHR-MO-66B ISOLATION Switch to "ISOL" and align valve operation to the RHR ASD Panel.	NA	PCS
CB-D	RW-AOV-AO82	DW FL DR SUMP DISCH	Lift leads at TB1207 to secure power to AO82.	CBD-10	RA
CB-D	RW-AOV-AO94	DW EQUIP DR SUMP DISCH	Lift leads at TB1207 to secure power to AO94.	CBD-10	RA
CB-D	RWCU-MOV-MO15	SUPPLY INBOARD ISO	Remove control power fuses for position 5C at MCC-R and operate MO15 using the starter.	CBD-11	RA
CB-D	SW-AOV-TCV451B-ASD	REC HX B OUTLET	Close IA-V-16. Remove pipe plug and open IA-V-26.	CBD-01	RA
CB-D	SW-MOV-37MV-ASD	SW PUMPS CROSSTIE	Remove control power fuses for position 7A and operate 37MV using the starter.	CBD-01	RA
CB-D	SW-MOV-651MV-ASD	REC HX B SW OUTLET	Remove control power fuses for position 6B at MCC-Y and operate 651MV using the starter.	CBD-01	RA
CB-D	SW-MOV-887MV-ASD	EMERGENCY SUPPLY TO REC SOUTH CRITICAL LOOP	Remove control power fuses for position 4D at MCC-RB and operate 887MV using the starter.	CBD-01	RA
CB-D	SW-MOV-889MV-ASD	EMERG RETURN FROM REC SOUTH CRITICAL LOOP	Remove control power fuses for position 5D at MCC-RB and operate 889MV using the starter.	CBD-01	RA
CB-D	SW-MOV-MO89B-ASD	RHR HX B SW OUTLET	Remove control power fuses for position 6C at MCC-Y and operate MO89B using the starter.	CBD-09	RA
CB-D	SW-STNR-B	SW STNR B	Manually open SW-V-194.	CBD-01	RA
RB-A	SW-AOV-TCV451B	REC HX B OUTLET	Open breaker 5 at the CCP1B Panel.	RBA-02	RA

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
RB-A	SW-MOV-MO89B	RHR HX B SW OUTLET	Remove control power fuses for position 6C at MCC-Y and operate MO89B using the starter.	RBA-03	RA
RB-B	PC-AOV-245AV	SUPPRESSION CHAMBER EXH OUTBOARD ISO	Close IA-V-16. Remove pipe plug and open IA-V-26.	RBB-01	RA
RB-CF	EE-CB-4160C-1CS	BRKR F/ FDR TO 4160V BUS C FROM SU XFMR	Remove control power fuses and open the 1CS breaker.	RBCF-02	RA
RB-CF	PC-MOV-231MV	DW EXH INBOARD ISO	Open breaker 2B at MCC-RA. Close PC-V-510 via handwheel. Close 231MV via handwheel.	RBCF-05	RA
RB-CF	RHR-MOV-MO16B	RHR P B & D MIN FLOW	De-energize RHR train A logic circuit by opening breaker 6 at AA2. Operate switch from MCR as required..	RBCF-06	RA
RB-CF	RW-AOV-AO82	DW FL DR SUMP DISCH	Lift leads at TB1207 to secure power to AO82.	RBCF-05	RA
RB-CF	RW-AOV-AO94	DW EQUIP DR SUMP DISCH	Lift leads at TB1207 to secure power to AO94.	RBCF-05	RA
RB-CF	SW-MOV-MO89B	RHR HX B SW OUTLET	Remove control power fuses for position 6C at MCC-Y and operate MO89B using the starter.	RBCF-10	RA
RB-DI	CRD-SOV-SO31A	SDV VENT & DR PILOT V SO-31A	Close IA-V-16. Remove pipe plug and open IA-V-26.	RBDI-01	RA
RB-DI	CRD-SOV-SO31B	SDV VENT & DR PILOT V SO-31B	Close IA-V-16. Remove pipe plug and open IA-V-26.	RBDI-01	RA
RB-DI	EE-CB-4160C-1CS	BRKR F/ FDR TO 4160V BUS C FROM SU XFMR	Remove control power fuses and open the 1CS breaker.	RBDI-02	RA
RB-DI	EE-CB-4160D-1DS	BRKR F/ FDR TO 4160V BUS D FROM SU XFMR	Remove control power fuses and open the 1DS breaker.	RBDI-02	RA
RB-DI	MS-SOV-SPV71A-PASSIVE	PILOT VALVE FOR MS-RV-71ARV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	RBDI-04	RA

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
RB-DI	MS-SOV-SPV71B-PASSIVE	PILOT VALVE FOR MS-RV-71BRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	RBDI-04	RA
RB-DI	MS-SOV-SPV71C-PASSIVE	PILOT VALVE FOR MS-RV-71CRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	RBDI-04	RA
RB-DI	MS-SOV-SPV71D	PILOT V F/ MSRV-71DRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	RBDI-04	RA
RB-DI	MS-SOV-SPV71E-PASSIVE	PILOT VALVE FOR MS-RV-71ERV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	RBDI-04	RA
RB-DI	MS-SOV-SPV71F	PILOT V F/ MS-RV-71FRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	RBDI-04	RA
RB-DI	MS-SOV-SPV71G-PASSIVE	PILOT VALVE FOR MS-RV-71GRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	RBDI-04	RA
RB-DI	MS-SOV-SPV71H-PASSIVE	PILOT VALVE FOR MS-RV-71HRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	RBDI-04	RA
RB-DI	NBI-LT-52A	REACTOR LEVEL TO FW CONTR	Close valve NBI-V-620.	RBDI-05	RA
RB-DI	NBI-LT-52C	REACTOR LEVEL TO FW CONTR	Close valve NBI-V-620.	RBDI-05	RA
RB-DI	NBI-LT-59A	REACTOR WTR LEVEL WIDE RANGE T	Close valve NBI-V-620.	RBDI-05	RA
RB-DI	NBI-LT-59C	REACTOR WTR LEVEL WIDE RANGE T	Close valve NBI-V-620.	RBDI-05	RA
RB-DI	NBI-LT-91A	REACTOR SHROUD LEVEL T	Close valve NBI-V-620.	RBDI-05	RA
RB-DI	NBI-LT-91C	REACTOR WTR LEVEL FUEL ZONE T	Close valve NBI-V-620.	RBDI-05	RA
RB-DI	NBI-PT-53A	REACTOR PRESS T	Close valve NBI-V-620.	RBDI-05	RA
RB-DI	NBI-PT-53C	REACTOR PRESS T	Close valve NBI-V-620.	RBDI-05	RA

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
RB-E	SW-AOV-TCV451A	REC HX A OUTLET	Open breaker 5 at the CCP1A Panel.	RBE-03	RA
RB-E	SW-MOV-MO89A	RHR HX A SW OUTLET	Remove control power fuses for position 8A at MCC-Q and operate MO89A using the starter.	RBE-03	RA
RB-FN	EE-MCC-R-1A	MCC R XFER SW	At MCC-S, unlock and place breaker 7B, MCC-R EMER FEEDER to "ON". At MCC-R, press red EMERG button at compartment 1A, "MCC-R Fed from MCC-S".	RBFN-06	RA
RB-FN	HPCI-ECCS	DUMMY COMPONENT FOR HPCI LOW RPV LEVEL/HIGH DRYWELL PRESSURE SPURIOUS INITIATION	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-FAN-GSE	HPCI GLAND SEAL EXH	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-FIC-108	P DISCH FLOW CONTR	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-MOV-MO14	ST SUPPLY TO TU	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-MOV-MO15	ST SUPPLY INBOARD ISO	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-MOV-MO16-PASSIVE	STEAM SUPPLY OUTBOARD ISOLATION VALVE	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-MOV-MO17	P SUCT FROM ECST	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-MOV-MO19	HPCI INJECT	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-MOV-MO20	HPCI P DISCH	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-MOV-MO21	HPCI-P-MP TEST BYPASS TO ECST	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-MOV-MO24	HPCI-P-MP TEST BYPASS REDUNDANT SHUTOFF	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
RB-FN	HPCI-MOV-MO25	HPCI-P-MP MIN FLOW BYPASS LINE ISO	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-MOV-MO58	HPCI P SUCT FROM SUPPRESSION POOL	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-P-ALOP	HPCI AUX LO P	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-P-CP	HPCI COND P	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-PI-109	P DISCH PRESS	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-PI-111	TU ST INLET PRESS	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-PI-112	TU ST EXH PRESS	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-PI-116	P SUCT PRESS	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-SI-2792	TU SI	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-FN	HPCI-TU-TURB	HPCI TU & CVL CHEST	Isolate and operate HPCI from HPCI ASD Panel.	RBFN-05	RA
RB-J	EE-CB-4160C-1CS	BRKR F/ FDR TO 4160V BUS C FROM SU XFMR	Remove control power fuses and open the 1CS breaker.	RBJ-01	RA
RB-J	SW-MOV-MO89B	RHR HX B SW OUTLET	Remove control power fuses for position 6C at MCC-Y and operate MO89B using the starter.	RBJ-03	RA
RB-K	EE-CB-4160C-1CS	BRKR F/ FDR TO 4160V BUS C FROM SU XFMR	Remove control power fuses and open the 1CS breaker.	RBK-01	RA
RB-K	EE-CB-4160D-1DS	BRKR F/ FDR TO 4160V BUS D FROM SU XFMR	Remove control power fuses and open the 1DS breaker.	RBK-01	RA
RB-K	EE-CB-4160F-RSWP1A	BRKR F/ RHR SWBP A	Remove control power fuses and operate the RSWP1A breaker as required.	RBK-03	RA

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
RB-K	SW-MOV-MO89A	RHR HX A SW OUTLET	Remove control power fuses for position 8A at MCC-Q and operate MO89A using the starter.	RBK-05	RA
RB-M	EE-CB-4160C-1CS	BRKR F/ FDR TO 4160V BUS C FROM SU XFMR	Remove control power fuses and open the 1CS breaker.	RBM-01	RA
RB-M	EE-CB-4160D-1DS	BRKR F/ FDR TO 4160V BUS D FROM SU XFMR	Remove control power fuses and open the 1DS breaker.	RBM-01	RA
RB-M	EE-CB-4160F-1FS	FDR BRKR TO 4160V BUS F FROM EMERG XFMR	Remove control power fuses and close the 1FS breaker as required.	RBM-02	RA
RB-M	EE-CB-4160F-RHRP1B	BRKR F/ RHR P B	Remove control power fuses and operate the RHRP1B breaker as required.	RBM-02	RA
RB-M	EE-CB-4160F-SS1F	BRKR F/ 480V SUB F	Remove control power fuses and operate the SS1F breaker as required.	RBM-02	RA
RB-M	EE-CB-4160G-1GS	4160V BUS G FDR BRKR FROM EMERG XFMR	Remove control power fuses and operate the 1GS breaker as required.	RBM-02	RA
RB-M	EE-CB-4160G-SS1G	BRKR F/ 480V SUB G	Remove control power fuses and operate the SS1G breaker as required.	RBM-02	RA
RB-M	MS-SOV-SPV71D	PILOT V F/ MSRV-71DRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	RBM-03	RA
RB-M	MS-SOV-SPV71F	PILOT V F/ MS-RV-71FRV	Open breaker 15 at Panel AA2. Open breaker 8 at Panel BB2.	RBM-03	RA
RB-M	PC-MOV-231MV	DW EXH INBOARD ISO	Open breaker 2B at MCC-RA. Close PC-V-510 via handwheel. Close 231MV via handwheel.	RBM-05	RA
RB-M	RCIC-ECCS	DUMMY COMPONENT FOR RCIC LOW RPV LEVEL SPURIOUS INITIATION	Place switch IS/RCIC in the Aux Relay Room to "ISOLATE".	RBM-08	RA
RB-M	RCIC-MOV-MO131	RCIC ST SUPPLY TO RCIC TU	Place IS/RCIC switch in the Aux Relay room to "ISOLATE".	RBM-08	RA

Attachment G - Table G-1 Recovery Actions and Activities Occurring at the Primary Control Station(s)

Fire Area	Component	Component Description	Actions	VFDR	RA/PCS
RB-M	RCIC-MOV-MO18	RCIC SUPPLY FROM COND STORAGE	Place IS/RCIC switch in the Aux Relay room to "ISOLATE".	RBM-08	RA
RB-P	EE-CB-4160C-1CS	BRKR F/ FDR TO 4160V BUS C FROM SU XFMR	Remove control power fuses and open 1CS breaker.	RBP-02	RA
RB-P	EE-CB-4160D-1DS	BRKR F/ FDR TO 4160V BUS D FROM SU XFMR	Remove control power fuses and open 1DS breaker.	RBP-02	RA
RB-V	EE-CB-4160C-1CS	BRKR F/ FDR TO 4160V BUS C FROM SU XFMR	Remove control power fuses and open 1CS breaker.	RBV-01	RA
RB-V	EE-CB-4160D-1DS	BRKR F/ FDR TO 4160V BUS D FROM SU XFMR	Remove control power fuses and open 1DS breaker.	RBV-01	RA
TB-A	EE-CB-4160C-1CS	BRKR F/ FDR TO 4160V BUS C FROM SU XFMR	Remove control power fuses and open 1CS breaker.	TBA-01	RA
TB-A	EE-CB-4160D-1DS	BRKR F/ FDR TO 4160V BUS D FROM SU XFMR	Remove control power fuses and open 1DS breaker.	TBA-01	RA
TB-A	EE-CB-4160DG1-EG1	BRKR F/ D GEN NU 1 OUTPUT	Remove control power fuses and close the EG1 breaker as required.	TBA-02	RA
TB-A	EE-CB-4160F-1FA	BRKR F/ TIE TO 4160V BUS A	Remove control power fuses and open the 1FA breaker as required.	TBA-02	RA
TB-A	EE-CB-4160F-1FS	FDR BRKR TO 4160V BUS F FROM EMERG XFMR	Remove control power fuses and open the 1FS breaker as required.	TBA-02	RA

ATTACHMENT H

NFPA 805 Frequently Asked Question Summary Table

4 Pages

Note: The NFPA 805 FAQ process will continue through the transition of non-pilot NFPA 805 plants. Final closure of the FAQs will occur when RG 1.205, which endorses the new revision of NEI 04-02, is approved by the NRC.

This table includes the approved FAQs that have not been incorporated into the current endorsed revision of NEI 04-02 and utilized in this submittal:

Table H-1 - NEI 04-02 FAQs Utilized in LAR Submittal

No.	Rev.	Title	FAQ Ref.	Closure Memo
06-0008	9	NFPA 805 Fire Protection Engineering Evaluations	ML090560170	ML073380976
06-0022	3	Acceptable Electrical Cable Construction Tests	ML090830220	ML091240278
07-0030	5	Establishing Recovery Actions	ML103090602	ML110070485
07-0032	2	Clarification of 10 CFR 50.48(c), 10 CFR 50.48(a) and GDC 3 clarification	ML081300697	ML081400292
07-0035	2	Bus Duct Counting Guidance for High Energy Arcing Faults	ML091610189	ML091620572
07-0038	3	Lessons learned on Multiple Spurious Operations	ML103090608	ML110140242
07-0039	2	Lessons Learned – NEI 04-02 B-2 Table	ML091420138	ML091320068
07-0040	4	Non-Power Operations Clarifications	ML082070249	ML082200528
08-0042	0	Fire Propagation from Electrical Cabinets	ML080230438	ML092110537
08-0043	1	Electrical Cabinet Fire Location	ML083540152	ML092120448
08-0044	0	Large Oil Fires	ML081200099	ML092110516
08-0046	0	Incipient Fire Detection Systems	ML081200120	ML093220426
08-0047	1	Spurious Operation Probability	ML082770662	ML082950750
08-0048	0	Fire Ignition Frequency	ML081200291	ML092190457
08-0049	0	Cable Fires	ML081200309	ML092100274
08-0050	0	Non Suppression Probability	ML081200318	ML092190555
08-0051	0	Hot Short Duration	ML083400188	ML100900052

Table H-1 - NEI 04-02 FAQs Utilized in LAR Submittal

No.	Rev.	Title	FAQ Ref.	Closure Memo
08-0052	0	Transient Fire Growth Rate and Control Room Non-Suppression	ML081500500	ML092120501
08-0053	0	Kerite-FR Cable Failure Thresholds	ML082660021	ML120060267
08-0054*	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	ML120410589	ML120750108

*Note: The FAQ submittal number was 08-0054 but the NRC closure memorandum cover letter for the FAQ was listed as 07-0054. FAQ 08-0054 was used to be consistent with the closure memorandum.

ATTACHMENT I

Definition of Power Block and Plant

2 Pages

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

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

The power block is the group of buildings composed of:

Table I-1 CNS Power Block Definition

Building/Structure	Fire Area(s)
Reactor Building	RB-A, RB-B, RB-CF, RB-DI, RB-E, RB-FN, RB-J, RB-K, RB-M, RB-N, RB-P, RB-T, RB-V, TB-C, DW
Control Building	CB-A, CB-A-1, CB-B, CB-C, CB-D
Turbine Generator Building	TB-A
Diesel Generator Building	DG-A, DG-B
Water Treatment Building	TB-A
Intake Structure	IS-A
Radwaste Building	TB-A
Augmented Radwaste Building	TB-A
Fire Pump House	YD*
Transformer Yard	YD*
Offgas Building	TB-A
Hydrogen Storage Building	TB-A
Multi-Purpose Facility (MPF)	TB-A
Yard	YD*

*A large area called the Yard (Fire Area YD) has been included in the power block and encompasses all locations inside the owner-controlled area that have equipment required for nuclear plant operations, and that are not contained in any other Nuclear Safety Capability Assessment (NSCA) fire areas. The equipment in YD includes such items as offsite power distribution equipment, portions of the non-safety power distribution system, and the fire pump house.

ATTACHMENT J

Fire Modeling V&V

8 Pages

Table J-1 - V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Flame Height (Method of Heskestad)	Calculates the vertical extension of the flame region of a fire.	• NUREG-1805, Chapter 3, 2004 • NUREG-1824, Volume 3, 2007 • SFPE Handbook, 4th Edition, Chapter 2-1, Heskestad, 2008	• The correlation is used in the NUREG-1805 fire model, for which V&V was documented in NUREG-1824. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability.
Plume Centerline Temperature (Method of Heskestad)	Calculates the vertical separation distance, based on temperature, to a target in order to determine the vertical extent of the ZOI.	• NUREG-1805, Chapter 9, 2004 • NUREG-1824, Volume 3, 2007 • SFPE Handbook, 4th Edition, Chapter 2-1, Heskestad, 2008 • NUREG/CR-6850, Appendix H - Damage Criteria, 2005	• The correlation is used in the NUREG-1805 fire model, for which V&V was documented in NUREG-1824. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability. • NUREG/CR-6850 generic screening damage criteria is used, which is considered conservative.
Radiant Heat Flux (Point Source Method)	Calculates the horizontal separation distance, based on heat flux, to a target in order to determine the horizontal extent of the ZOI.	• NUREG-1805, Chapter 5, 2004 • NUREG-1824, Volume 4, 2007 • SFPE Handbook, 4th edition, Chapter 3-10, Beyler, C., 2008 • NUREG/CR-6850, Appendix H - Damage Criteria, 2005	• The correlation is used in the NUREG-1805 fire model, for which V&V was documented in NUREG-1824. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability. • NUREG/CR-6850 generic screening damage criteria is used, which is considered conservative.

Table J-1 - V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Plume Radius (Method of Heskestad)	Calculates the horizontal radius, based on temperature, of the plume at a given height.	• FIVE-Rev1, Referenced by EPRI Report 1002981, 2002 • SFPE Handbook, 4th Edition, Chapter 2-1, Heskestad, G., 2008 • NUREG/CR-6850, Appendix H - Damage Criteria, 2005	• The correlation is used in the FIVE-Rev1 fire model. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • NUREG/CR-6850 generic screening damage criteria is used, which is considered conservative.
Hot Gas Layer (Method of MQH)	Calculates the hot gas layer temperature for a room with natural ventilation.	• NUREG-1805, Chapter 2, 2004 • NUREG-1824, Volume 3, 2007 • SFPE Handbook, 4th Edition, Chapter 3-6, Walton W. and Thomas, P., 2008	• The correlation is used in the NUREG-1805 fire model, for which V&V was documented in NUREG-1824. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability.
Hot Gas Layer (Method of Beyler)	Calculates the hot gas layer temperature for a closed compartment with no ventilation.	• NUREG-1805, Chapter 2, 2004 • NUREG-1824, Volume 3, 2007 • SFPE Handbook, 4th Edition, Chapter 3-6, Walton W. and Thomas, P., 2008	• The correlation is used in the NUREG-1805 fire model, for which V&V was documented in NUREG-1824. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability.

Table J-1 - V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Hot Gas Layer (Method of Foote, Pagni, and Alvares [FPA])	Calculates the hot gas layer temperature for a room with forced ventilation.	• NUREG-1805, Chapter 2, 2004 • NUREG-1824, Volume 3, 2007 • SFPE Handbook, 4th Edition, Chapter 3-6, Walton W. and Thomas, P., 2008	• The correlation is used in the NUREG-1805 fire model, for which V&V was documented in NUREG-1824. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability.
Hot Gas Layer (Method of Deal and Beyler)	Calculates the hot gas layer temperature for a room with forced ventilation.	• NUREG-1805, Chapter 2, 2004 • NUREG-1824, Volume 3, 2007 • SFPE Handbook, 4th Edition, Chapter 3-6, Walton W. and Thomas, P., 2008	• The correlation is used in the NUREG-1805 fire model, for which V&V was documented in NUREG-1824. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability.
Ceiling Jet Temperature (Method of Alpert)	Calculates the horizontal separation distance, based on temperature at the ceiling of a room, to a target in order to determine the horizontal extent of the ZOI.	• FIVE-Rev1, Referenced by EPRI Report 1002981, 2002 • NUREG-1824, Volume 4, 2007 • SFPE Handbook, 4th Edition, Chapter 2-2, Alpert, R., 2008 • NUREG/CR-6850, Appendix H - Damage Criteria, 2005	• The correlation is used in the FIVE-Rev1 fire model, for which V&V was documented in NUREG-1824. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability. • NUREG/CR-6850 generic screening damage criteria is used, which is considered conservative.

Table J-1 - V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Hot Gas Layer Calculations using CFAST (Version 6)	Calculates the upper and lower layer temperatures for various compartments, the layer height, and smoke obscuration.	• NIST Special Publication 1086, 2008 • CFAST Version 6 • NUREG-1824, Volume 5, 2007	• V&V of the CFAST code is documented in the NIST Special Publication 1086. • The V&V of CFAST specifically for Nuclear Power Plant applications has also been documented in NUREG-1824. • NUREG-1824 concluded that CFAST models the hot gas layer height, temperature and smoke concentration in an appropriate manner. Furthermore, the predictions of HGL height and temperature are deemed to be within the bounds of experimental uncertainty.
Smoke Detection Actuation Correlation (Method of Heskestad and Delichatsios)	Alpert Ceiling Jet used to determine temperature and Heskestad and Delichatsios temperature to smoke density for smoke detection timing estimates.	• NUREG-1805, Chapter 11, 2004 • NUREG-1824, Volume 4, 2007 • SFPE Handbook, 4th Edition, Chapter 4-1, Custer R., Meacham B., and Schifiliti, R., 2008 • SFPE Handbook, 4th Edition, Chapter 2-2, Alpert, R., 2008	• The smoke detection correlation is used in the NUREG-1805 fire model. • Alpert's ceiling jet correlation V&V is documented in NUREG-1824. • The temperature to smoke density correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability.
Heat Detection Actuation Correlation	Alpert Ceiling Jet used to determine temperature for heat detection timing estimates.	• NUREG-1805, Chapter 11, 2004 • NFPA Handbook, 19th Edition, Chapter 3-9, Budnick, E., Evans, D., and Nelson, H., 2003	• The heat detection correlation is used in the NUREG-1805 fire model. • The correlation is documented in an authoritative publication of the NFPA Fire Protection Handbook. • The correlation is used within the limits of its range of applicability.

Table J-1 - V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Sprinkler Activation Correlation	Used to estimate sprinkler actuation timing based on ceiling jet temperature, velocity, and thermal response of sprinkler.	• NUREG-1805, Chapter 10, 2004 • NFPA Handbook, 19th Edition, Chapter 3-9, Budnick, E., Evans, D., and Nelson, H., 2003	• The sprinkler actuation correlation is used in the NUREG-1805 fire model. • The correlation is documented in an authoritative publication of the NFPA Fire Protection Handbook. • The correlation is used within the limits of its range of applicability.
Control Room Abandonment Calculation using FDS	Evaluates the time at which control room abandonment is necessary based on smoke obscuration and average HGL temperature, and smoke detector actuation.	• FDS Version 5 • NIST Special Publication 1018-5, Volume 2: Verification • NIST Special Publication 1018-5, Volume 3: Validation • NUREG-1824, Volume 7, 2007 • NUREG/CR-6850, Appendix H - Damage Criteria, 2005	• V&V of the FDS is documented in the NIST Special Publication 1018-5. • The V&V of FDS specifically for Nuclear Power Plant applications has also been documented in NUREG-1824. • NUREG-1824 concluded that FDS models the radiant heat and gas temperature in an appropriate manner. Furthermore, the predictions radiant heat and temperature are deemed to be within the bounds of experimental uncertainty. • NUREG/CR-6850 generic screening damage criteria is used, which is considered conservative.
Temperature Sensitive Equipment Hot Gas Layer Study	Determine the upper and lower gas layer temperatures for various compartments, and the layer height, for use in assessing damage to temperature sensitive equipment.	• NIST Special Publication 1086, 2008 • CFAST Version 6 • NUREG-1824, Volume 6, 2007 • NUREG/CR-6850, Appendix H - Damage Criteria, 2005	• V&V of the CFAST code is documented in the NIST Special Publication 1086. • The V&V of CFAST specifically for Nuclear Power Plant applications has also been documented in NUREG-1824. • NUREG-1824 concluded that CFAST models the hot gas layer height, temperature and smoke concentration in an appropriate manner. Furthermore, the predictions of HGL height and temperature are deemed to be within the bounds of experimental uncertainty. • NUREG/CR-6850 generic screening damage criteria is used, which is considered conservative.

Table J-1 - V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Temperature Sensitive Equipment Zone of Influence Study	Determine the radiant heat flux ZOI at which temperature sensitive equipment will reach damage thresholds.	• FDS Version 5 • NIST Special Publication 1018-5, Volume 2: Verification • NIST Special Publication 1018-5, Volume 3: Validation • NUREG-1824, Volume 7, 2007 • NUREG/CR-6850, Appendix H - Damage Criteria, 2005	• V&V of the FDS is documented in the NIST Special Publication 1018-5. • The V&V of FDS specifically for Nuclear Power Plant applications has also been documented in NUREG-1824. • NUREG-1824 concluded that FDS models the radiant heat and gas temperature in an appropriate manner. Furthermore, the predictions radiant heat and temperature are deemed to be within the bounds of experimental uncertainty. • NUREG/CR-6850 generic screening damage criteria is used, which is considered conservative.
Plume/Hot Gas Layer Interaction Study	Determine the point at which hot gas layer and plume interact and establish limits for plume temperature application.	• FDS Version 5 • NIST Special Publication 1018-5, Volume 2: Verification • NIST Special Publication 1018-5, Volume 3: Validation • NUREG-1824, Volume 7, 2007 • NUREG/CR-6850, Appendix H - Damage Criteria, 2005	• V&V of the FDS is documented in NIST Special Publication 1018-5. • The V&V of FDS specifically for Nuclear Power Plant applications has also been documented in NUREG-1824. • NUREG-1824 concluded that FDS models the hot gas layer height, temperature and smoke concentration in an appropriate manner. Furthermore, the predictions of HGL height and temperature are deemed to be within the bounds of experimental uncertainty. • NUREG/CR-6850 generic screening damage criteria is used, which is considered conservative.

Table J-1 - V & V Basis for Fire Models / Model Correlations Used

Calculation	Application	V & V Basis	Discussion
Corner and Wall HRR	Determines a heat release rate adjustment factor for fires that are proximate to a wall or corner.	• IMC 0609, Appendix F, 2005 • SFPE Handbook, 4th Edition, Chapter 2-14, Lattimer, 2008	• The correlation is recommended by IMC 0609 for fires near walls and corners. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability.
Correlation for Heat Release Rates of Cables (Method of Lee)	Used to correlate bench-scale data to heat release rates from cable tray fires.	• NUREG/CR-6850, Appendix R, 2005 • SFPE Handbook, 4th Edition, Chapter 3-1, Babrauskas, 2008	• The correlation is recommended by NUREG/CR-6850. • The correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The correlation is used within the limits of its range of applicability.
Correlation for Flame Spread over Horizontal Cable Trays (FLASH-CAT)	Used to predict the growth and spread of a fire within a vertical stack of horizontal cable trays.	• NUREG/CR-7010, Section 9, 2010 • NUREG/CR-6850, Appendix R, 2005	• The correlation is recommended by NUREG/CR-7010 and follows guidance set forth in NUREG/CR-6850. • The FLASH-CAT model is validated in NUREG/CR-7010, Section 9.2.3, through experimentally measured HRRs compared with the predictions of the FLASH-CAT model. • The correlation is used within the limits of its range of applicability.

ATTACHMENT K

Existing Licensing Action Transition

11 Pages

Attachment K - Exisiting Licensing Action Transition

Licensing Action No.:	01
Licensing Action Title:	The Service Water Intake Structure
Licensing Action:	Exemption from 10 CFR 50 Appendix R, Section III.G.2.b.: 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 combustible or fire hazards.
Transitioned:	No

Licensing Action Documentation:

Initial Exemption Request:

In the Exemption Request dated June 28, 1982, NPPD stated: "[The Service Water Intake Structure (903-IS-I)] currently does not comply with the specific provisions of 10 CFR 50 Appendix R, Section III.G.2 in that twenty feet of horizontal separation free of intervening combustibles, automatic detection and automatic suppression does not exist between the service water pumps. Redundant and diverse automatic suppression and automatic detection does exist within the fire area. Nebraska Public Power District proposes to comply with the requirements of 10 CFR 50 Appendix R by extending automatic suppression and detection to all areas separating the pumps provided an exemption is granted from this requirement for twenty feet between the pumps."

Supplemental Exemption Request:

None applicable.

Exemption SER:

This Exemption Request was granted in an NRC letter dated September 21, 1983.

Discussion:

This Exemption is no longer required for transition because an RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

Attachment K - Exisiting Licensing Action Transition

Licensing Action No.: 02

Licensing Action Title: Cable Spreading Room (Southeast Corner)

Licensing Action: Exemption from 10 CFR 50 Appendix R, Section III.G.2.c.: Enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating.

Transitioned: No

Licensing Action Documentation:

Initial Exemption Request:

In the Exemption Request dated June 28, 1982, NPPD stated: "[The Cable Spreading Room (918-C-II)] currently does not comply with the specific provisions of 10CFR50 Appendix R, Section III.G.2 in that less than twenty feet of horizontal separation exists between safe shutdown circuits. Redundant and diverse means of automatic detection, an automatic suppression system and excellent manual fire fighting capabilities exist in the Cable Spreading Room. Nebraska Public Power district proposes to comply with 10CFR50 Appendix R through an exemption from the twenty foot separation or one-hour fire barrier requirement on the basis of the multiple layers of existing protection, provided by automatic detection and suppression, the existing fire retardants, and the low potential for a significant fire."

Supplemental Exemption Request:

Supplemental information was presented to the NRC in letters dated March 18, 1983, and June 2, 1983.

In the March 18, 1983, letter, the Exemption Request was limited to the southeast corner of the Cable Spreading Room, based on the control circuits needed for the alternative shutdown capability. The submittal also described various design enhancements being implemented.

The June 2, 1983, letter provided more details of proposed design changes, and included photos and a drawing of the area.

Exemption SER:

This Exemption Request was granted in an NRC letter dated September 21, 1983.

Discussion:

This Exemption is no longer required for transition because an RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

Attachment K - Exisiting Licensing Action Transition

Licensing Action No.:	03
Licensing Action Title:	Cable Expansion Room
Licensing Action:	Exemption from 10 CFR 50 Appendix R, Section III.G.2.c.: Enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating.
Transitioned:	No

Licensing Action Documentation:

Initial Exemption Request:

In the Exemption Request dated June 28, 1982, NPPD stated: "[The Cable Expansion Room (918-C-CER)] does not comply with the specific provisions of 10 CFR 50 Appendix R, Section III.G.2 in that less than twenty feet of horizontal separation exists between safe shutdown circuits. Redundant and diverse means of automatic detection, an automatic suppression system and excellent manual firefighting capabilities exist in the Cable Expansion Room. Nebraska Public Power District seeks to comply with 10 CFR 50 Appendix R through an exemption from the twenty-foot separation or one-hour fire barrier requirement on the basis of the multiple layers of existing protection provided by automatic detection and suppression, the existing fire retardants, and the low potential for a significant fire."

Supplemental Exemption Request:

Supplemental information was presented to the NRC in letters dated March 18, 1983, and June 2, 1983.

In the March 18, 1983, letter, NPPD provided additional details regarding the proposed extension of the suppression system, and the installation of a plume impingement shield.

The June 2, 1983, letter provided additional details of proposed enhancements and photos of the area under discussion, along with a summary of the Exemption Request.

Exemption SER:

This Exemption Request was granted in an NRC letter dated September 21, 1983.

Discussion:

This Exemption is no longer required for transition because an RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

Attachment K - Exisiting Licensing Action Transition

Licensing Action No.:	04
Licensing Action Title:	Reactor Building, Northeast Corner
Licensing Action:	Exemption from 10 CFR 50 Appendix R, Section III.G.2.c.: Enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area.
Transitioned:	No

Licensing Action Documentation:

Initial Exemption Request:

In the Exemption Request dated June 28, 1982, NPPD stated: "[The Reactor Building, Northeast Corner (903-R-VII)] currently does not comply with the specific provisions of 10 CFR 50 Appendix R, Section III.G.2, in that a one-hour fire barrier does not separate redundant divisions already protected by automatic suppression and detection. Fire retardants (i.e., conduits, tray covers, asbestos boards, and fire resistant cable) automatic detection, and automatic suppression are used in this area. On the basis of the existing protection provided by these retardants, automatic suppression, automatic detection, and separation from the likely sources of fire, Nebraska Public Power District requests an exemption from the requirement of a one hour fire barrier between redundant divisions in the northeast and northwest corners. The District proposes additional fire retardants in the northwest corner beyond the range of effective water suppression contingent upon the granting of an exemption from automatic suppression."

Supplemental Exemption Request:

Supplemental information was presented to the NRC in letters dated March 18, 1983, and June 2, 1983.

In the March 18, 1983, letter NPPD provided additional details regarding the area of interest, including the specific power feeds within the Division 2 conduit bank.

The June 2, 1983, letter provided photos of the area under discussion, along with a summary of the Exemption Request.

Exemption SER:

This Exemption Request was granted in an NRC letter dated September 21, 1983.

Discussion:

This Exemption is no longer required for transition because an RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

Attachment K - Exisiting Licensing Action Transition

Licensing Action No.:	05
Licensing Action Title:	Control Building Basement
Licensing Action:	Exemption from 10 CFR 50 Appendix R, Section III.G.2.c.: Enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating. In addition, fire detectors and automatic suppression system shall be installed in the area.
Transitioned:	No

Licensing Action Documentation:

Initial Exemption Request:

In the Exemption Request dated June 28, 1982, NPPD stated: "[The Control Building Basement (882-C-II)] currently does not comply with the specific provisions of 10 CFR 50 Appendix R, Section III.G.2 in that less than twenty feet of horizontal separation free of intervening combustibles and automatic suppression does not exist between safe shutdown cables. Automatic detection, manual water suppression, and portable [carbon dioxide] extinguisher are located within the area. In addition, manual foam suppression would also be available. On the basis of the existing protection afforded by fire propagation retardants in the form of conduits and cable essentially qualified to IEEE-STD-383 which inhibit fire propagation and delay the onset of damage, the low combustible loading, adequate cable separation off the floor and below the ceiling, an exemption is requested from the requirement for 20 feet of horizontal separation and an automatic suppression system."

Supplemental Exemption Request:

Supplemental information was presented to the NRC in letters dated March 18, 1983, and June 2, 1983.

In the March 18, 1983, letter NPPD provided a revised approach for this area. NPPD committed to enclose a conduit traversing the ceiling of the area in a one-hour fire barrier. NPPD also indicated that this approach may change to rerouting the conduit or replacing the existing conduit and cable with copper jacketed mineral insulated cable.

The June 2, 1983, letter provided photos of the area under discussion along with a summary of the Exemption Request, and commited to protecting a single division of 125 VDC power with a one-hour fire barrier.

Exemption SER:

This Exemption Request was granted in an NRC letter dated September 21, 1983.

Discussion:

This Exemption is no longer required for transition because an RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

Attachment K - Exisiting Licensing Action Transition

Licensing Action No.: 06

Licensing Action Title: Auxiliary Relay Room

Licensing Action: Exemption from 10 CFR 50 Appendix R, Section III.G.3: Fire detection and fixed fire suppression system shall be installed in the area, room, or zone under consideration.

Transitioned: No

Licensing Action Documentation:

Initial Exemption Request:

In the Exemption Request dated June 28, 1982, NPPD stated: "[The Auxiliary Relay Room (903-C-IV)] currently does not comply with the specific provisions of 10 CFR 50, Appendix R, Section III.G.2, in that twenty feet of horizontal separation and automatic suppression does not exist in the area. Redundant and diverse automatic detection, fire retardants (i.e., panels, conduits, and fire resistant cable) and excellent manual fire fighting capability does exist. On the basis of the existing protection provided by these retardants and manual suppression, Nebraska Public Power District proposes to comply with the provisions of 10 CFR 50 Appendix R by requesting an exemption from the requirements for automatic suppression and barrier separation between redundant divisions in the room. In addition, an exemption from the requirements for automatic suppression is requested for the area MCC LX and MCC TX on the basis of horizontal separation in excess of twenty feet free of interviewing combustibles, the use of fire retardants (i.e., panels, conduits, and fire resistant cable), the presence of automatic detection, and the remote potential for a significant fire."

Supplemental Exemption Request:

Supplemental information was presented to the NRC in letters dated March 18, 1983, and June 2, 1983.

In the March 18, 1983, letter NPPD reiterated the commitment to provide alternate shutdown independent of the area.

The June 2, 1983, letter noted design features of the room, and repeated the commitment for alternate shutdown.

Exemption SER:

This Exemption Request was granted in an NRC letter dated September 21, 1983.

Discussion:

This Exemption is no longer required for transition because an RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

Attachment K - Exisiting Licensing Action Transition

Licensing Action No.:	07
Licensing Action Title:	Control Room
Licensing Action:	Exemption from 10 CFR 50 Appendix R, Section III.G.3: Fire detection and fixed fire suppression system shall be installed in the area, room, or zone under consideration.
Transitioned:	No

Licensing Action Documentation:

Initial Exemption Request:

In the Exemption Request dated June 28, 1982, NPPD stated: "[The Control Room (932-C-V)] currently does not comply with the specific provisions of 10 CFR 50 Appendix R, Section III.G.2 in that one-hour rated fire barrier and automatic suppression does not exist between redundant divisions. Automatic detection does exist in the continuously manned Control Room. In addition, fire retardants exist in the form of fire resistant cable and closed panels. Nebraska Public Power District proposes to comply with the requirements of Appendix R and seeks an exemption from the provisions for redundant divisional separation by a one hour fire barrier and automatic suppression."

Supplemental Exemption Request:

Supplemental information was presented to the NRC in letters dated March 18, 1983, and June 2, 1983.

In the March 18, 1983, letter NPPD reiterated the commitment to provide alternate shutdown independent of the area.

The June 2, 1983, letter repeated the words of the March 18, 1983, submittal.

Exemption SER:

This Exemption Request was granted in an NRC letter dated September 21, 1983.

Discussion:

This Exemption is no longer required for transition because an RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

Attachment K - Exisiting Licensing Action Transition

Licensing Action No.: 08

Licensing Action Title: Fire Area Boundaries - Reactor Building 931' Elevation

Licensing Action: Exemption from 10 CFR 50 Appendix R, Section III.G.2.b.: 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 combustible or fire hazards. In addition, fire detectors and an automatic suppression system shall be installed in the area.

Transitioned: No

Licensing Action Documentation:

Initial Exemption Request:

In the Exemption Request dated June 28, 1982, NPPD stated: "Fire barriers defining fire zones at Cooper Nuclear Station currently do not comply with the specific provisions of Appendix R in that all penetrations are not necessarily equal to or greater than 3 hours in rating. The fuel loading in any zone at Cooper Nuclear Station is generally less than one hour in equivalent fire severity with each zone complemented by detection and/or suppression systems. Nebraska Public Power District requests an exemption from the requirement for continuous 3 hour fire barriers for fire doors, cable penetrations, pipe chases, HVAC ducts, and open areas where little potential fire hazard exists."

Supplemental Exemption Request:

Supplemental information was presented to the NRC in letter dated March 18, 1983, following denial of the original Exemption Request. It was noted that in the June 28, 1982, submittal, this area was generally looked at in light of the requirements of Appendix R, Section III.G.2, (i.e., either redundant equipment being in separate fire areas or zones and separated by an equivalent 3-hour fire barrier or equivalent spatial separation). As such it led to the request for exemption from a total 3-hour-rated barrier as required in Section III.G.2.a. A closer look at the area and the safe shutdown equipment revealed that a more appropriate Exemption Request was from the requirements Section III.G.2.b.

In the March 18, 1983, letter, NPPD provided a detailed description of the Reactor Building 931' Elevation. Specifically, that the area did not comply with the specific provisions of 10 CFR 50 Appendix R, Section III.G.2, with regard to having automatic suppression in the vicinity of redundant Reactor Vessel Level and Pressure Instrument Racks A and B. However, the racks are separated by a minimum of 80 feet with a low combustible loading between them, and the area is provided with an automatic detection system. Therefore, an Exemption was requested from the requirement for an automatic suppression system.

Exemption SER:

This Exemption Request was granted in an NRC letter dated September 21, 1983.

Discussion:

This Exemption is no longer required for transition because an RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

Attachment K - Exisiting Licensing Action Transition

Licensing Action No.: 09

Licensing Action Title: Fire Area Boundaries - Reactor Building 903' Elevation (Excluding Northeast Corner)

Licensing Action: Exemption from 10 CFR 50 Appendix R, Section III.G.2.b.: 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 combustible or fire hazards. In addition, fire detectors and an automatic suppression system shall be installed in the area.

Transitioned: No

Licensing Action Documentation:

Initial Exemption Request:

In the Exemption Request dated June 28, 1982, NPPD stated: "Fire barriers defining fire zones at Cooper Nuclear Station currently do not comply with the specific provisions of Appendix R in that all penetrations are not necessarily equal to or greater than 3 hours in rating. The fuel loading in any zone at Cooper Nuclear Station is generally less than one hour in equivalent fire severity with each zone complemented by detection and/or suppression systems. Nebraska Public Power District requests an exemption from the requirement for continuous 3 hour fire barriers for fire doors, cable penetrations, pipe chases, HVAC ducts, and open areas where little potential fire hazard exists."

Supplemental Exemption Request:

Supplemental information was presented to the NRC in letter dated March 18, 1983, following denial of the original Exemption Request. It was noted that in the June 28, 1982, submittal, this area was generally looked at in light of the requirements of Appendix R, Section III.G.2, (i.e., either redundant equipment being in separate fire areas or zones and separated by an equivalent 3-hour fire barrier or equivalent spatial separation). As such it led to the request for exemption from a total 3-hour-rated barrier as required in Section III.G.2.a. A closer look at the area and the safe shutdown equipment revealed that a more appropriate Exemption Request was from the requirements Section III.G.2.b.

In the March 18, 1983, letter, NPPD provided a detailed description of the Reactor Building 903' Elevation (excluding the northeast corner). Specifically, that it did not comply with specific provisions of 10 CFR 50 Appendix R, Section III.G.2, with regard to having automatic suppression in the vicinity of the redundant Division I and II safe shutdown equipment and cables in conduit. However, the cables and equipment are separated by a minimum of 75 feet without continuity of combustibles, and the area contains automatic detection. Therefore, an Exemption was requested from the requirement for an automatic suppression system.

Exemption SER:

This Exemption Request was granted in an NRC letter dated September 21, 1983.

Discussion:

This Exemption is no longer required for transition because an RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

Attachment K - Exisiting Licensing Action Transition

Licensing Action No.:	10
Licensing Action Title:	Fire Area Boundaries -Reactor Building 859' and 881' Elevations - Quadrants and Torus Area
Licensing Action:	Exemption from 10 CFR 50 Appendix R, Section III.G.2.b.: 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 combustible or fire hazards. In addition, fire detection and an automatic fire suppression system shall be installed in the area.
Transitioned:	No

Licensing Action Documentation:

Initial Exemption Request:

In the Exemption Request dated June 28, 1982, NPPD stated: "Fire barriers defining fire zones at Cooper Nuclear Station currently do not comply with the specific provisions of Appendix R in that all penetrations are not necessarily equal to or greater than 3 hours in rating. The fuel loading in any zone at Cooper Nuclear Station is generally less than one hour in equivalent fire severity with each zone complemented by detection and/or suppression systems. Nebraska Public Power District requests an exemption from the requirement for continuous 3 hour fire barriers for fire doors, cable penetrations, pipe chases, HVAC ducts, and open areas where little potential fire hazard exists."

Supplemental Exemption Request:

Supplemental information was presented to the NRC in letter dated March 18, 1983, following denial of the original Exemption Request. It was noted that in the June 28, 1982, submittal, this area was generally looked at in light of the requirements of Appendix R, Section III.G.2, (i.e., either redundant equipment being in separate fire areas or zones and separated by an equivalent 3-hour fire barrier or equivalent spatial separation). As such it led to the request for exemption from a total 3-hour-rated barrier as required in Section III.G.2.a. A closer look at the area and the safe shutdown equipment revealed that a more appropriate exemption request was from the requirements Section III.G.2.b.

In the March 18, 1983, letter, NPPD provided a detailed description of the Reactor Building 859' and 881' Elevations. Specifically, that these elevations, quadrants and torus area, do not comply with specific provisions of 10 CFR 50 Appendix R, Section III.G.2, with regard to having automatic suppression in the vicinity of the redundant Division I and II safe shutdown equipment and cables in conduit. However, the cables and equipment are separated by a minimum of 75 feet without continuity of combustibles, and the area contains automatic detection. Therefore, an Exemption was requested from the requirement for an automatic suppression system.

Exemption SER:

This Exemption Request was granted in an NRC letter dated September 21, 1983.

Discussion:

This Exemption is no longer required for transition because an RI-PB analysis (NFPA 805, 4.2.4.2 - Fire Risk Evaluation) was performed, and evaluated the condition as acceptable.

ATTACHMENT L

NFPA 805 Chapter 3 Requirements for Approval (10 CFR 50.48(c)(2)(vii))

25 Pages

Approval Request 1

NFPA 805 (2001), Section 3.3.1.2(1) states:

> *Wood used within the power block shall be listed pressure-impregnated or coated with a listed fire-retardant application.*
>
> *Exception: Cribbing timbers 6 in. by 6 in. (15.2 cm by 15.2 cm) or larger shall not be required to be fire-retardant treated.*

CNS Procedure 0.7.1, "Control of Combustibles" requires that all wood in the Power Block be treated with fire retardant material. However, this requirement is intended to apply to raw materials such as plywood and lumber products, and does not apply to commercially available products which utilize small quantities of wood as an integral part of a finished product (e.g., tools, janitorial supplies, special fixtures, M&TE, and office type furniture).

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 possible to procure commercially available products that only have treated wood as an integral part of the finished product.
- Commercially available products utilizing small quantities of non-treated wood, as described above are used throughout the site. It would be impractical to ban such products altogether from the Power Block.

This is similar to the request made by Fort Calhoun Station (ADAMS Accession Number ML11276A118).

Acceptance Criteria Evaluation

Nuclear Safety and Radiological Release Performance Criteria:

The presence of small quantities of non-treated wood used in commercially finished products such as hand tools, which can contain wooden components such as handles, does not affect nuclear safety. The small quantities of wood do not contribute significantly to the combustible loading of a given area, and therefore, a small fire size is anticipated. Moreover, fire detection and fire suppression systems are installed throughout the station and provide alarm notification in the Control Room. Control Room notification and CNS procedures ensure rapid Fire Brigade response. Accordingly, there is no impact on the NSPC.

The presence of small quantities of non-treated wood used in commercially finished products has no impact on the radiological release performance criteria. The radiological release review was performed based on the fire suppression activities in areas containing or potentially containing radioactive materials, and is not impacted by a small quantity of non-treated wood. These non-treated wood tools do not change the radiological release evaluation, which concluded that potentially contaminated water is contained and smoke is monitored. This

minimal non-treated wood does not add additional radiological materials to the area or challenge system boundaries.

Safety Margin and Defense-in-Depth:

The presence of small quantities of non-treated wood used in commercially finished products does not produce a fire hazard and will not adversely impact the ability of the station to achieve and maintain fire safe shutdown. Therefore, the safety margin inherent in the analysis for a fire event has been preserved.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) Preventing fires from starting (e.g., combustible/hot work controls),
2) Detecting fires quickly and extinguishing those that occur, thereby limiting damage (e.g., fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and
3) 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 (e.g., fire barriers, fire rated cable, success path remains free of fire damage, recovery actions).

The presence of small quantities of non-treated wood used in commercially finished products does not impact fire protection defense-in-depth. Their presence does not compromise administrative fire prevention controls, automatic fire detection and suppression functions, manual fire suppression functions, or post-fire safe shutdown capabilities.

Conclusion:

NPPD 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).

Approval Request 2

NFPA 805 (2001), Section 3.3.3, "Interior Finishes," states:

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.

CNS utilizes paints and coatings which do not have the necessary documentation to demonstrate testing under ASTM E-136, or an equivalent test method, and therefore, do not meet the exact requirements of NFPA 805 Section 3.3.3.

NFPA 101 requirements for interior floor finishes state that the floor finish shall be characterized by a critical radiant flux not less than 0.45 W/cm^2. In addition, the NRC issued Information Notice (IN) 2007-26 to address the combustibility of epoxy floor coatings at commercial nuclear power plants. Per IN 2007-26, the NRC defined a noncombustible material as:

- A material which in the form in which it is used, and under the conditions anticipated, will not ignite, burn, support combustion, or release flammable vapors when subjected to fire or heat; and
- Material having a structural base of noncombustible material, as defined above, with a surfacing not over 1/8-inch thick that has a flame spread rating not higher than 50 when measured using the test protocol of American Society for Testing and Materials (ASTM) E 84, Standard Test Method for Surface Burning Characteristics of Building Materials.

NFPA 805 has re-defined the IN 2007-26 definition of non-combustible material to limited combustible material:

Material that, in the form in which it is used, has a potential heat value not exceeding 3500 Btu/lb (8141 kJ/kg) and either has a structural base of noncombustible material with a surfacing not exceeding a thickness of 1/8 in. (3.2 mm) that has a flame spread rating not greater than 50, or has another material having neither a flame spread rating greater than 25 nor evidence of continued progressive combustion, even on surfaces exposed by cutting through the material on any plane.

NFPA 805 defines non-combustible material as:

Material that, in the form in which it is used and under the conditions anticipated, will not ignite, burn, support combustion, or release flammable vapors when subjected to fire or heat.

Therefore, several paints and floor coatings utilized at CNS are considered to be a deviation from this NFPA 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 these paints and coatings throughout the plant. As discussed in the Acceptance Criteria Evaluation below, this deviation presents no adverse impact to the NSPC and radiological release performance criteria, or to safety margin and defense-in-depth.

This is similar to the request made by Arkansas Nuclear One, Unit 2.

Acceptance Criteria Evaluation

Engineering Evaluation 12-009 evaluated the acceptability of these unqualified paints and coatings in response to a Fire Protection Quality Assurance audit, which determined that the combustible loading calculation did not include the potential for paints and coatings within the plant as being combustible. NPPD reviewed all paints and coatings listed in Procedure 7.0.15, "Station Painting Procedure," in order to determine whether or not they were acceptable for use. Several paints and coatings were determined to not be tested under ASTM E-136 or an equivalent test method. NPPD evaluated the contribution that these paints and coatings may contribute to combustible loading in the plant. However, the contribution was determined not to present a challenge to the plant's fire barriers, and is considered to be negligible overall.

NPPD determined that the maximum thickness of these paints and coatings was 40 mils. Materials less than 125 mils thick have a flame spread rating less than 50 under ASTM E-84 testing. The heat content for the floor coatings is assumed to be 130,000 Btu/gallon, which is documented under the NFPA Handbook. The density of coatings is assumed to be 75 lb/ft^3 based off of field samples from Duane Arnold Energy Center. Using manufacturer recommended application rates of 50 square feet per gallon, or 30 mils thickness, the estimated fire load is less than 2 minutes per coat. The most coats placed on any floor are four, resulting in a fire load of less than 8 minutes. Therefore, if the applied coatings were to be considered combustible, the overall fire loading produced would be insignificant.

These paints and coatings are located on concrete walls, ceilings, or floors. These thin interior finishes are considered to present an insignificant fire hazard due to the ability of the concrete to absorb heat during the early stages of fire and mitigate fire growth of the coating.

Nuclear Safety and Radiological Release Performance Criteria:

The use of these paints and coatings do not affect nuclear safety, as in general, they meet the definition of a limited combustible material with isolated thickness excesses. The paints and coating materials were evaluated to have a negligible effect on combustibility. Application of the paints and coatings is controlled via a CNS procedure, and these unqualified coatings are no longer allowed to be used at CNS. Therefore, there is no impact on the NSPC.

The use of these paints and coatings has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing, or potentially containing, radioactive materials and is not dependent on the paints and coating materials. The coatings do not change the radiological release evaluation, which concluded that potentially contaminated water is contained and smoke monitored. Coatings do not add additional radiological materials to the area or challenge system boundaries.

Safety Margin and Defense-in-Depth:

The use of unqualified paints and coatings do not affect safety margin, as in general, they meet the definition of a limited combustible material with isolated thickness excesses. The paints and coating materials were evaluated to have a negligible effect on combustibility. Application of paints and coatings is controlled via a CNS procedure, and these unqualified coatings are no longer allowed to be used at CNS. These precautions and limitations on the use of these

materials have been defined by the limitations of the analytical methods used in the development of the Fire PRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) Preventing fires from starting (e.g., combustible/hot work controls),
2) Detecting fires quickly and extinguishing those that occur, thereby limiting damage (e.g., fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and
3) 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 (e.g., fire barriers, fire rated cable, success path remains free of fire damage, recovery actions).

The use of these paints and coatings does not impact fire protection defense-in-depth. It does not compromise administrative fire prevention controls, automatic fire detection and suppression functions, manual fire suppression functions, or post-fire safe shutdown capabilities.

Conclusion:

NPPD 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).

Approval Request 3

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, and most of the wiring and cabling that is installed above the suspended or dropped ceiling is in conduit and/or is IEEE-383 qualified. However, some of the wiring installed above the suspended ceilings in the Power Block does not comply with the requirements of this code section. The wiring in these locations that is not approved for plenum use and not installed in conduit includes lighting/power receptacle circuits, Gai-tronics cables, fire detection circuits, and/or communications cables associated with computers, telephones, televisions, or projectors.

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 rewire the non-conforming wiring installations that are above the suspended ceilings in the Power Block. 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.

This is similar to the request made by Fort Calhoun Station (ADAMS Accession Number ML11276A118), and Arkansas Nuclear One, Unit 2.

Acceptance Criteria Evaluation

This deviation is technically acceptable based on the following:

- Only a minimal amount of the wiring installed above the suspended ceilings in the Power Block is not rated for plenum use or routed in conduit.

- Based on visual inspection of the areas above the suspended or dropped ceiling in the Power Block, there are no ignition sources in the areas above the suspended ceilings. Cabling is routed in conduit and/or is IEEE-383 qualified.

- EDP-06, Attachment 1, Section F1 includes the requirements for electrical wiring above suspended ceilings and specifically references NFPA 805, Section 3.3.5.1 as part of the fire protection design considerations.

Nuclear Safety and Radiological Release Performance Criteria:

The presence of non-rated and non-enclosed wiring above the suspended ceilings in the Power Block does not affect nuclear safety. The amount of wiring that is not rated for plenum use and

is not located in conduit is limited and the presence of ignition sources located above the ceiling is minimal. Therefore there is no impact on the nuclear safety performance criteria.

The location of non-rated and non-enclosed wiring 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 the type of wiring or locations of suspended ceilings.

Safety Margin and Defense-in-Depth:

The amount of non-rated and non-enclosed wiring above the ceilings in the Power Block is minor and does not present a significant fire hazard. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) Preventing fires from starting (e.g., combustible/hot work controls),
2) Detecting fires quickly and extinguishing those that occur, thereby limiting damage (e.g., fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and
3) 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 (e.g., fire barriers, fire rated cable, success path remains free of fire damage, recovery actions).

The introduction of the non-listed wiring routed above the suspended ceilings does not impact fire protection defense-in-depth. The wiring located above the ceilings in the Power Block does not compromise administrative fire prevention controls, and does not directly result in challenging automatic fire detection and suppression functions, manual fire suppression functions, or post-fire safe shutdown capability.

Conclusion:

NPPD 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).

Approval Request 4

NFPA 805 (2001), Section 3.3.5.2 states:

> *Only metal tray and metal conduits shall be used for electrical raceways. Thin wall metallic tubing shall not be used for power, instrumentation, or control cables. Flexible metallic conduits shall only be used in short lengths to connect components.*

CNS Procedure 7.3.55, Raceway Installation, permits plastic conduit when installed embedded in concrete or below grade.

This is considered to be a deviation from this NFPA element, for which NRC approval is requested.

Basis for Request

The basis for the approval request of this deviation is:

- In certain Fire Areas, CNS uses plastic conduit when installed embedded in concrete or below grade. It is impractical to replace these configurations with metal conduits. 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.

This is similar to the request made by Waterford Steam Electric Station, Unit 3 (ADAMS Accession Number ML113220230), and Arkansas Nuclear One, Unit 2.

Acceptance Criteria Evaluation

The intent of the NFPA 805 requirement is mainly to provide protection of conductors from an exposure fire and to ensure the raceway (i.e. plastic conduit) does not contribute to the fire source/overall combustible loading of the fire area. For the locations where NRC approval is requested, fire ignition or self-sustaining combustion of the plastic conduit is not plausible. In addition, plastic conduit embedded in concrete or located below grade is not expected to be exposed to a fire in an adjacent area given the fire resistive characteristics of concrete.

CNS Procedure 7.3.55 requires the plastic conduit to be limited to the above-mentioned locations, and must be evaluated and approved by Design Engineering for its intended use.

Nuclear Safety and Radiological Release Performance Criteria:

The presence of plastic conduit currently embedded in concrete or installed below grade, and potential future similar configurations does not affect nuclear safety as the embedded or below grade conduit are not subject to failure from an external source and circuit damage would not occur. Therefore, there is no impact on the NSPC.

The presence of plastic conduit in the above-mentioned locations has no impact on the radiological release performance criteria. The radiological release review was performed based on the fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the type of conduit material. The conduit material does not change the radiological release evaluation, which concluded that potentially contaminated water

is contained and smoke is monitored. The plastic conduit does not add additional radiological materials to the area or challenge system boundaries.

Safety Margin and Defense-in-Depth:

The presence of plastic conduit embedded in concrete or installed below grade are not subject to fire damage via an external source and circuit damage will not occur. The type of conduit material will not adversely impact the ability of the station to achieve and maintain fire safe shutdown. Therefore, the safety margin inherent in the analysis for a fire event has been preserved.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) Preventing fires from starting (e.g., combustible/hot work controls),
2) Detecting fires quickly and extinguishing those that occur, thereby limiting damage (e.g., fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and
3) 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 (e.g., fire barriers, fire rated cable, success path remains free of fire damage, recovery actions).

The presence of plastic conduit in the above-mentioned locations does not impact fire protection defense-in-depth. The presence and location of the plastic conduit does not compromise administrative fire prevention controls, automatic fire detection and suppression functions, manual fire suppression functions, or post-fire safe shutdown capabilities.

Conclusion:

NPPD 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).

Approval Request 5

NFPA 805 (2001), Section 3.3.7.2 states:

Outdoor high-pressure flammable gas storage containers shall be located so that the long axis is not pointed at buildings.

Bulk storage of hydrogen gas, in D.O.T.-approved high pressure cylinders, is located in a separate building approximately 80 feet east of the Water Treatment Building. The long axis of the hydrogen storage containers is pointed towards the Intake Structure to the north.

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:

- The hydrogen gas cylinders are located in an existing building. It is not possible to reconfigure the cylinders to resolve this deviation within the existing structure.
- It is impractical to relocate or modify the building that stores the hydrogen gas cylinders, based on the separation distance of 100 feet.

This is similar to the request made by Virgil C. Summer Nuclear Station Unit 1 (ADAMS Accession Number ML11321A172).

Acceptance Criteria Evaluation

The walls of the hydrogen storage structure are constructed of robust reinforced poured concrete. Per the hydrogen storage system vendor manual, each D.O.T. hydrogen cylinder is provided with two (2) mounting frames which provide the necessary support and restraint in the event of failure. A separation distance of 100 feet between the Hydrogen Storage Building and the Intake Structure offers additional protection to the one (1) foot thick poured concrete structure housing the hydrogen cylinders. The hydrogen storage building is constructed with an open east boundary to the Missouri River, and therefore, the development of an explosive atmosphere is not postulated for the structure.

Nuclear Safety and Radiological Release Performance Criteria:

The location/configuration of the bulk hydrogen storage cylinders at CNS does not affect nuclear safety due to the above mentioned storage building construction features and separation distance from the Intake Structure. A hydrogen system fire in this building would not impact any safety related targets. Therefore, there is no impact on the NSPC.

The radiological release review was performed based on the fire suppression activities in areas containing or potentially containing radioactive materials and is not impacted by the orientation of the hydrogen containers. The Hydrogen Storage Building does not contain any radioactive materials, and therefore, the configuration with the long axis of the hydrogen storage containers pointing towards the Intake Structure has no impact on the radiological release performance criteria.

Safety Margin and Defense-in-Depth:

The bulk hydrogen storage cylinders located in a separated, detached building of robust concrete construction will not adversely impact the ability of the plant to achieve and maintain fire safe shutdown, as a fire involving the hydrogen storage will not impact safety related targets. Therefore, the safety margin inherent in the analysis for a fire event has been preserved.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) Preventing fires from starting (e.g., combustible/hot work controls),
2) Detecting fires quickly and extinguishing those that occur, thereby limiting damage (e.g., fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and
3) 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 (e.g., fire barriers, fire rated cable, success path remains free of fire damage, recovery actions).

The location/configuration of the bulk hydrogen storage cylinders does not impact fire protection defense-in-depth. The location/configuration of the bulk hydrogen storage cylinders does not compromise administrative fire prevention controls, automatic fire detection and suppression functions, manual fire suppression functions, or post-fire safe shutdown capabilities.

Conclusion:

NPPD 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).

Approval Request 6

NFPA 805 (2001), Section 3.5.3 states:

> *Fire pumps, designed and installed in accordance with NFPA 20, "Standard for the Installation of Stationary Pumps for Fire Protection," shall be provided to ensure that 100 percent of the required flow rate and pressure are available assuming failure of the largest pump or pump power source.*

NFPA 20 (1999), Section 7-5.2.3 states:

> *Manual Electric Control at Remote Station. Where additional control stations for causing nonautomatic continuous operation of the pumping unit, independent of the pressure-actuated switch, are provided at locations remote from the controller, such stations shall not be operable to stop the motor.*

The electric motor driven fire pump (FP-P-E) is normally configured for automatic start on low system pressure. This pump requires manual operation to stop once started. FP-P-E can be stopped locally at the pump controller, and remotely in the Control Room.

The ability to stop FP-P-E after auto-start from the Control Room 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:

- The ability to stop FP-P-E from the Control Room after auto-start resolved Atomic Energy Commission (AEC) Concern 102 during the licensing of CNS in 1973. This AEC Concern regarded the ability to quickly shut off the fire water supply in case of internal flooding. Although this design change was not specifically committed to by NPPD, or credited in the AEC Safety Evaluation Report for CNS, NPPD does not believe it would be appropriate to eliminate this AEC-requested feature.

This is similar to the request made by Fort Calhoun Station (ADAMS Accession Number ML11276A118).

Acceptance Criteria Evaluation

In the case where the electric motor-driven fire pump is stopped, either remotely or locally, the redundant diesel engine drive fire pump FP-P-D will operate upon loss of fire water supply header pressure as designed, and is sized to provide water for the largest fire suppression system demand plus the simultaneous flow of 1,000 gpm from manual hose stations for two hours. An alarm will annunciate in the Control Room if the automatic start of FP-P-E is overridden, precluding inadvertent performance of this action.

This deviation to NFPA 20 was accepted by ANI (Design Change 88-222F) based on prior NEPIA approval in 1973 of the NPPD resolution to AEC Concern #102, regarding the installation of emergency stop switches in the Control Room to provide a quick shut-off capability of the fire pumps in the event of line break or other catastrophe which might cause flooding of critical equipment needed for safe shutdown. NEPIA comments were incorporated

into the design and installation of the emergency stop switches. As part of the post-TMI Detailed Control Room Design Review, the emergency stop switches were modified to be pull-to-lock switches, therefore requiring ANI review and acceptance documented in Design Change 88-222F.

Nuclear Safety and Radiological Release Performance Criteria:

The ability to remotely stop the electric motor driven fire pump does not affect nuclear safety, as there are strict administrative controls placed over the monitoring and control of the fire pump. Redundant diesel driven fire pump FP-P-D would be present to ensure site wide fire protection systems remain in service in the event the electric motor driven fire pump is taken off line. Therefore there is no impact on nuclear safety performance criteria.

The radiological release review was performed based on the fire suppression activities in areas containing or potentially containing radioactive materials and is not impacted by the remote stopping capability of the electric fire pump. This does not change the radiological release evaluation, which concluded that potentially contaminated water is contained and smoke is monitored. This pump remote stop capability does not add additional radiological materials to the area or challenge system boundaries and therefore, has no impact on the radiological release performance criteria.

Safety Margin and Defense-in-Depth:

As adequate site wide fire protection capability is provided via the redundant diesel driven fire pump, the remote stopping capability of the electric motor driven fire pump will not adversely impact the ability of the plant to achieve and maintain fire safe shutdown. Therefore, the safety margin inherent in the analysis for a fire event has been preserved.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) Preventing fires from starting (e.g., combustible/hot work controls),
2) Detecting fires quickly and extinguishing those that occur, thereby limiting damage (e.g., fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and
3) 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 (e.g., fire barriers, fire rated cable, success path remains free of fire damage, recovery actions).

The means of remotely stopping the electric motor-driven fire pump does not impact fire protection defense-in-depth. This feature does not compromise administrative fire prevention controls, automatic fire detection and suppression functions, manual fire suppression functions, or post-fire safe shutdown capability. Means are available to ensure that the fire pump, or the redundant fire pump, is operable during a fire event.

Conclusion:

NPPD 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).

Approval Request 7

NFPA 805 (2001), Section 3.6.1 states:

For all power block buildings, Class III standpipe and hose systems shall be installed in accordance with NFPA 14, "Standard for the Installation of Standpipe, Private Hydrant, and Hose Systems."

NFPA 14 (1974), Section 322, states:

The number of hose stations for Class II service in each building and each section of a building divided by firewalls shall be such that all portions of each story of the building are within 30 feet of a nozzle when attached to not more than 100 feet of hose.

NFPA 14 (1974), Section 442, states:

Where the pressure at any standpipe outlet exceeds 100 pounds per square inch, an approved device shall be installed at the outlet to reduce the pressure with required flow at the outlet to 100 pounds per square inch.

NFPA 14 (1974), Section 625, states:

Valves shall be of the approved extra heavy flanged pattern where the system pressure (including the pressure at shutoff as measured at the discharge flange of the permanently installed fire pump) exceeds 175 pounds per square inch.

NFPA 14 (1974), Section 671, states:

Water flow alarms should be provided on all standpipe risers where required by the authority having jurisdiction.

The NFPA 14 deviations are listed as follows for each referenced NFPA section:

NFPA 14, Section 322 - Fire Zones 14A-D, 15, 16, 18B-E, 20A-B, 23A-C and 25 are not protected by standpipe and hose stations. Accordingly, the limitation of using a maximum of 100 feet of hose is exceeded.

NFPA 14, Section 442 - Locations where the standpipe outlet pressures exceed 100 psi are present throughout CNS. Pressure reducers are not installed at any of these locations.

NFPA 14, Section 625 - The Fire Water System pressure is slightly above 175 psi for standpipe hose valves installed at elevations below grade (903'). The standpipe hose valves at these locations are rated at 175 psi, but are not of the approved extra heavy flanged pattern.

NFPA 14, Section 671 - Fire Zones 19B, 19C, and 24 are the only zones equipped with water flow alarms.

Basis for Request

The basis for the approval request of this deviation is:

- It is impractical to modify the Fire Water System to conform to the above criteria. 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

NFPA 14, Section 322

Although Fire Zones 14A-D, 15, 16, 18B-E, 20A-B, 23A-C and 25 are not protected by stand-pipe and hose stations, fire protection is provided by outside hydrants and hoses. NFPA limits the length of hose from the hydrant to 500 feet. None of the above fire zones requires more than 350 feet of hose to reach any portion of the fire zone. Therefore, this fire protection arrangement is considered to be an equivalent level of protection.

All portions of Fire Zones 1F, 10B, 11L, 12B and 12F cannot be reached with 100 feet of hose plus 30 feet of hose stream. However, all of these fire zones, except for 1F and 11L (discussed below), can be reached with a maximum additional 50 feet of hose. Procedure 5.1INCIDENT ensures that the Fire Brigade has additional lengths of hose available for these locations, which subsequently ensures that all portions of these fire zones can be reached with a hose stream. Hose stations protecting these zones are supplied by a minimum 4-inch stand-pipe which minimizes friction loss even with heavy hose stream.

- Fire zone 1F is the Torus and 11L is the pipe chase between the Reactor Building and the Turbine Building. Both of these fire zones have a negligible combustible loading and transient combustibles are controlled in accordance with Procedure 0.7.1. In addition, in the event of a fire in these areas, Procedure 5.1INCIDENT ensures that the Fire Brigade has additional lengths of hose available to ensure that all portions of these two fire zones can be reached with a hose stream. The site fire pumps are capable of providing sufficient outlet pressure for the additional required hose. Therefore, the location and spacing of the existing hose stations is considered to be adequate.

NFPA 14, Section 442

The intent of this section is to limit hose pressures seen by fire hose handlers. However, in accordance with Procedure 0.23, only Fire Brigade qualified personnel are allowed to utilize fire hose stations at CNS. All Fire Brigade personnel are trained in the use of heavy hose streams, such as CNS fire hose stations at full system pressure. Furthermore, NFPA 14 (current edition) permits static pressures of 175 psi at Class I (2 ½") and Class II (1 ½") hose stations, well above the 1974 requirement of 100 psi. Although NFPA 14 (current) edition requires the installation of a pressure reducer when the residual pressure exceeds 100 psi for a Class II hose station (175 psi residual pressure for a Class I hose station), the intent of the requirement is to limit the pressure so that "trained occupants" can use the hose outlet. As discussed, only trained Fire Brigade personnel are permitted to use the hose stations. Therefore, hose station residual pressures between 100 psi and 175 psi are deemed acceptable for use by the Fire Brigade. All hose stations are rated for 175 psi per NFPA 14, Section 625 discussion below.

NFPA 14, Section 625

These standpipe hose valves are rated at 175 psi, but are hydrostatically pressure tested to 350 psi. This hydrostatic test pressure is well above the system pressure. Therefore, the existing valves are deemed acceptable as installed, and are capable of performing their intended function.

NFPA 14, Section 671

The intent of this section is to alert station personnel of a fire, based on fire hose station water flow. However, only Fire Brigade members are allowed to use the fire hose stations at CNS. Therefore, if the Fire Brigade is using the hose station, then the fire has already been reported to the Control Room, and the fire alarm system has been activated. A water flow alarm is not necessary to notify the Control Room that a hose station is in use. This minor deviation does not prevent the system from performing its intended function.

Nuclear Safety and Radiological Release Performance Criteria:

The above referenced standpipe system features do not affect nuclear safety, as adequate manual fire fighting capability is provided for these areas. Therefore, there is no impact on the nuclear safety performance criteria.

The radiological release review was performed based on the fire suppression activities in areas containing or potentially containing radioactive materials and is not impacted by these standpipes system features. This does not change the radiological release evaluation, which concluded that potentially contaminated water is contained and smoke is monitored. The above referenced standpipe system features do not add additional radiological materials to the area or challenge system boundaries, and therefore, have no impact on the radiological release performance criteria.

Safety Margin and Defense-in-Depth:

The above referenced standpipe system features will not adversely impact the ability of the station to achieve and maintain fire safe shutdown, as adequate manual fire suppression capability is provided for these areas. Therefore, the safety margin inherent in the analysis for a fire event has been preserved.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) Preventing fires from starting (e.g., combustible/hot work controls),
2) Detecting fires quickly and extinguishing those that occur, thereby limiting damage (e.g., fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and
3) 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 (e.g., fire barriers, fire rated cable, success path remains free of fire damage, recovery actions).

The above referenced standpipe system features does not impact fire protection defense-in-depth. Its presence does not compromise administrative fire prevention controls, automatic fire

detection and suppression functions, manual fire suppression functions, or post-fire safe shutdown capabilities.

Conclusion:

NPPD 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).

Approval Request 8

NFPA 805 (2001), Section 3.7 states:

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 the fire areas due to radiological conditions.

NFPA 10 (1975), Section 3-3 provides requirements for fire extinguisher size and placement for Class B fires *other than for fires in flammable liquids of appreciable depth.*

CNS Fire Zones 5B, 11C, 11D, 11H, 11J, 11K, 12C, 13A (behind bio-shield wall only), and 18D contain Class B hazards (flammable liquids) that are not of appreciable depth. Class B fire extinguishers are provided in, and adjacent to, the stated fire zones; however their placement is deficient as the travel distance requirements of Table 3-3.1.2 are exceeded.

Basis for Request

The basis for the approval request of this deviation is as follows:

- It is impractical to modify the existing fire extinguisher size and placement to fully conform to the above criteria. Moreover, the fire brigade utilizes their own equipment, and does not rely on the installed extinguishers. 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

Fire Zones 11D, 12C, and 13A (behind bio-shield wall only) are provided with automatic suppression systems protecting the lube oil hazards. In addition, large wheeled fire extinguishers are provided in adjacent fire zones and can be used within Fire Zones 11D, 12C, and 13A. The presence of fire extinguishers adjacent to the "ALARA concerned" areas satisfies the intent of NFPA 805, Section 3.7.

Fire Zones 5B, 11C, 11H, 11J, 11K, and 18D are provided with automatic suppression systems. Additional Class B fire extinguishers are also provided in adjacent fire zones.

All of these fire zones are provided with fire detection (smoke, heat, or flame) which alarms in the Control Room to initiate prompt fire investigation and Fire Brigade response. Additionally, standpipes are provided within, or adjacent to, these fire zones.

Nuclear Safety and Radiological Release Performance Criteria:

The number and locations of existing Class B fire extinguishers does not affect nuclear safety as adequate manual and fixed suppression capability are provided for these areas. Therefore, there is no impact on the NSPC.

The radiological release review was performed based on the fire suppression activities in areas containing or potentially containing radioactive materials and is not impacted by the location and number of portable fire extinguishers. This does not change the radiological release evaluation,

which concluded that potentially contaminated water is contained and smoke is monitored. The number and locations of existing Class B fire extinguishers do not add additional radiological materials to the area or challenge system boundaries and therefore have no impact on the radiological release performance criteria.

Safety Margin and Defense-in-Depth:

The number and locations of existing Class B fire extinguishers will not adversely impact the ability of the station to achieve and maintain fire safe shutdown, as adequate manual and fixed suppression capability is provided in these areas.. Therefore, the safety margin inherent in the analysis for a fire event has been preserved.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) Preventing fires from starting (e.g., combustible/hot work controls),
2) Detecting fires quickly and extinguishing those that occur, thereby limiting damage (e.g., fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and
3) 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 (e.g., fire barriers, fire rated cable, success path remains free of fire damage, recovery actions).

The number and locations of existing Class B fire extinguishers does not impact fire protection defense-in-depth. It does not compromise administrative fire prevention controls, automatic fire detection and suppression functions, manual fire suppression functions, or post-fire safe shutdown capabilities.

Conclusion:

NPPD 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).

Approval Request 9

NFPA 805 (2001), Section 3.10.5 states:

> *Provisions for locally disarming automatic gaseous suppression systems shall be secured and under strict administrative control.*

The Diesel Generator (DG) carbon dioxide (CO_2) system is provided with local abort switches. However, they are not key-controlled, and thus are not "secured" as intended by this NFPA 805 element.

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 as follows:

- It is impractical to redesign the DG CO_2 system to include abort switches that are key-locked, as there are adequate administrative controls and procedures currently in place to ensure only authorized personnel can lock-out the system. 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

Only authorized personnel are allowed to abort the system. This would include Fire Brigade members and Operations personnel, who are trained in the methods of manually actuating the system while in the "Abort" position. Although the DG CO_2 system is not provided with keyed abort switches, they are provided with abort switches that will annunciate at the local panel and in the Control Room upon operation. Fire Brigade members and Control Room Operators are trained in the locations of station CO_2 systems, automatic and manual methods of actuation, system interlocks and time delays, abort switch operation, and actions required upon abort switch annunciation on the Control Room suppression panel.

CNS Procedure 2.2.2 provides instructions/safeguards required to safely reset the system to normal operation following system discharge or system abort. The procedure also provides instructions required to safely abort the system or take the system out of service. A fire watch, established in accordance with CNS Procedure 0.39.1, is required when the system is aborted or taken out of service. Therefore, these procedures ensure that the systems are not left in the disarmed mode following the manual aborting action; or that if the systems are intended to be left out of service, procedural controls ensure that a fire watch is in place to ensure adequate fire protection is maintained for the DG areas.

Nuclear Safety and Radiological Release Performance Criteria:

The administrative controls described above provide assurance that the presence of a "non-keyed" system abort switch does not affect nuclear safety, as a fire watch is required when the system is aborted or taken out of service. Therefore, there is no impact on the NSPC. The functional attributes of the abort switch remain the same.

The radiological release review was performed based on the fire suppression activities in areas containing or potentially containing radioactive materials and is not impacted by the presence of a "non-keyed" abort switch. The DG areas do not contain any radioactive materials, and therefore, the presence of a "non-keyed" system abort switch has no impact on the radiological release performance criteria.

Safety Margin and Defense-in-Depth:

The presence of a "non-keyed" system abort switch will not adversely impact the ability of the station to achieve and maintain fire safe shutdown, as adequate manual suppression capability is provided in the event that the CO_2 system is aborted or out-of-service. Therefore, the safety margin inherent in the analysis for a fire event has been preserved.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) Preventing fires from starting (e.g., combustible/hot work controls),
2) Detecting fires quickly and extinguishing those that occur, thereby limiting damage (e.g., fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and
3) 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 (e.g., fire barriers, fire rated cable, success path remains free of fire damage, recovery actions).

The presence of a "non-keyed" system abort switch does not impact fire protection defense-in-depth. Its presence does not compromise administrative fire prevention controls, automatic fire detection and suppression functions, manual fire suppression functions, or post-fire safe shutdown capabilities.

Conclusion:

NPPD 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).

Approval Request 10

NFPA 805 (2001), Section 3.10.7

NFPA 805, Section 3.10.7 states:

> *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.*

The Diesel Generator (DG) carbon dioxide (CO_2) systems is not provided with an odorizer.

This is considered to be a deviation from this NFPA element, for which NRC approval is requested.

Basis for Request

The basis for the approval request of this deviation is:

- It is impractical to redesign the DG CO_2 system to include an odorizer, as there are adequate safeguards currently in place. 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

The intent of the NFPA 805 odorizer requirement is to provide notification to personnel to exit an enclosed space prior to an impending and toxic carbon dioxide system discharge. As stated in Section 1.7 (Equivalency) of NFPA 805, "Nothing in this standard is intended to prevent the use of systems, methods, or devices of equivalent or superior quality, strength, fire resistance, effectiveness, durability, and safety over those prescribed by this standard." The following safeguards are present to ensure personnel safety:

- Strobe lights located inside the protected space and outside the protected space serve as an alert in the event of carbon dioxide system actuation.

- Audible and visual pre-discharge signals have been provided inside and outside of the protected space.

- A 50-second pneumatic time delay is provided following actuation of the carbon dioxide system to allow for personnel evacuation prior to system discharge.

- Appropriate warning signs are affixed outside the area covered by the carbon dioxide system. Another placard is located inside the hazard area, advising immediate evacuation in the event of the system alarm sounding.

It is noted that NFPA 12 (either the 1972 code of record edition or the current 2011 edition) does not require the presence of an odorizer, however, it is suggested in the Appendix as a potential safeguard (i.e. "above and beyond" required features) for alerting personnel to imminent system

discharge. Per the NFPA 12 Code Compliance Review, all "required" safeguards for "Hazards to Personnel" are satisfied for the as-installed carbon dioxide system configuration.

Nuclear Safety and Radiological Release Performance Criteria:

The lack of an odorizer is a personnel safety issue that does not affect nuclear safety. Adequate fixed suppression capability is provided by the CO_2 system, and therefore, there is no impact on the NSPC. The functional attributes of the carbon dioxide system remains the same.

The radiological release review was performed based on the fire suppression activities in areas containing or potentially containing radioactive materials and is not impacted by the lack of an odorizer. The DG areas do not contain any radioactive materials, and therefore, the lack of an odorizer has no impact on the radiological release performance criteria.

Safety Margin and Defense-in-Depth:

The lack of an odorizer will not adversely impact the ability of the station to achieve and maintain fire safe shutdown as adequate fixed suppression capability is provided by the CO_2 system. Therefore, the safety margin inherent in the analysis for a fire event has been preserved.

NEI 04-02, Section 5.3.5.2 describes three echelons for defense-in-depth:

1) Preventing fires from starting (e.g., combustible/hot work controls),
2) Detecting fires quickly and extinguishing those that occur, thereby limiting damage (e.g., fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and
3) 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 (e.g., fire barriers, fire rated cable, success path remains free of fire damage, recovery actions).

The lack of an odorizer does not impact fire protection defense-in-depth. It does not compromise administrative fire prevention controls, automatic fire detection and suppression functions, manual fire suppression functions, or post-fire safe shutdown capabilities.

Conclusion:

NPPD 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).

ATTACHMENT M

License Condition Changes

4 Pages

NPPD proposes to replace the current CNS fire protection license condition 2.C(4) with the standard license conditions in Regulatory Position C.3.1 of Regulatory Guide 1.205, Revision 1, as shown below. In support of this change, NPPD has developed a fire Probabilistic Risk Assessment (PRA) which has been reviewed and been found acceptable by a fire PRA BWROG/GEH peer review conducted during April 2010 and January 2011. Outstanding high level findings from the peer review are included in Attachment V of this report. Any future changes to the Fire PRA will be subject to peer review in accordance with the guidance provided in NEI 07-12 and applicable ASME/ANS PRA standards.

Nebraska Public Power District shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated ___________ (and supplements dated _________) and as approved in the safety evaluation report dated ____________ (and supplements 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 CNS. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

a) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

b) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1 10^{-7}/year (yr) for CDF and less than 1 10^{-8}/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the

engineering evaluation and conclude that the change has not affected the 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 report dated ________ to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) below, risk-informed changes to NPPD'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) as provided in Table S-2 of the Cooper Nuclear Station License Amendment Request dated April 27, 2012, prior to startup from the first refueling outage greater than 12 months following the issuance of the License Amendment.
3) The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

The license condition to be replaced is restated below.

License Condition 2.C.(4):

> The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Cooper Nuclear Station (CNS) Updated Safety Analysis Report and as approved in the Safety Evaluations dated November 29, 1977; May 23, 1979; November 21, 1980; April 29, 1983; April 16, 1984; June 1, 1984; January 3, 1985; August 21, 1985; April 10, 1986;

September 9, 1986; November 7, 1988; February 3, 1989; August 15, 1995; and July 31, 1998, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

It is NPPD's understanding that implicit in the replacement of this license condition, all prior fire protection program SERs and commitments will be superseded in their entirety by the revised license condition.

No other license conditions need to be replaced or revised.

ATTACHMENT N

Technical Specification Changes

2 pages

Technical Specification 5.4.1 d. will be deleted.

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

d. ~~Fire Protection Program implementation.~~ Not Used

NPPD determined that this change to the Technical Specifications is acceptable for adoption of the new fire protection licensing basis since the requirement for establishing, implementing, and maintaining fire protection procedures is embodied in 10 CFR 50.48(c)(1), which approves the incorporation of NFPA 805 by reference. NFPA 805 Section 3.2.3, "Procedures," in turn, states:

> Procedures shall be established for implementation of the fire protection program.

Removal of Administrative Controls Technical Specifications that are redundant to other regulatory requirements is consistent with established NRC guidance.[1]

A mark-up of the proposed change to the Technical Specifications is provided in Attachment 1 of this LAR. A clean, retyped copy of the page is provided in Attachment 2 of this letter.

[1] Letter from William T. Russell (U.S. NRC) to Chairperson of the Westinghouse, Combustion Engineering, Babcox & Wilcox, and BWR Technical Specifications Committees, dated October 25, 1993, "Content of Standard Technical Specifications, Section 5.0, Administrative Controls."

ATTACHMENT O

Orders and Exemptions

2 Pages

Exemptions

Rescind the following exemptions granted against 10 CFR 50, Appendix R dated September 21, 1983 (see LAR Section 2.2):

- Exemption for lack of twenty foot separation between redundant service water pumps in the service water intake structure.
- Exemption for lack of twenty foot separation free of intervening combustibles or one-hour barriers between redundant trains in the cable spreading room.
- Exemption for lack of twenty foot separation or one-hour barriers between redundant trains in the cable expansion room.
- Exemption from the requirement for one-hour-rated fire barriers for redundant conduits and an area wide automatic suppression system in the northeast corner room of the 903'-6" Elevation of the Reactor Building.
- Exemption from the requirement for an automatic suppression system in the Control Building Basement.
- Exemption from a fixed fire suppression system in the Auxiliary Relay Room.
- Exemption from a fixed fire suppression system in the Control Room.
- Exemption from the requirement for a three-hour fire barrier, or twenty feet of separation free of intervening combustibles combined with automatic suppression and detection in the 931' Elevation of the Reactor Building.
- Exemption from the requirement for a three-hour fire barrier, or twenty feet of separation free of intervening combustibles combined with automatic suppression and detection, and lack of alternate shutdown capability independent of the area in the 903'-6" Elevation of the Reactor Building.
- Exemption from the requirement for a three-hour fire barrier, or twenty feet of separation free of intervening combustibles combined with automatic suppression and detection, and lack of alternate shutdown capability independent of the area in the 859' and 881' Elevations of the Reactor Building.

Specific details regarding these exemptions are contained in Attachment K. The following exemptions and their bases will be transitioned to the new licensing basis under 10 CFR 50.48(a) and 50.48(c) as previously approved (NFPA Section 2.2.7) and are therefore compliant with the new regulation.

- None

Orders

NPPD has determined that no Orders need to be superseded or revised.

ATTACHMENT P

RI-PB Alternatives to NFPA 805 10 CFR 50.48(c)(4)

No risk-informed or performance-based alternatives to compliance with NFPA 805 (per 10 CFR 50.48(c)(4)) were utilized by NPPD.

ATTACHMENT Q

No Significant Hazards Evaluations

4 Pages

The Nebraska Public Power District (NPPD) has evaluated whether or not a significant hazards consideration is involved with the proposed license amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below.

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

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

   Response: No.

   Operation of the Cooper Nuclear Station (CNS) in accordance with the proposed amendment does not result in a significant increase in the probability or consequences of accidents previously evaluated. The proposed amendment does not affect accident initiators or precursors as described in the CNS Updated Safety Analysis Report (USAR), nor does it adversely alter design assumptions, conditions, or configurations of the facility, and it does not adversely impact the ability of structures, systems, or components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the way in which safety-related systems perform their functions as required by the accident analysis. The SSCs required to safely shut down the reactor and to maintain it in a safe shutdown condition will remain capable of performing their design functions.

   The purpose of this amendment is to permit CNS to adopt a new risk-informed, performance-based fire protection licensing basis that complies with the requirements in 10 CFR 50.48(a) and 10 CFR 50.48(c), as well as the guidance contained in Regulatory Guide (RG) 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 Part 50, Appendix R, fire protection features (69 FR 33536; June 16, 2004). Engineering analyses, which may include engineering evaluations, probabilistic risk assessments, and fire modeling calculations, have been performed to demonstrate that the performance-based requirements of NFPA 805 have been met.

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

   Based on the above, the implementation of this amendment to transition the Fire Protection Plan (FPP) at CNS to one based on NFPA 805, in accordance with 10 CFR 50.48(c), does not result in a significant increase in the probability of any accident previously evaluated. In addition, all equipment required to mitigate an accident remains capable of performing the assumed function. Therefore, the consequences of any accident previously evaluated are not significantly increased with the implementation of this License Amendment Request.

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

   Response: No.

   Operation of CNS in accordance with the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. Any scenario or previously analyzed accident with offsite dose consequences was included in the evaluation of design basis accidents (DBA) documented in the USAR as a part of the transition to NFPA 805. The proposed amendment does not impact these accident analyses. The proposed change does not alter the requirements or functions for systems required during accident conditions, nor does it alter the required mitigation capability of the fire protection program, or its functioning during accident conditions as assumed in the licensing basis analyses and/or DBA radiological consequences evaluations.

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

   The purpose of the proposed amendment is to permit CNS to adopt a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and 10 CFR 50.48(c) and the guidance in Revision 1 of RG 1.205. As indicated in the Statements of Consideration, 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 fire protection features.

   The requirements in NFPA 805 address only fire protection and the impacts of fire effects on the plant have been evaluated. The proposed fire protection program changes do not involve new failure mechanisms or malfunctions that could initiate a new or different kind of accident beyond those already analyzed in the USAR. Based on this, as well as the discussion above, the implementation of this amendment to transition the FPP at CNS to one based on NFPA 805, in accordance with 10 CFR 50.48(c), 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 a margin of safety?

   Response: No.

   Operation of CNS in accordance with the proposed license amendment does not involve a significant reduction in a margin of safety. The transition to a new risk-informed, performance-based fire protection licensing basis that complies with the requirements in 10 CFR 50.48(a) and 10 CFR 50.48(c) 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 license amendment does not adversely affect existing plant safety margins or the reliability of equipment assumed in the USAR to mitigate accidents. The proposed change does not adversely impact systems that respond to safely shut down the plant and maintain the plant in a safe shutdown condition. In addition, the proposed license amendment will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without implementation of appropriate compensatory measures.

The purpose of the proposed license amendment is to permit CNS to adopt a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and 10 CFR 50.48(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 risk evaluations for plant changes, in part as they relate to the potential for reducing a safety margin, were measured quantitatively for acceptability using the delta risk guidance contained in RG 1.205. Engineering analyses, which may include engineering evaluations, probabilistic safety assessments, and fire modeling calculations, have been performed to demonstrate that the performance-based methods of NFPA 805 do not result in a significant reduction in the margin of safety.

As such, the proposed changes are evaluated to ensure that risk and safety margins are kept within acceptable limits. Based on the above, the implementation of this amendment to transition the FPP at CNS to one based on NFPA 805, in accordance with 10 CFR 50.48(c), will not significantly reduce a margin of safety.

Conclusion

NFPA 805 continues to protect public health and safety and the common defense and security because the overall approach of NFPA 805 is consistent with the key principles for evaluating risk-informed licensing basis changes, as described in RG 1.174, is consistent with the defense-in-depth philosophy, and maintains sufficient safety margins. Based on the above discussion, the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the license amendment request to transition the FPP at CNS to one based on NFPA 805, in accordance with 10 CFR 50.48(c), involves no significant hazards consideration.

ATTACHMENT R

Environmental Considerations Evaluation

2 Pages

The proposed license amendment transitions the fire protection program at Cooper Nuclear Station (CNS) to one based on NFPA 805, in accordance with 10 CFR 50.48(c), which subsequently impacts a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, as well as changing certain inspection and surveillance requirements.

The purpose of the proposed license amendment is to permit CNS to adopt a new fire protection licensing basis which complies with the requirements of 10 CFR 50.48(a) and (c) and the guidance in Revision 1 of Regulatory Guide 1.205. The NRC considers that NFPA 805 provides an acceptable methodology and performance criteria for licensees to identify fire protection requirements that are an acceptable alternative to the 10 CFR 50 Appendix R required fire protection features (69 FR 33536, June 16, 2004).

The proposed amendment does not involve:

1. A significant hazards consideration.

   As stated in Attachment Q, the proposed license amendment does not involve a significant hazards consideration.

2. A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

   Compliance with NFPA 805 explicitly requires the attainment of performance criteria, objectives, and goals for radioactive releases to the environment. This radioactive release goal is to provide reasonable assurance that a fire will not result in a radiological release that affects the public, plant personnel, or the environment. The NFPA 805 transition based on fire suppression activities, but not involving fuel damage, has been evaluated and does not create any new source terms. Therefore, the proposed license 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. The results of the evaluation performed regarding fire fighting activities provides assurance that the proposed license amendment will not significantly alter the types, or increase the amounts, of individual or cumulatirve occupational radiation exposures.

Therefore, the Nebraska Public Power District has determined that the proposed license amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this proposed license amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the license amendment.

ATTACHMENT S

Plant Modifications and Items to be Completed During Implementation

12 Pages

Tables S-1, Plant Modifications Completed, and S-2, Plant Modifications Committed, provided below, include 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 legend applies to the risk ranking indicated in Tables S-1 and S-2:

- High = Modification would have an appreciable impact on reducing overall fire CDF.
- Medium = 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.

Table S-1 Plant Modifications Completed

Item	Rank	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
S-1.1	Medium	Electrical Panels HPI-CBX-1, 2, 3 require de-energization to remove a fixed ignition source in Fire Zones 2A-2 and 2A-3 (RB-CF and RB-DI).	Permanently remove power to HPI-CBX-1, 2, 3 when at power.	Y	None	Risk is reduced, as change reduces the fire frequency and eliminates need for an RA. Defense-in-depth is also improved.

Table S-2 Plant Modifications Committed

Item	Rank	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
S-2.1 RB-K	High	Control cables associated with 4kV circuit breakers 1FA and 1FS and first-level undervoltage circuitry are routed from the A Non-Critical Switchgear Room to the F Critical Switchgear Room, but pass through the G Critical Switchgear Room. In addition, control and feeder cables associated with 4kV circuit breakers 1FA and 1FS are also routed similarly.	Cables to be re-routed such that they do not traverse the G Critical Switchgear Room.	Y	Y	Risk is reduced as change provides for AC power availability without an RA. Defense-in-depth is improved. Compensatory measure for NFPA 805: Appropriate compensatory measures will be established per CNS Procedure 0.23, as required, until the modification is implemented. Compensatory measure for 10 CFR 50 Appendix R: Yes. Alternate compensatory measures in the form of Operator Manual Actions as documented in Procedure 5.4 Post-Fire, Attachment 18, are in place for this issue. The alternate compensatory measures are implemented per CNS Procedure 0.23.
S-2.2 TB-A	High	Control cables associated with 4kV circuit breakers 1FA and 1FS and first-level undervoltage circuitry are routed along the same path as the 1GB and 1GS through the Non-Critical Switchgear Room.	Cables to be re-routed such that they do not traverse any fire scenarios common with the control cables associated with the 1GB and 1GS breakers in the Non-Critical Switchgear Room.	Y	Y	Risk without crediting all current RA is reduced as change provides for ac power availability without RA for design basis scenarios. However, some existing RA provide for restoring power for beyond design basis scenarios. Compensatory measure for NFPA 805: Appropriate compensatory measures will be established per CNS Procedure 0.23, as required, until the modification is implemented.

Table S-2 Plant Modifications Committed

Item	Rank	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
						Compensatory measure for 10 CFR 50 Appendix R: Yes. Alternate compensatory measures in the form of Operator Manual Actions as documented in Procedure 5.4 Post-Fire, Attachment 24, are in place for this issue. The alternate compensatory measures are implemented per CNS Procedure 0.23.
S-2.3 CB-A	Medium	Feeder cables to 1a and 1B 125 and 250V DC Battery Chargers are routed along the same path through the Control Building Controlled Corridor resulting in a loss of all battery chargers.	Cables MLX36 and MLX37 to be re-routed through the RPS Rooms to provide a minimum of a channel of battery chargers to support long term battery usage.	Y	Y	Risk is reduced as DC power availability is improved. Defense-in-depth is improved. Compensatory measure for NFPA 805: Appropriate compensatory measures will be established per CNS Procedure 0.23, as required, until the modification is implemented. Compensatory measure for 10 CFR 50 Appendix R: Yes. Alternate compensatory measures in the form of Operator Manual Actions as documented in Procedure 5.4 Post-Fire, Attachment 3, in place for this issue. The alternate compensatory measures are implemented per CNS Procedure 0.23.
S-2.4 CB-D	High	Control Room abandonment is required along with the usage of the alternate shutdown procedures for a fire in Panel 9-32 or 9-33.	Install incipient detection in Panel 9-32 and 9-33 in the Auxiliary Relay Room (Fire Zone 8A) allows for shutdown from the Control Room with minimal field actions.	Y	Y	Risk is reduced considerably as the installation reduces the frequency of Control Room abandonment. Defense-in-depth is improved.

Table S-2 Plant Modifications Committed

Item	Rank	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
						Compensatory measure for NFPA 805: Appropriate compensatory measures will be established per CNS Procedure 0.23, as required, until the modification is implemented. Compensatory measure for 10 CFR 50 Appendix R: None. Fire Area CB-D is deterministically compliant with 10 CFR 50 Appendix R.
S-2.5 RB-M	Medium	Transient fires impacting multiple trains of conduit outside the Critical Switchgear Room common passageway.	Install conduit shields to prevent damage to conduit banks from transient fires in Fire Area RB-M/Fire Zone 3C.	Y	Y	Risk is reduced as equipment available to provide core cooling is protected. Defense-in-depth is improved. Compensatory measure for NFPA 805: Appropriate compensatory measures will be established per CNS Procedure 0.23, as required, until the modification is implemented. Compensatory measure for 10 CFR 50 Appendix R: Yes. Alternate compensatory measures in the form of Operator Manual Actions as documented in Procedure 5.4 Post-Fire, Attachment 19, are in place for this issue. The alternate compensatory measures are implemented per CNS Procedure 0.23.
S-2.6 RB-M	Medium	Transient fires impacting vertical cable trays in corner of RB-M for opposite train.	Install bottom tray covers to prevent damage to cable tray risers from	Y	Y	Risk is reduced as equipment available to provide core cooling is protected. Defense-in-depth is

Table S-2 Plant Modifications Committed

Item	Rank	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
			transient fires in Fire Area RB-M/Fire Zone 3C.			improved. Compensatory measure for NFPA 805: Appropriate compensatory measures will be established per CNS Procedure 0.23, as required, until the modification is implemented. Compensatory measure for 10 CFR 50 Appendix R: Yes. Alternate compensatory measures in the form of Operator Manual Actions as documented in Procedure 5.4 Post-Fire, Attachment 19, are in place for this issue. The alternate compensatory measures are implemented per CNS Procedure 0.23.
S-2.7 CB-D	Medium	LNK 6 fire impacts conduits and trays in vicinity requiring Control Room abandonment along with usage of the alternate shutdown procedure.	Installation of board shielding for cable trays and conduit to prevent damage from fires involving panel PMIS-MUX-LNK6 and PMIS-MUX-LNK7 in the Cable Spreading Room (Fire Zone 9A).	Y	Y	Risk is reduced as equipment available to provide core cooling is protected. Defense-in-depth is improved. Compensatory measure for NFPA 805: Appropriate compensatory measures will be established per CNS Procedure 0.23, as required, until the modification is implemented. Compensatory measure for 10 CFR 50 Appendix R: None. Fire Area CB-D is deterministically compliant with 10 CFR 50 Appendix R.

Table S-3: Items provided below are certain items (procedure changes, process updates, and training to affected plant personnel) that will be completed pursuant to the implementation of the new NFPA 805 Fire Protection Program. This will occur within six (6) months after NRC approval of the NFPA 805 Transition License Amendment Request.

Table S-3 Implementation Items

Item	Description	LAR Section / Source
S-3.1	During the implementation of the NFPA 805 licensing basis, performance-based surveillance frequencies will be established as described in Electric Power Research Institute (EPRI) Technical Report 1006756, "Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features". The performance-based surveillance frequencies will be evaluated in the monitoring program in accordance with NFPA 805 FAQ 10-0059.	Attachment A
S-3.2	Enhanced transient and combustible controlled zones will be established in high risk Fire Zones 8A and 9A. Enhanced transient and combustible controlled locations will be established in the following specific fire zone locations to address high risk transient fire scenarios: Fire Zone 2C above the TIP Room, and Fire Zones 3C and 3D in the areas around instrument racks 25-5 and 25-6.	4.5 and Attachment W
S-3.3	Post-fire operating procedures will be updated to reflect new NSCA strategies and training performed as necessary.	Attachments G and V
S-3.4	Technical, Operations, and administrative procedures and documents that relate to non-power modes of plant operating states will be revised as needed for implementation of NFPA 805.	4.3.2 and Attachment D
S-3.5	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.	4.7.1
S-3.6	A confirmatory demonstration (field validation walk-through) of the feasibility for the credited NFPA 805 RA will be performed. This will include field validation of: (1) Transit times (i.e., travel times to/from recovery action manipulated plant equipment). (2) Execution times (i.e., time required to physically perform the action, such as handwheel a valve open, open a breaker, etc.). (3) Communications for adequacy between the controlling location and RA locations for areas which involve actions.	4.2.1.3 and Attachment G

Table S-3 Implementation Items

Item	Description	LAR Section / Source
	(4) Adequate lighting (either fixed or portable) for access/egress and local lights are provided for the component to be operated.	
S-3.7	CNS calculations will be reviewed and updated based on the results of the field walkdowns of the recovery actions from Implementation Item S-3.6.	4.2.1.3 and Attachment G
S-3.8	The FPP procedures will be revised to specify application of the NFPA 805 Section 2.7.3 quality requirements.	Section 4.7.3
S-3.9	Procedure 0.23 will be revised to identify the Authority Having Jurisdiction for the various areas of the Fire Protection Program.	Attachment A
S-3.10	Administrative procedures will be revised to control the use of portable electric heaters and revised to document that portable fuel-fired heaters are not permitted in plant areas containing equipment important to nuclear safety, or where there is a potential for radiological release resulting from a fire.	Attachment A
S-3.11	Emergency procedures will be updated to allow use of Service Water (SW) pumps alone to provide cooling to Residual Heat Removal (RHR) heat-exchangers in the event RHR SW booster pumps are rendered unavailable.	Attachment W
S-3.12	Procedures will be revised to require new cable installations to meet the requirements of IEEE-383, or similar.	Attachment A
S-3.13	Procedure 0.7.1 will be revised to include requirement that bulk gas storage not be allowed inside structures housing systems, equipment, or components important to nuclear safety.	Attachment A
S-3.14	Procedures will be revised to include the requirement for the inspection of the transformer spill containment area.	Attachment A
S-3.15	For personnel performing fire modeling or Fire PRA development and evaluation, NPPD will develop and maintain qualification requirements for individuals assigned various tasks. Position Specific Guides will be developed to identify and document required training and mentoring to ensure individuals are appropriately qualified per the requirements of NFPA 805 Section 2.7.3.4 to perform assigned work.	4.7.3

Table S-3 Implementation Items

Item	Description	LAR Section / Source
S-3.16	Procedures will be revised to inventory which pre-fire plans are contained in the fire lockers, and ensure that updates of the pre-fire plans include replacing the updated pages in each of the inventoried locations throughout the plant.	Attachment A
S-3.17	Procedures will be revised to ensure that pre-fire plan drawings are maintained in the Control Room and to ensure that the latest revisions are available.	Attachment A
S-3.18	The fire brigade training program will be updated to include guidance to ensure fire drills are conducted in various plant areas, especially in those areas identified to be essential to plant operation and to contain significant fire hazards.	Attachment A
S-3.19	Not used.	
S-3.20	Pre-fire plans and training materials will be revised to address radioactive *release requirements of* NFPA 805.	Attachment E
S-3.21	Revise Procedure 0.7.1 to include the requirement for ventilation duct materials to be non-combustible or listed by a nationally recognized testing Laboratory, such as Factory Mutual or Underwriters Laboratory, Inc., for flame spread index of 25 or less and a smoke development index of 50 or less.	Attachment A
S-3.22	Procedures will be revised to ensure that the fire protection system is not to be used for non-emergency usage.	Attachment A
S-3.23	The NFPA 805 Monitoring Program will be developed and implemented, as described in Section 4.6.	Section 4.6
S-3.24	The Fire PRA database will be controlled as an electronic document in the same way the Internal Events PRA model (CAFTA model) is controlled.	Attachment V
S-3.25	Fire Zone 9A (Cable Spreading Room) and Fire Zone 8A (Auxiliary Relay Room) will be designated as enhanced transient and hot work controlled fire zones.	Attachment V
S-3.26	The CNS Updated Safety Analysis Report will incorporate the applicable subject matter described in FAQ 12-0062, at a level of detail consistent with NEI 98-03, “Guidelines for Updating Final Safety Analysis Reports.”	Section 5.4

Table S-3 Implementation Items

Item	Description	LAR Section / Source
S-3.27	The configuration control procedures which govern the various CNS documents and databases that currently exist will be revised to reflect the new NFPA 805 licensing bases requirements.	Section 4.7.2
S-3.28	Control procedures and processes for NSCA supporting information, Non-Power Mode Review, Fire Modeling Calculations, Fire Safety Assessments, risk evaluations, etc., will be developed.	Section 4.7.2
S-3.29	System level Design Criteria Documents will be revised to reflect the NFPA 805 role that the system components now play.	Section 4.7.2

ATTACHMENT T

Clarification of Prior NRC Approvals

None.

ATTACHMENT U

Internal Events PRA Quality

23 Pages

The CNS PRA has undergone a RG 1.200 Rev. 1 Peer Review against the ASME PRA Supporting Requirements by a team of knowledgeable industry (vendor and utility) personnel. The review was conducted in May of 2008. Based on the Peer Review that was performed, NPPD concludes that the CNS PRA model substantially meets the ASME PRA Standard and can be used to support risk-informed applications.

The Peer Review is discussed in greater detail in Section 4.5.1.1. The PRA Review Report will be made available upon request. The resumes and qualifications of the team members are provided in the report. The following table details the ASME PRA Standard Category II open supporting requirements, the actions taken to address them, and the impact on the NFPA 805 application. The other findings associated with SR in which Capability Category II was met are also included.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
HR-G7	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)* 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)]	SR not met. Dependencies between post-initiator actions have been accounted for in general and appear adequate. However, there are some dependencies which do not appear to have been evaluated (or at least documented). In particular, the use of two different "floors" for joint HEPs is a little questionable, as is its application. The Cooper HRA identifies cutsets with multiple HFEs and provides a method for assessing the degree of those dependencies. May want to consider using more recent dependency models. The Peer Review referred to the following five dependencies sets of examples: %FLSWRBM * FLD-XHE-FO-MSWRB * FLD-XHE-FO-SWRS1 * SWS-XHE-FO-SWNHP FPS-XHE-FO-RPVIN * HVC-XHE-FO-ALTQC ADS-XHE-FO-TRANS * HVC-XHE-FO-CB7A ECS-XHE-FO-TRANS * SWS-XHE-FO-SWBPS ADS-XHE-FO-3ALEG * SWS-XHE-FO-SWBPS	This finding has been addressed. Each of the identified human error probability (HEP) combinations identified by example in the finding was reexamined. These combinations of HEPs represent combinations of HEPs that were evaluated after the original dependent HEP cutset calculation. These were evaluated during the final model review and determined to be composed of independent HEPs that led to combined probabilities above or equal to the applicable floor. As a result, no additional dependent combinations were required to be developed and the PRA dependency evaluation was found acceptable with no model changes required.	The resolution of the Peer Review finding validated adequacy of the dependency evaluation and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
HR-I3	DOCUMENT the key assumptions and key sources of uncertainty associated with the human reliability analysis.	SR not met. The Cooper PSA generally provides very detailed documentation, but, there is no discussion of sources of uncertainty regarding HRA consistent with the intent of SR HR-I3. Given the NRC sensitivity to the issue of sources of uncertainty (as evidenced by NRC Memorandum, "Notice of Clarification to Rev. 1 of Regulatory Guide 1.200", July 27, 2007, NRC ADAMS Accession number ML071170054), and the ASME Standard highlighting this specific issue in all Technical Elements, the intent of SR HR-I3 is judged not met by the current Cooper PSA documentation.	This finding has been addressed. Findings related to PRA base model uncertainty characterization have been resolved subsequent to the peer review. These peer review findings are identified as DA-E3-02, HR-13-01, IE-D3-01, IF-F3-01, and SC-C3-01. Resolution of the findings involved validation of the CNS PRA against applicable industry guidelines and required no changes to the PRA or the PRA documentation. Resolution was completed through validation that the guidance provided by NUREG-1855 and EPRI TR-1016737 was appropriately used to characterize the uncertainty relevant to the base CNS PRA. This guidance was in draft form during the Peer Review and therefore not available. The lack of final guidance on characterizing uncertainty in the base PRA model resulted in the CNS peer review findings in the supporting requirements subject to uncertainty evaluation.	The resolution of the Peer Review finding validated adequacy of documentation of the key assumptions and sources of uncertainty and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
IE-D3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the initiating event analysis.	SR not met. The requirement is to document the key assumptions and key sources of uncertainty. The assumptions used for initiating events were scattered throughout the document (CNS PSA-001). Uncertainty bounds were established, but sources of uncertainty were not discussed.	This finding has been addressed. Findings related to PRA base model uncertainty characterization have been resolved subsequent to the Peer Review. These peer review findings are identified as DA-E3-02, HR-13-01, IE-D3-01, IF-F3-01, and SC-C3-01. Resolution of the findings involved validation of the CNS PRA against applicable industry guidelines and required no changes to the PRA or the PRA documentation. Resolution was completed through validation that the guidance provided by NUREG-1855 and EPRI TR-1016737 was appropriately used to characterize the uncertainty relevant to the base CNS PRA. This guidance was in draft form during the peer review and therefore not available. The lack of final guidance on characterizing uncertainty in the base PRA model resulted in the CNS peer review findings in the supporting requirements subject to uncertainty evaluation.	The resolution of the Peer Review finding validated adequacy of documentation of the key assumptions and sources of uncertainty and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
IF-B2	For each potential source of flooding, IDENTIFY the flooding mechanisms that would result in a fluid release. INCLUDE: (a) failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc. (b) human-induced mechanisms that could lead to overfilling tanks, diversion of flow through openings created to perform maintenance; inadvertent actuation of fire suppression system (c) other events resulting in a release into the flood area	SR not met. PSA-012 Appendix E identifies failure modes of pipes and components for each source. The components are not specifically identified, but are included in the totals. The only failure mode of components identified is rupture. Other failure modes are not discussed. Human induced floods are dismissed in section 2.2.9.1. The main argument is that the generic pipe rupture frequencies already included these types of failures. This seems reasonable for pipe and component ruptures; however it does not include other types of spill scenarios (such as tank overfills). It is likely that these types of releases can also be screened due to alarms or other process parameters, but it is not in the documentation. Maintenance induced spills are dismissed in part by saying that personnel are available to detect the spill because they are the ones doing maintenance. This probably covers most maintenance activities, but it is not necessarily true for operational events that are performed remotely.	This finding has been addressed. Subsequent to the PRA Peer Review, a supplemental evaluation was performed to estimate the potential contribution for human induced flooding. This evaluation concluded that the contribution of internal flooding events due to maintenance errors is quantitatively included in existing PRA quantifications.	The resolution of the Peer Review finding validated adequacy of identification of potential flooding sources and did not result in changes to the PRA model. Also, internal flooding events and modeling thereof do not impact fire risk. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
IF-F3	DOCUMENT the key assumptions and key sources of uncertainty associated with the internal flooding analysis.	SR not met. Significant sources of uncertainty have been identified. However, there is a lack of treatment in the uncertainty analysis regarding modeling assumptions and structure. For internal flooding, this may involve the assumptions regarding doors terminating flood propagation or varying the flood target population. For these reasons this SR is "Not Met".	This finding has been addressed. Findings related to PRA base model uncertainty characterization have been resolved subsequent to the peer review. These peer review findings are identified as DA-E3-02, HR-13-01, IE-D3-01, IF-F3-01, and SC-C3-01. Resolution of the findings involved validation of the CNS PRA against applicable industry guidelines and required no changes to the PRA or the PRA documentation. Resolution was completed through validation that the guidance provided by NUREG-1855 and EPRI TR-1016737 was appropriately used to characterize the uncertainty relevant to the base CNS PRA. This guidance was in draft form during the Peer Review and therefore not available. The lack of final guidance on characterizing uncertainty in the base PRA model resulted in the CNS Peer Review findings in the supporting requirements subject to uncertainty evaluation.	The resolution of the Peer Review finding validated adequacy of documentation of the key assumptions and sources of uncertainty and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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LE-G5	IDENTIFY limitations in the LERF analysis that would impact applications.	SR not met. There is a requirement to discuss limitations in the LERF analysis that would impact applications. This was performed in the Level 1 analysis, but there is no evidence in the Level 2 analysis of a limitations discussion.	This finding has been addressed. Assumptions associated with the containment event trees (CET) are summarized by the individual CET node in the applicable Appendix C section of the PRA Summary Notebook. These assumptions introduce the uncertainties that apply to the use of the Level 2 analysis. Furthermore, there are no limitations that would impact projected applications that are not identified as part of the uncertainty evaluation in the PRA Summary Notebook (Appendices A, B, E). Accordingly, no changes were made to the model.	The resolution of the Peer Review finding validated adequacy of the identification of limitations in the LERF analysis and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.
MU-B2	PRA Configuration Control - Changes that would impact risk-informed decisions should be prioritized to ensure that the most significant changes are incorporated as soon as practical.	SR not met. CNS Procedure ESPD-13 (PSA Model Maintenance and Update Procedure) mentions examples of some applications that need to be addressed or updated. However, there is no discussion of prioritization or urgency. Prioritization seems to be focused on base model and future applications rather than past applications and risk-informed decisions. Also, a timetable should be established to state when the application impacts need to be incorporated (e.g., six months after the PSA update is officially released).	This finding has been addressed. Procedure (ESPD-13) was re-written to detail a new model maintenance and update process. The new process meets the requirements of the latest ASME PRA standard.	The resolution of the Peer Review finding validated adequacy of the revised model and maintenance update process and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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MU-E1	The PRA configuration control process shall include a process for maintaining control of computer codes used to support PRA quantification.	SR not met. Software (including versions) should be specifically in SQA program, and the versions used should be consistent with the SQA program.	This finding has been addressed. The PRA configuration control procedure was revised to ensure that the software control procedures are now used to control PRA software.	The resolution of the Peer Review finding validated adequacy of the revised PRA configuration control process and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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MU-F1	The PRA configuration control process shall be documented. Documentation typically includes: (a) Description of the process used to monitor PRA inputs and collect new information (b) Evidence that the aforementioned process is active (c) Descriptions of proposed changes (d) Descriptions of changes in PRA due to each Update or Upgrade (e) Record of the performance and result of the appropriate PRA reviews (f) Record of the process and results used to address the cumulative impact of pending changes (g) Record of the process and results used to evaluate changes on previously implemented risk-informed decisions (pursuant to MU-D1) (h) Description of the process used to maintain software configuration control.	SR not met. Process exists to review past PRA applications and determine if an update to the risk informed application is required when the PRA model is updated. Cannot see evidence that the aforementioned process is active. List of applications is not up-to-date.	This finding has been addressed. The PRA configuration control procedure was revised to ensure that past PRA applications are identified and reviewed to ensure updates to the application are required.	The resolution of the Peer Review finding validated adequacy of the revised PRA configuration control process and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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QU-F5	DOCUMENT limitations in the quantification process that would impact applications.	SR not met. No discussion of limitations for applications in the documentation. A limitation is that not including IE fault trees in the main model yields incorrect importance measures for events/components in the IE fault tree. (QU-F5-01)	This finding has been addressed. Tracking Item # 2009-008 was entered into the PRA modification requests data base to revise the PRA Maintenance and Update desk top guide and application-specific guidelines to ensure limitations in the quantification are documented.	The NFPA 805 application assessed the fact that initiating event (IE) fault trees are not integrated into the main model and concluded that this had no impact on the modeling of fire risk. The IE fault trees discussed by this finding are not related to fire events and are not required for the NFPA 805 application. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.
IE-A7	REVIEW plant-specific operating experience for initiating event precursors, for the purpose of identifying additional initiating events. For example, plant specific experience with intake structure clogging might indicate that loss of intake structures should be identified as a potential initiating event.	Capability Category I met. There is no evidence in the notebook that a precursor review was performed. Response to the question was Table 2.3-3 and LER review performed. The items in the table were all plant scrams. The LER review would contain non-scram precursors. However, a question was asked to the CNS team, and the response pointed back to the support system initiator development, which is covered by another SR. This SR of Category I (no requirement for precursor review).	This finding has been addressed. There was an extensive plant-specific review of operational experience to identify precursors. The systematic search for plant-unique and plant-specific support system initiators is documented in Section 2 of the initiating event notebook. The search for precursors included the interview of the system managers, operators, and a review of Cooper LERs. The Cooper Initiating Event notebook provides a detailed review of Cooper specific design and the identification of IE precursors; see Section 2.3 of CNS PSA-001. Each CNS specific system/subsystem) potential IE impact is discussed; for example, loss of steam tunnel and/or turbine building HVAC is discussed in Sections 2.3.3.20.1 and 2.3.3.20.2 as potential IE precursors as well as how each is handled.	The resolution of the Peer Review finding validated adequacy of the initiating event analysis and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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			In addition, the PRA industry has exhaustively identified initiating event categories in countless IE studies over the past 30 years. Further, other SRs (e.g., IE-B3) already require individual support systems to be reviewed as potential initiating events. This is documented in Section 2 of the Initiating Event Notebook. In addition, the loss of intake initiating event was extensively studied. See Initiating Event Notebook Section 2 and Appendices C and J.	
QU-E3	ESTIMATE the uncertainty interval of the overall CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G9, IEC13), taking into account the "state-of-knowledge" correlation.	Capability Category I met. This is described in Appendix A of the Quantification Notebook. Type-code database deals with "state of knowledge" correlation. But, for many components with plant-specific data, even though there are multiple identical components with the same failure probability, type codes were not used. This means that the "state-of-knowledge" correlation is not correctly taken into account. Therefore, only Category 1 is met.	This finding has been addressed. The UNCERT database differs from the master basic event database. This is due primarily to the use of initial HEP values in the master basic event database set to higher values to ensure that low frequency cutsets are adequately "drawn in" to the cutsets, i.e., avoids premature truncation. These HEPs are subsequently reset to their nominal values by QRecover. As a result, the master basic event database does not reflect the basic events, distributions, and uncertainties used in the UNCERT model. Therefore, the master basic event database was not used for the UNCERT evaluation, but a modified database was used that includes: - Error factors and distributions for the	The resolution of the Peer Review finding validated adequacy of the uncertainty analysis and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
			dependent operator actions. - The "real" values of some basic events that are set to screening values and then replaced by the recovery file post quantification. This modified database for UNCERT was available and was used in the Cooper uncertainty calculations reported in the PRA, however, it was unintentionally omitted from the documents provided to the PRA Peer Review Team as part of their evaluation. It is concluded that, had this database been available for the peer team, the Capability Category II SR would have been met. There is no impact on applications or the base model.	
AS-A2	For each modeled initiating event, IDENTIFY the key safety functions that are necessary to reach a safe, stable state and prevent core damage.	Capability Category I/II/III met. Some event sequences are terminated when core damage has not occurred within 24 hours. In some, a stable state has not been reached based on the associated supporting thermal hydraulic calculation. For example, in 1A-L1-HPCI (which supports sequence GTR-002), containment temperature is still increasing at the end of 24 hours and has reached 250 °F. The operators will be directed to depressurize when temp reaches 280 °F. Another node appears to be needed in the tree to get to a stable	This finding has been addressed. The Peer Review characterization of CNS adherence to SR AS-A9 is documented to be "Met CC III", i.e. CNS use(s) realistic, plant specific thermal hydraulic analysis to determine the accident progression parameters (e.g., timing, temperature, pressure, steam) that could potentially affect operability of the mitigating systems". Additionally, the Peer Review characterized CNS adherence to SR AS-A2 as "Met" with documented significance of the finding being "The Class II sequences (which are known to be long term sequences) appear to have	The resolution of the Peer Review finding did not result in changes to the PRA model and hence the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
		state.	been evaluated appropriately". CNS review of the results of the referenced thermal-hydraulic calculation (1A-L1-HPCI) indicate that wet-well is trending downward starting at the 10th hour form the start of the postulated accident scenario, and drywell temperature peaking and starting to trend down at the 22nd hour. Hence it is not expected to see containment temperature to reach 280 °F. Based on the high marks received from the Peer Review on both SRs AS-A2 and AS-A9, the low significance of the finding as provided by the peer reviewers, and reexamination of the results of the example provided (i.e., MAAP run showed declining trends) , CNS sees no merit to further analysis.	

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SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
SC-B3	When defining success criteria, USE thermal/ hydraulic, structural, or other analyses/evaluations appropriate to the event being analyzed, and accounting for a level of detail consistent with the initiating event grouping (HLR-IE-B) and accident sequence modeling (HLR-AS-A and HLR-AS-B).	Capability Category I/II/III met. Appendix F of PSA-003 dismisses the need for long term core spray in large LOCA scenarios based on MAAP calculations. While consistent with existing PRAs, this needs to be addressed further. MAAP does not treat steaming in the low power bundles precisely. It is OK if recovery is imminent or if the core is going to a melt state, however for long term steady state at low water level it will over-predict the two phase level in the low power bundles. MAAP calculates an overall steaming rate and applies it evenly across all bundles. This provides an adequate collapsed level in each bundle, but the two-phase will be too high in the low power bundles. MAAP also does not behave as expected when calculating the individual node core power. Due to the way it handles the uranium group, the power shape calculated is flatter than expected. This could affect the two phase level as well.	This finding has been addressed. GE calculations are the basis for the success criteria – not relying solely on MAAP calculations. The success criteria that do not require core spray for large LOCA mitigation are based primarily upon GE calculations (NEDO 24708A, OG00-0170-062, and DRF-E22-00135-01). The Success Criteria Notebook in Appendix F identifies that the DBA calculations by GE do not show fuel or clad melting for the identified cases in question. Rather, the GE design calculations show that 10CFR50 App. K requirements for a DBA of < 17% clad oxidation cannot be assured. However, this is not a criterion for core damage as specified in the ASME PRA Standard or in the Cooper PRA. Therefore, these criteria do not need to be satisfied to allow success in the Level 1 PRA. The Cooper success criteria are consistent with all BWR PRAs reviewed under the BWROG Certification Program and NUREG-1150. The CNS success criteria are consistent with all boiling water reactor (BWR) PRAs reviewed under the BWROG certification program and NUREG-1150.	The resolution of the Peer Review finding determined that both GE analysis and MAAP were utilized in development of success criteria. Thus, no changes to the PRA model were required, and the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
SC-C3	Document the key assumptions and key sources of uncertainty associated with the development of success criteria.	Capability Category I/II/III met. It is possible that the success criteria uncertainty is addressed implicitly in the other elements; however the treatment of the uncertainty is characterized as an increase or decrease in reliability. Changes to success criteria would be a logic change and is more difficult to deal with in sensitivity analyses.	This finding has been addressed. Findings related to PRA base model uncertainty characterization have been resolved subsequent to the Peer Review. These Peer Review findings are identified as DA-E3-02, HR-13-01, IE-D3-01, IF-F3-01, and SC-C3-01. Resolution of the findings involved validation of the CNS PRA against applicable industry guidelines and required no changes to the PRA or the PRA documentation. Resolution was completed through validation that the guidance provided by NUREG-1 855 and EPRI TR-1016737 was appropriately used to characterize the uncertainty relevant to the base CNS. PRA. This guidance was in draft form during the peer review and therefore not available. The lack of final guidance on characterizing uncertainty in the base PRA model resulted in the CNS peer review findings in the supporting requirements subject to uncertainty evaluation.	The resolution of the Peer Review finding validated adequacy of documentation of the key assumptions and sources of uncertainty and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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SY-A4	PERFORM plant walkdowns and interviews with system engineers and plant operators to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	Capability Category II/III met. IPE system notebook and internal flooding walkdowns are listed in the self-assessment as references for SR-A4. However, the IPE walkdowns were not recently performed and the internal flooding walkdowns were performed with different goals in mind. Also, operator interviews were conducted for the HRA analysis and accident sequence modeling, but were not performed for the system analysis and documented.	This finding has been addressed. There is no requirement in SR SY-A4 that system walkdowns be performed at each update. Numerous system walkdowns have been performed over the years in support of the CNS PRA and its applications. The value of the walkdowns and information gained must be balanced against the dose received performing the walkdowns. Some areas are inaccessible for routine walkdowns. The level of effort required to continually perform new walkdowns is inconsistent with the usefulness of such effort. Accordingly, no changes were made to the model.	The resolution of the Peer Review finding did not result in changes to the PRA model and hence the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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SY-A14	In meeting SY-A12 and SY-A13, contributors to system unavailability and unreliability (i.e., components and specific failure modes) may be excluded from the model if one of the following screening criteria is met: (a) A component may be excluded from the system model if the total failure probability of the component failure modes resulting in the same effect on system operation is at least two orders of magnitude lower than the highest failure probability of the other components in the same system train that results in the same effect on system operation. (b) One or more failure modes for a component may be excluded from the systems model if the contribution of them to the total failure rate or probability is less than 1% of the total failure rate or probability for that component, when their effects on system operation are the same.	Capability Category I/II/III met. SY-A13 imposes the requirement to explicitly include in the model failure modes such as "Fails to Remain Open/Closed". SA-A14 provides criteria for excluding these failures modes. The failure modes were generally excluded from the CNS system fault trees, but no documented assessment of criteria in SY-A14 was found.	The CNS PRA did use criteria detailed by this SR. Appendix B of the CNS PSA-010, Component Data Notebook, documents use of the requirements of SR SY-A15. These criteria were used in scoping component failure events for the PRA. This finding highlights the need to document the assessments of the criteria. PRA tracking Item # 2009-009 was entered into the PRA modification requests data base to document the basis for the exclusion of low probability failure modes, as part of the next PRA update.	The resolution of the Peer Review finding represents a documentation improvement for the model and hence the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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DA-E1	DOCUMENT the data analysis in a manner that facilitates PRA applications, upgrades, and peer review.	Capability Category I/II/III met. The Maintenance Rule program processes and procedures and other plant data sources are relied upon to meet many of the aspects of the Data Analysis section. The processes that are used to screen and incorporate data collected in the Maintenance Rule program into the PRA plant specific data used in the model are not found in the PRA documentation.	This finding has been addressed. The basis for significance of this item is that the documentation doesn't facilitate peer review. Although addressing this item will help facilitate peer review, documentation issues such as this should be considered suggestions rather than findings. This documentation issue does not impact the technical adequacy of the PRA or its capability. PRA notebook CNS PSA-010, Component Data notebook provides the documentation for data analysis to ensure data bases are provided for users of the PRA. Use of maintenance rule data to assist in PRA data analysis are considered to be part of the skills sets for a PRA practitioner and not required to be documented as a process.	The resolution of the Peer Review finding did not result in changes to the PRA model and is representative of insights into the area of documentation of this SR. Hence the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

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DA-E3	DOCUMENT the key assumptions and key sources of uncertainty associated with the data analysis.	Capability Category I/II/III met. A source of uncertainty not treated relates to failure modeling for certain equipment failure modes. A normally open MOV, which is required to remain open, can spuriously close. This can happen during the 24-hr mission time after the initiator. It is also possible that it could happen during plant operation and not be detected. This should be discussed for all equipment failure modes that are modeled and which are subject to failure either prior to the initiator or after the initiator. Even though one or the other contribution to failure may be evaluated to be negligible, this should be addressed. The negligible contributor still makes a contribution to uncertainty.	This finding has been addressed. Findings related to PRA base model uncertainty characterization have been resolved subsequent to the Peer Review. These Peer Review findings are identified as DA-E3-02, HR-I3-01, IE-D3-01, IF-F3-01, and SC-C3-01. Resolution of the findings involved validation of the CNS PRA against applicable industry guidelines and required no changes to the PRA or the PRA documentation. Resolution was completed through validation that the guidance provided by the NUREG-1855 (issued subsequent to the peer review) and EPRI TR-1016737 was appropriately used to characterize the uncertainty relevant to the base CNS PRA. This guidance was in draft form during the peer review process and therefore not available. The lack of final guidance on characterizing uncertainty in the base PRA model resulted in the CNS Peer Review documenting findings in the supporting requirements subject to uncertainty evaluation.	The resolution of the Peer Review finding validated adequacy of documentation of the key assumptions and sources of uncertainty and did not result in changes to the PRA model. Hence, the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
HR-G6	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.	Capability Category I/II/III met. The HRA documentation mentions that the resulting HEPs were reviewed against each other. However, it isn't clear how this was done. For example, the HEP for the operator action to initiate drywell spray (RHR-XHE-FO-SPRAY) is about a factor of 3 times lower than the action to initiate torus cooling (RHR-XHE-FO-RHRE & RHR-XHEFO-RHRL). It's not obvious why this is logical given that the action to align torus cooling should be one of the most reliable actions given that it is performed fairly routinely (i.e., following any plant trip or manual shutdown). Also, according to the HRA calculator worksheets for these HFEs, the time available to align torus cooling is 886 minutes, while it's only 290 minutes for drywell spray. In addition, the operator action to perform emergency depressurization (ADS-XHE-FOTRANS) has a slightly lower HEP than the action to perform torus cooling, even though the time available for performing emergency depressurization is less than 30 minutes and under higher stress conditions.	This finding has been addressed. It is true that the initial HRA quantification of HEPs was performed by multiple analysts. However, once the initial analysis was completed, the results were independently reviewed by a senior analysis and review insights were incorporated into each of the HEP analysis. Subsequently, the final HEPs were developed and reviewed again for consistency and reasonableness. Each of the specific examples identified by this finding were reviewed and the HEP values used by the PRA were found consistent, reasonable and acceptable for use.	The resolution of the Peer Review finding did not result in changes to the PRA modeling of HEP and events. Hence the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
IF-B1	For each flood area, IDENTIFY the potential sources of flooding [Note (1)]. INCLUDE: (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, component cooling water system, feedwater system, condensate and steam systems) (b) plant internal sources of flooding (e.g., tanks or pools) located in the flood area (c) plant external sources of flooding (e.g., reservoirs or rivers) that are connected to the area through some system or structure (d) in-leakage from other flood areas (e.g., back flow through drains, doorways, etc.).	Capability Category I/II/III met. Components evaluated as flood initiators should be specifically identified in PSA-012 Appendix E. The components should be identified similar to the way that the pipes are.	This finding has been addressed. The equipment that could be sources of flooding in each area are identified in the Walkdown Sheets located in the Internal Flood Walkdown Notebook. Identification in the walkdown sheets and inclusion in the internal flood walkdown notebook documents that components were evaluated and included as required.	The resolution of the Peer Review finding did not result in changes to the PRA model and hence the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

Table U-1 Internal Events PRA Peer Review – Facts and Observations

SR	Category II Requirements	CNS Peer Review Facts and Observations Summary	CNS Disposition	Expected Impact on NFPA 805 Application
QU-D4	REVIEW a sampling of no significant accident cutsets or sequences to determine they are reasonable and have physical meaning.	Capability Category I/II/III met. Could not find evidence of review of non-significant cutsets to determine if they are reasonable. Documentation is available that shows review of high level cutsets (top 100-200).	This finding has been addressed. No documentation requirement exists in the ASME PRA Standard for this item. It appears that the review team is interpreting that detailed write-ups are required for every detailed step taken in the development of the PRA. The subject review of non-significant cutsets was performed multiple times during draft quantifications of the model, as well as the final documented dominant sequence and cutset discussions in the PRA Summary Notebook. The PRA does not maintain hand mark-ups of draft quantifications and associated fixes. The peer review team recommendation for detailed write-ups of more cutsets and sequences, and write-ups of hand mark-ups and corrections in draft quantifications is judged by NPPD to be beyond the intent of this SR. Reviews done by the PRA modelers during development and quantification is deemed adequate for this SR.	The resolution of the Peer Review finding did not result in changes to the PRA model and hence the finding has no impact on use of the CNS Internal Events PRA in the NFPA 805 application.

ATTACHMENT V

Fire PRA Quality

60 Pages

V.1 Overview

The Fire PRA is adequate to support the NFPA 805 Licensing Basis. A Peer Review was conducted during April 2010. A follow-on focused peer review was conducted in February 2011 to specifically address multi-compartment analyses and final results quantification. The Peer Review noted a number of facts and observations (F&O). The F&O and the disposition of the F&O are provided in Table V-1. F&O relevant to this application have been addressed.

The Fire PRA meets Capability Category (CC) II in most, but not all cases. There were 49 SR identified that did not meet Capability Category II. The impact of those areas where only the Capability Category I requirement was met was evaluated and is summarized in Table V-2.

V.2 Unreviewed Analysis Methods

The CNS Fire PRA did not use unreviewed analysis methods. However, the generic treatments of transient fires for heat release rate and transient-free zone fire frequencies were refined. In addition, incipient detection was modeled consistent with FAQ 08-0046. However, rather than modeling the increased potential for suppressing the fire, the analysis only modeled "early detection" and then applied human reliability analyses to model operator response to the early detection.

Transient Fire Frequency: Fire Zone 9A (Cable Spreading Room) and Fire Zone 8A (Auxiliary Relay Room) will be designated as enhanced transient and hot work controlled fire zones (see Implementation Item S-3.25 of Attachment S, Table S-3. The Fire Ignition Frequency Calculation utilized the relative ranking (ranking scheme developed in NUREG/CR-6850) to assign weighting factors for ignition frequency bins involving transient combustibles or activities. Occupancy level, storage of flammable materials, and type and frequency of maintenance activities in a zone are the three most important influencing factors of the likelihood of fire ignition involving a transient combustible or activity. For these fire zones a "Very Low" ranking was utilized to reflect a significantly lower-than-average level of the factor. A very low maintenance factor (0.05) was applied to these fire zones as maintenance and hot work are strictly controlled during at-power operation. A very low storage factor (0.1) was applied, as the storage of transient materials is also strictly controlled. These very low factors are not specified in the NUREG/CR-6850 methodology. However, since the administrative restrictions that will be put in place at CNS significantly reduce the likelihood of fire ignition in these fire zones, and since the likelihood that these administrative restrictions are violated is judged to be very small considering how these restrictions are applied at CNS, it is concluded that the very low factors are appropriate for use in the CNS Fire PRA analysis.

For three specific locations within fire zones (Fire Zone 3C and 3D, the area located above the TIP Room on the 903'-6" Elevation of the Reactor Building, and Fire Zone 2C, the floor areas immediately located around Instrument Racks 25-5 and 25-6 on the 931'-6" Elevation of the Reactor Building), enhanced transient controlled fire ignition frequencies were refined as follows:

1. For these three enhanced transient controlled locations (i.e. transient combustibles are strictly controlled during plant operation), the transient bin frequency (bins 7, 25, and 37) was reduced by a 0.1 multiplier. This is the screening human error probability (HEP) value for violating plant procedures due to human error.
2. A procedural restriction on cutting, welding and grinding at power in these enhanced transient controlled locations (i.e., strictly controlled during plant operation) is being established; therefore, the transient bin frequency (bins 6, 24, and 36) was reduced by a 0.1 multiplier. This is the screening HEP value for violating plant procedures due to human error.

Transient Fire Heat Release Rate (HRR): Fire Zone 9A (Cable Spreading Room) and Fire Zone 8A (Auxiliary Relay Room) have been designated as enhanced transient and hot work controlled fire zones, and detailed fire modeling in these zones has utilized a heat release rate for transient fire scenarios based on transient combustibles in these zones being minimal based on administrative restrictions. Therefore, a reduction in the transient fire size from the 317 kW (98^{th} percentile fire size) recommended by NUREG/CR-6850 has been justified in these fire zones and transient fires have been characterized as a 69 kW fire. Transients caused by welding and cutting are not expected in these fire zones as they are considered "Enhanced Transient Controlled" and "Enhanced Hot Work Controlled" locations, and welding and cutting are strictly controlled in these zones while the plant is at power.

Incipient Detection: Based on insights from analyses for two panels (9-32 and 9-33) in the Auxiliary Relay Room (part of fire area CB-D), an incipient detection system will be installed (see Implementation Item S-2.4 of Attachment S, Table S-2). The purpose of the system is to provide early indication of the potential for a fire inside one of these panels. Very Early Warning Fire Detection Systems (VEWFDS) can be credited for electrical/electronic components with a voltage of less than or equal to 250VDC or 480 VAC and that contain internal components that exhibit gradual degradation.

The VEWFDS will provide indication in the Control Room so that an operator/auxiliary operator can respond to the Auxiliary Relay Room, confirm that the incipient detector for one of these panels has activated, and inform the Control Room. The Control Room operators can then respond in the Control Room using procedures for these panels. Given this type of scenario, a few long-term RA will remain.

Absent the incipient detection, there is an increased potential for the Control Room operators to implement alternate shutdown (ASD), as a fire in either of these panels would impact automatic operation of core cooling equipment and Control Room instrumentation. In addition to installation of an incipient detection system, fire response procedures will be changed such that the Control Room will be the command and control center for reaching safe and stable conditions (see Implementation Item S-3.3 of Attachment S, Table S-3).

In summary, the modeling follows FAQ 08-0046, except that instead of increased potential for suppressing a fire, the early indication of a potential fire was used as a cue to support improved operator response. The modeling is as follows:

1. Incipient detection works AND Operators respond to alert AND Operators successfully respond from the Control Room with some local, long-term recovery actions (combined probability of 0.99*0.99*0.99 = 0.97) AND conditional core damage probability (CCDP)/conditional large early release probability (CLERP) due to equipment failures (0.01), which results in 0.97*0.01 = 0.0097 PLUS

2. Incipient detection fails OR Operators fail to respond to alert OR Operators fail to respond from the Control Room) (combined probability of 0.01+0.01+0.01 = 0.03) AND CCDP/CLERP using the ASD path (0.1), which results in 0.03*0.1 = 0.003.

3. Thus the total CCDP/CLERP is 0.0097 + 0.003 = 0.0127 versus a CCDP/CLERP of 0.1 if only ASD is modeled.

Note 1: Failure probability for incipient detection is conservatively assumed to be 0.01.

Note 2: HEP for operators failing to respond to indication used a screening value of 0.01. At least 1 hour would be available.

Note 3: HEP for operators failing to respond from the Control Room used a screening value of 0.01. Fire response procedures will be changed, as short and intermediate term actions can be accomplished from the Control Room (see Implementation Item S-3.3 of Attachment S, Table S-3).

Note 4: CCDP/CLERP (equipment failures) for response from the Control Room used a screening value of 0.01.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
PP-C3	PP – Plant Partitioning	Closed	1-14: Unique features are described and the basis for crediting as a partition boundary are provided using reference to site calculations, analyses, engineering evaluations, etc. Conclusions of these documents are not provided, however. Suggestion: provide a brief summary of what the document says about the feature. One feature of interest is the Promat boundary protecting the conduit in Fire Zone 9A. Additional details should include the estimated fire resistance, the resistance to mechanical damage from a high energy arcing fault, and the maximum transient fire size the boundary protects the conduit from.	Calculation NEDC 10-021, "Embedded Conduit, Concrete Cable Enclosures, and Promat-H Board Review," and the detailed fire modeling reports review and document unique features credited in the analysis.
PP-B2	PP – Plant Partitioning	Closed	4-1: The justification of partitioning between fire zone barriers to prevent full room burn-up not provided in some areas. These include 8H/8E, 8F/8E, 11A/11B, and 14A/14C.	Calculation NEDC 10-004, "Plant Boundary Definition and Partitioning," was updated to address fire compartments with multiple fire zones in which the fire zone boundaries may have been credited and whole room burnout approach taken to "screen" fire zones from detailed fire modeling. A review of the boundaries of the "screened" fire zones was performed via plant walkdown in order to confirm that these barriers are substantial enough to preclude fire spread to adjacent fire zones within the fire compartment. Detailed assessment of these barriers is provided in Calculation NEDC 10-024, "Multi-Compartment Analysis."
ES-D1	ES – Equipment Selection	Closed	1-15: NEDC 09-078 and 09-080 adequately document the ES process. Reference 14 of NEDC 09-78 needs to be an [a] controlled reference. It can be a controlled piece of software or the data extracted and controlled. NFPA 805	The Fire PRA database will be controlled as an electronic document in the same way the Internal Events PRA model (CAFTA model) is controlled. See Implementation Item S-3.24 of Attachment S, Table S-3.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			requires that all referenced items be originated (i.e. signature), reviewed, and readily retrievable. This same database CNSFire.mdb is reference 15 of the PRM (NEDC 09-079). As the PRM impacts can be reproduced without this mdb, the same issue applies to PRM.	
CS-A7	CS – Cable Selection	Closed	5-2: It has not been verified that Kerite cable used in CNS should be treated as thermoset.	Kerite has been treated as thermoset material with respect to flame spread and the yielding heat release rate. Preliminary testing of Kerite cable and insulation presented in the Sandia National Laboratory Report SAND2010-4936, "A Preliminary Look at the Fire-Induced Electrical Failure Behavior of Kerite FR Insulated Cables," suggests that a temperature of 250°C should be used for the lower-bound estimate of Kerite FR cable failure threshold. This value has been utilized for determining the damaging heat release rate in a sensitivity study in NEDC 09-85, "Task 7.14 Fire Risk Quantification." Because this preliminary report does not contain information on a damaging heat flux value, the thermoplastic damaging heat flux threshold has been used.
SY-C3	PRM – Plant Response Model	Closed	1-21: Significant system model changes have been implemented. However, the documentation is not sufficient for review with respect to the pertinent information listed in SR SY-A2 and other SY SRs. Additionally, plant specific uncertainty due to PRA model changes is not discussed in Table 5 of the uncertainty analysis.	In NEDC 09-079, "Task 7.5 Fire-Induced Risk Model," Section 4.2.4, Systems Analysis, an item was added for the random failure of an in-series check valve that would cause system failure if a simultaneous spurious open of the MOV/AOV providing isolation were to occur. Attachment D, "New Components and New Random Failure Modes Added to Support the Fire PRA," was added. Table D-1 documents the new components and/or new random failure modes added to support the Fire PRA. Table D-2 describes

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
				how the specific SY and/or DA SLRs (Supporting Level Requirements) are met and documented. Uncertainty: The systems analyses on which the peer review was conducted are documented in NEDC 09-079, "Task 7.5 Fire-Induced Risk Model." Table 5 of the uncertainty and sensitivity calculation, NEDC 09-086, "Task 7.15 Uncertainty and Sensitivity," provides a summary of assumptions and uncertainties from the NEDC 09-079 calculation. A revision to the NEDC 09-079 calculation addresses F&O associated with this task. Included in NEDC 09-079 is an expanded discussion on assumptions and uncertainties to explicitly address SR SY-C3.
SY-A2	PRM – Plant Response Model	Closed	2-3: CNS PRM report NEDC 09-079 Rev. 0 Section 4.2.4 includes some discussion on system models. The internal events PRA fault trees were modified to include the failure modes caused by fires and to add ISLOCA pathways, and IORVs caused by spurious component operation. In addition, the fault trees, while sufficient for use in the internal events PRA, were enhanced to ensure they were sufficient to meet the needs of the fire PRA as well. Significant system model changes have been implemented. However, the documentation is not sufficient for review with respect to the pertinent information listed in SR SY-A2 and other SY SRs. Moreover, the feedwater system model is significantly enhanced in the fire model, which would be appropriate to update the internal events system model to capture all the updated information. The following	The purpose of NEDC 09-079, "Task 7.5 Fire-Induced Risk Model," was to develop a risk logic model to enable identification and quantification of all CDF and LERF sequences that could result from a fire initiating event. The internal events model was used as the foundation for the Fire PRA model. The Fire PRA model includes fire-induced impacts on the systems, trains, and components modeled in the internal events fault tree logic. Fire PRA modeling was performed in a manner that maintains the integrity of the internal events fault tree model and allows for proper evaluation of both the Internal Events PRA and the Fire PRA. The three issues of FINDING 2-3 are addressed below: 1. Fire PRA modeling was performed in a manner that maintains the integrity of the internal events fault tree model and allows for proper evaluation of both the Internal Events PRA and the Fire PRA. To evaluate internal events, the fire initiators are all set to the default value of 0.0, and fire events are set to

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			includes additional details for the identified issue. 1. The system boundary may be changed due to the updates to the system models in fire PRA, such as the feedwater system model. The addition of instruments may also require the updates to the system boundaries. However, such documentation and model changes are not evident. 2. Currently the failure modes identified for fire PRA models are not included in the system notebooks. SY-A14 and A15 are considered met since a review verified that substantial model changes have been performed in fire PRA to identify all failure modes. 3. The new HFEs for fire model are addressed in the HRA report. However, their incorporation into the system models is not documented (SY-A17 requirement).	either 0.0 or 1.0 depending on how they input to their parent gate. For example, if a fire event is under an "OR" gate, it would be set to 0.0 for the internal events cases. PRAQuant solves each fire scenario by setting all internal events initiators to 0.0 and setting fire initiators and those basic events representing components impacted by the fire to 1.0. As such, Fire PRA modeling does not represent a change to the internal events modeling and appropriately is not included in the internal events documentation (e.g., system notebooks). Fire-induced failures (or new components) and supporting fault tree logic such as a check-valve internal leakage (ISLOCA logic), spurious operations, and/or new failure modes not included in the internal events modeling are documented in NEDC 09-079, "Task 7.5 Fire-Induced Risk Model," Attachment D "New Components and New Random Failure Modes Added to Support Fire PRA." 2. Although the Fire PRA builds off the internal events model, the logic used for fire modeling is documented in the Fire PRA calculations and is not required by the Standard to be included with the internal events documentation (e.g., system notebooks). 3. Human failure events added to the Fire PRA model to represent post-fire human actions are not required to become part of the Internal Events PRA documentation. The post-fire human reliability analysis considers operator actions (human failure events) as needed for safe-shutdown, including those called out in relevant fire response procedures. The post-fire human actions added to the Fire PRA model include the human performance shaping factors associated with a fire and are not applicable to an internal events non-fire initiator.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
PRM-B12	PRM – Plant Response Model	Closed	2-14: The majority of component failure probability values, excluding spurious operation, do not need reanalysis for the Fire PRA. However, a query of BE table in the cnsone.rr database shows that several events with a style of 'fire' may be added in the fire models and require DA reanalysis: CRD-SOV-CCSO140A, CRD-SOV-CC-SO140B, LCS-CKV-LK-18CV, LCS-CKV-LK-19CV, RCI-CKV-LK-18CV, RCICKV-LK-19CV, RHR-CKV-LK-26CV, RHR-CKV-LK-27CV. As a result, PRM-B13 is considered not met. Check valve leakage can be uncovered through surveillance. See SR DA-C9.	Added Attachment D to NEDC 09-079, "Task 7.5 Fire Fire-Induced Risk Model," to document new component and new random failure modes that were added to support the Fire PRA. The branches in the model that contain these basic events require a fire for the logic to propagate upward in the tree. Language was added to the report describing how data for these basic events is determined. Specifically, the TYPE CODE (i.e., SOV CC and CKV LK) for both failure modes was used to import data from the existing CAFTA parameter file. Therefore, no new data analysis was necessary. Note: Initially this F&O included basic events EAC-DG1-OVERLOAD and EAC-DG2-OVERLOAD which are fire-induced scenarios that cause the Diesel Generators to overload. These basic events do not represent a specific component failure, but rather they are ANDed with a combination gate that requires two or more breakers to spuriously close. The peer review team members concurred these basic events should be removed from F&O 2-14, as they are not explicitly defined as components. Review of the final Peer Review Report shows they were removed from the F&O; however, they remain listed within the discussions for PRM-B12 and B13.
PRM-B9	PRM – Plant Response Model	Closed	4-5: Failure during operation failure modes (fail to remain open, fail to remain closed, and similar) are excluded from the internal events model when a demand failure mode exists, without justification. As a result, the internal events Cooper model obtained a finding. For fire events, these failure modes are clearly important as new fire cable induced faults could be applicable. This is recognized in Reference 14. The events appear to general[ly] be added to fault tree	This F&O refers to a finding CNS received during the Internal Events Peer Review. Engineering Study PRA-ES096, "Dispositioning of CNS Findings and Observations," documents how the findings were dispositioned. The finding referenced reads, in part: "The failure modes [e.g., Fails to Remain Open/Closed] were generally excluded from the CNS fault trees, but no documented assessment of criteria in SY-A14 was found." CNS dispositioned this finding by stating

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			under the preview of fire system modeling. But the system notebooks have not been updated.	these failure modes are low probability events, and that PRA modification Tracking Item #2009-009 requires action to document the data at a later date. The Fire PRA model considers the impact of a fire on active and passive failures. For each active failure, a corresponding fire-induced failure is modeled. If a passive failure (i.e., a spurious operation) could place the component in an undesirable configuration, the passive failure is modeled as well. However, if a passive failure places a component in an acceptable configuration, modeling is not included because the spurious operation would be similar to modeling a "failure" as a "success." For example, for a valve Fails-To-Close in the internal events model, the fire model would have a corresponding Fails-To-Close Due-To-Fire basic event. There would not be a passive failure (Spurious Close) for this valve in this branch of the fault tree, as the success position is "closed." NEDC 09-078, "Task 7.2 Component Selection," documents components added to the Fire PRA model and their failure modes.
PRM-B5	PRM – Plant Response Model	Closed	4-7: Event ADS-SRV-OO-RECLF is used redundantly to a fire opening relief valve gate IE-IORV. This does not seem appropriate for a fire-induced open event.	The fault tree logic has been changed. ADS-SRV-OO-RECLF is tied to non-fire induced SRV opening events. Fire-induced SRV opening was moved out from under the gate containing ADS-SRV-OO-RECLF to gate IE-IORV-SL, which is higher in the logic structure.
PRM-B11	PRM – Plant Response Model	Closed	4-9: The LERF HRA actions listed in Attachment E of NEDC 09-079 are not addressed for fire impacts or indication dependencies. The first four HFE on the chart were not found in the HRA report: HCI-XHEFO-CRMLT at 33%, SLC-XHE-FO-L2SLC at 2.8%, PCV-WWV-FO-	NEDC 09-079, "Task 7.5 Fire-Induced Risk Model," Attachment E (Attachment F in later revision), is the Level 2 Human Failure Events Review. Those HFEs that are not post-fire operator actions have been denoted as such. Those that are post-fire operator actions make reference to NEDC 09-083, "Task

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			VNTNG at 1%, CNT-XHE-FOWWVCL.	7.12 Fire Human Reliability Analysis." The four HFEs identified in this F&O are not in the HRA report because they are Level 2 events, not LERF events. All Attachment F HFEs were reviewed and LERF HFEs were verified to be included in the HRA report, NEDC 09-083.
PRM-B11	PRM – Plant Response Model	Closed	4-10: The NEDC-09-083 Table 4.3 impacts are based on evaluating the indication impacts on a fire zone basis. This requires SAFE to determine the instrument cabling impacted by zone and a logic analysis. SAFE is not referenced and the logical evaluation was not documented in a traceable fashion.	Existing Emergency Operating Procedure (EOP) operator actions were identified from a review of the Internal Events HRA analysis. The fire impacts due to fire damage to instrumentation are identified as part of NEDC 09-078, "Tasks 7.2 Component Selection," and NEDC 09-075, "Task 7.3 Cable Selection/Location." For any existing operator action where the instrumentation cables were not traced, the HFE was set to 1.0. It was assumed that these instruments would be unavailable for every fire, and with no instrumentation available for diagnosis the HEP is 1.0. NEDC-09-083, "Task 7.12 Fire Human Reliability Analysis" shows the HFEs retained in the Fire PRA and the instrumentation required for diagnosis for each HFE.
HRA-A1	HRA – Human Reliability Analysis	Closed	4-11: Actions NEDC 09-083, Table 4-4 should be set to 1.0 in the fire quantification or cables selection for the indication and a new fire HRA value assessment must be completed. These action[s] appear in Table 4-4 and the April 12th 2010 cutsets with a non-one value: PCV-XHE-FO-SWTCH, RCI-XHE-FO-BYPTP, SWS-XHE-FO-SWBPS, FPS-XHE-FO-RHR25A, SPC-XHE-FO-RCVR, SWS-XHE-FO-RCVR Although FPS-XHE-FO-RHR25A appears as a 1.0 in the tree, in one of the fire cutset files it is assigned a 0.1 value. 1.07E-08 FIRE-IE0000SPM FPS-XHE-FO-RHR25A	Table 4-4 was updated to Table 4 for the latest revision of NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis," and these conditions no longer apply. Those EOP HFEs modeled in the Fire PRA have been removed from Table 4. The fire scenarios were further reviewed, and additional detailed analysis was performed. This included not only detailed fire modeling and fire human reliability analysis, but also detailed circuit analysis and circuit failure likelihood analysis. Significant fire induced failures of these scenarios were reviewed, and no additional reductions in failure probabilities were found.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			ESC-XHE-FI-CS12BMCCY NBIPIS-TM-P52B SWS-MDP-FS-SWPB EAC-TRN-TM-SU.	Disposition of HFEs noted with non-1.0 values: FPS-XHE-FO-RHR25A now 1.0 for all cases. PCV-XHE-FO-SWTCH is retained at internal events value of 2E-4. The internal events model double counts this operator action. This HFE is included as part of the Hard Pipe Vent (HPV) actions. RCI-XHE-FO-BYPTP is retained at internal events value of 5.4E-3. The internal events model double counts this operator action. This HFE is included as part of the HPV actions. FPS-XHE-FO-RHR25A has 1.0 for internal events analysis due to RB conditions. For fire, the same conditions would be present, so operator action is not credited for Fire PRA. SPC-XHE-FO-RCVR & SWS-XHE-FO-RCVR - Internal events values are set to 1.0. The HEP of 1.0 is retained for the Fire PRA.
PRM-A1	PRM – Plant Response Model	Closed	4-12: Some fire impacts are inappropriately noted crediting beneficial failures. Gate ADS-SYS-000-0 is an ANDnor gate which nors events where the SRVs are both open and closed. Although the right intent, the nor side includes gate ADS-SRV-002-2 which models the actuation logic. Spurious logic actuation would not trump any direct fire failures associated with the SRVs. This also bring[s] up a general concern with the use of nested NOT gates. This [is] not handled well by the quantification engines.	SRV logic in the fault tree has been rebuilt and ties open SRVs to a fire-induced LOCA event (refer to Gate IE-IORV-SL). Actuation logic has been removed from under the AND nor gate. Nested NOTs have been reviewed and revised as needed.
PRM-A1	PRM – Plant Response Model	Closed	4-13: NEDC 09-079 PRM describes several items that do not appear in the tree. Per CNS these correction[s] should be made to	1. Gate IE-LOOP2-1 is correct as an AND gate. A turbine trip causes a loss of the normal transformer. In addition, the loss of both the Startup Transformer and the Emergency Transformer are needed to

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			align the tree with the documentation: 1. Make Gate IE-LOOP2-01 an OR gate 2. FIRE-DW-MSO should be added to DI5 3. Fire event SP-VS-III-1 missing as input to gate SP-VS-FAIL-I-III 4. Fire event FC1-DVR-2 missing as input to gate FC1-DVR See question sheet Peer review comment responses 4-13-10 - 2PM.xlsx for additional details.	produce a loss of offsite power event to the critical buses as modeled in the Internal Events PRA. 2. The internal events Level 2 model is used directly for Node DI. FIRE-DW-MSO no longer exists. 3. The internal events Level 2 model is used directly for Node SP. SP-VS-III-1 no longer exists. 4. The internal events Level 2 model is used directly for Node FC. FC1-DVR-2 no longer exists. The three events listed in items 2, 3, and 4 have been removed from NEDC 09-079 "Task 7.5 Fire-Induced Risk Model."
FSS-C1	FSS – Fire Scenario Selection	Closed	1-27: Transient fires in areas where the transient is next to a floor level cabinet or cable is not applied.	Transient fires have been postulated for all locations in the available floor spaces in each fire zone where detailed fire modeling was performed. The HRR distribution for each scenario is discretized into two points for the Cooper Fire PRA. The first point corresponds to the minimum HRR required to damage the nearest target, and its fire severity factor (SF) represents the fraction of fires that will damage only the ignition source itself. The second point corresponds to the 98th percentile HRR, and its SF represents the fraction of fires that will damage all targets within the Zone of Influence (ZOI) of the 98th percentile HRR, excluding the fraction of fires that will damage only the ignition source itself.
FSS-D7	FSS – Fire Scenario Selection	Closed	1-28: The NUREG 6850 suppression and detection numbers include reliability and unavailability. Adding the plant specific unavailability to the generic unavailability is double counting.	NEDC 10-022, "Review of CNS Fire Protection System Impairments," documents the method used to calculate plant specific system unavailability for use in the detailed fire modeling workbooks. The use of plant specific unavailability numbers is based on page P-6 of NUREG/CR-6850 which states "*These values provide realistic estimates of*

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
				system unreliability. However, the estimates do not include maintenance contributions to unavailability, credit for manual actuation of the system, dependent failures, and plant specific data." Furthermore, the reliability values recommended by NUREG/CR-6850 are from: NSAC 179L [W. Parkinson, et al, Automatic and Manual Suppression Reliability Data for Nuclear Power Plant Fire Risk Analyses, NSAC-179L, February 1994]. Page 1-1 of this report states that "*This study focused on providing data for systems failing to actuate or operate (i.e., reliability data). Hence, the resulting data do not consider the following failure contributions: unavailability contributions from maintenance and ineffective operation (i.e., when the system actuates but the fire is not extinguished).*"
FSS-E1	FSS – Fire Scenario Selection	Closed	1-29: The smoke detector that shuts down the HVAC (SD-1001) system does not use the recommended FDS values. This obscuration threshold is listed in R1906-07-011b-001 but the FDS actuation parameters are not the recommended values (for the Rev. 0 input files) and are not documented in R1906-07-011b-001.	Calculation NEDC 08-041, "Main Control Room Abandonment," includes the basis and justification for the smoke detector thresholds implemented in the abandonment analysis. All FDS input values are clearly documented in the latest revision to Calculation NEDC 08-041.
FSS-E3	FSS – Fire Scenario Selection	Closed	1-30: Review of the detailed fire modeling calculations NEDC-09-091 through NEDC-09-101, section 7.3 characterizes the uncertainty parameters. Mean values and distributions for these parameters are not established.	Refer to Calculation NEDC 09-086, "Task 7.15 Uncertainty and Sensitivity." A consensus approach for meeting Capability Category II is not available. Prevailing good practice addresses this supporting requirement qualitatively, and thus is intended to meet Capability Category I. A reasonable "qualitative" characterization of the conditional probability of an FDS, given a fire, is a 90% range of -10 to +5 on the calculated point estimate.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
FSS-A4	FSS – Fire Scenario Selection	Closed	2-21: During walkdown, the following issues have been noted. Due to the incomplete status of the multicompartment analysis, these issues are tracked in this F&O: 1. While traveling from fire zone 13A to 12D, some Div I cable trays are identified in the stairwell. Do not know if there is fire scenario for the stairwell, which normally may not be modeled. 2. There is a damper connecting fire zone 8E to the outside corridor area, which is blocked open. This may not be covered in current PP / FSS evaluations.	This F&O for NEDC 10-042, "Multi-compartment Analysis," has been addressed and a best practice received during the focused scope peer review for MCA. Transient fire scenarios have been developed for stairwells containing PRA targets in Fire Zone 13A in Calculation NEDC 09-095, "EPM Report R1906-711-TB-A - Detailed Fire Modeling Report Fire Compartment TB-A." The barrier connecting Fire Zone 8E and Fire Zone 8D and the associated features have been considered in NEDC 10-024 "Multi-Compartment Analysis".
FSS-D1	FSS – Fire Scenario Selection	Closed	3-1: The detection model is critical for application of the severity factor and non-suppression probabilities. The detection model used in NUREG 1805 is not validated and may not be conservative. The method essentially correlates detector actuation to a 10°C (18°F) temperature increase at the location of the detector. Although the ceiling jet/plume model is validated, the empirical link between detection and plume temperature has not. The NUREG 1805 smoke detector actuation spreadsheet does not recommend changing the activation temperature for the smoke detector but allows the user to change the ambient temperature. The spreadsheet consolidates three methods, so when the Alpert method is employed, the temperature differential between ambient and actuation should be set to 10°C. For this situation, the actuation	The detection correlation has been addressed in NEDC 10-020, "Verification and Validation of Fire Modeling Tools and Approaches for Use in NFPA 805 and Fire PRA Applications." The detailed fire modeling workbooks included as attachments to the detailed fire modeling calculations NEDC-09-091 through NEDC-09-101, and NEDC 10-043 through NEDC 10-049 have been revised to ensure the temperature differential between ambient and activation is 10°C.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			temperature should be changed; however, this is not consistently done in the detailed fire modeling. The result in an underestimate of the heat release rate threshold for detection for steady heat release rates.	
FSS-D1	FSS – Fire Scenario Selection	Closed	3-9: The combined effects of a hot gas layer and plume or radiant damage to a target are not considered in the detailed fire modeling cases. There is no specific threshold over which the detailed fire modeling tools are identified as being used outside their limits. The V&V basis document for the fire model simulator specifically states that they are not valid when the hot gas layer effects are significant.	The effects of the hot gas layer on zone of influence have been analyzed in Appendix B of NEDC 10-020, "Verification and Validation of Fire Modeling Tools and Approaches for Use in NFPA 805 and Fire PRA Applications." Detailed Fire Modeling Calculations NEDC-09-091 through NEDC-09-101, and NEDC 10-043 through NEDC 10-049 contain the results of the hot gas layer effects on zone of influence calculations for the specific fire zones being modeled.
FSS-C1	FSS – Fire Scenario Selection	Closed	3-10: An effective one point model is used in the detailed analyses (NEDC-091 - NEDC-101). The one point model typically used in the detailed fire modeling typically identifies a threshold fire size and evaluates the time to damage given a 98th percentile growing fire. The one point model meets the criteria for a Cat I FPRA and is reasonable for non-risk significant scenarios.	The HRR distribution for each scenario is discretized into two points for the Cooper Fire PRA. The first point corresponds to the minimum HRR required to damage the nearest target, and its fire SF represents the fraction of fires that will damage only the ignition source itself. The second point corresponds to the 98th percentile HRR, and its SF represents the fraction of fires that will damage all targets within the Zone of Influence (ZOI) of the 98th percentile HRR, excluding the fraction of fires that will damage only the ignition source itself.
FSS-D1	FSS – Fire Scenario Selection	Closed	3-12: There are no clear limits on the applicability of the zone of influence parameters. This includes the maximum flame height relative to the enclosure height and the applicability of the ceiling jet zone when the plume zone is below the ceiling. The zone of influence dimensions are	The zone of influence parameters are identified in NEDC 10-020, "Verification and Validation of Fire Modeling Tools and Approaches for Use in NFPA 805 and Fire PRA Applications." Detailed Fire Modeling Calculations NEDC-09-091 through NEDC-09-101, and NEDC 10-043 through NEDC 10-049 document the justification and evaluation of

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			provided without such constraint and used in determining the fire damage states.	potential limitations and impact on the zone of influence parameters on a scenario-by-scenario basis, where applicable.
FSS-D1	FSS – Fire Scenario Selection	Closed	3-13: The enclosure temperature fire modeling assumes a single open door and no mechanical ventilation in all cases. In many spaces, the doors to the spaces are closed. This approach is a reasonable screening method, and is ideal for the MQH methods used because it was correlated to spaces with door sized openings and diverges for small opening sizes. For small fires in moderate sized spaces where there is adequate oxygen available for sustaining the heat release rate, the assumption that the door is open may result in non-conservative room temperature predictions. There are fire modeling tools available (those covered by NUREG 1824 V&V in particular) that can be used to estimate the room temperature for situations where the door is closed, with or without forced ventilation.	The fire modeling workbooks included as attachments to the detailed fire modeling calculations NEDC-09-091 through NEDC-09-101, and NEDC 10-043 through NEDC 10-049 have been revised for fire zones in which the McCaffrey, Quintiere, and Harkleroad (MQH) method may have resulted in non-conservative results. The latest revision of the workbooks include the three methods identified in NUREG 1805 (MQH, Beyler, and FPA), for natural ventilation, closed compartment, and forced ventilation. Detailed Fire Modeling Calculations NEDC-09-091 through NEDC-09-101, and NEDC 10-043 through NEDC 10-049 document the justification for the fire modeling method for estimating room temperature used for each fire zone.
FSS-D4	FSS – Fire Scenario Selection	Closed	3-14: R1906-711-01, R1906-07-011b-001, and EPM-DP-FP-001. The assumed elevation of a panel fire (top of panel) is not documented in the detailed fire modeling, so it is not possible to determine if this location is consistent with the guidance in FAQ-0043 (1 ft below the top when vented or at the location of the top vent).	The use of fire elevation 1 foot below the top of the cabinet from FAQ-0043 is identified as a screening tool for the SDP, and the NRC Draft Interim Position on FAQ-0043 specifically states that "This assumption should be applied until detailed review of the cabinet contents can be performed for additional location determination, per Section G.3.2 of NUREG/CR-6850." Additionally the NRC Draft Interim Position on FAQ-0043 states "For cabinets that are neither vented nor considered sealed per FAQ 08-0042, the fire location would be assumed at the top of the door or opening which is expected to

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
				fail when fire damage occurs." Detailed Fire Modeling calculations were based on actual vent location, as visually examined in the field. For cabinets that were considered non-vented, but not robustly secured, the fire elevation was based on the top of the cabinet, which is consistent with the discussion in NRC Draft Interim Position on FAQ-0043. Also, applying the 1 foot below the top of the cabinet assumption could be non-conservative in some cases due to an open top or the door warping.
FQ-A1	FSS – Fire Scenario Selection	Closed	4-18: A review of the top cutsets indicates many large opportunities for improvement. Both the 3B and 3A zone failure evaluations would benefit from creating fire scenarios which would save the bus duct for the emergency transformer and in the case of 3B the cross power cables.	Detailed fire modeling workbooks have been developed for all fire zones deemed risk significant; including Fire Zones 3A (NEDC 10-046), 3B (NEDC 10-047), 8E (NEDC 10-043), 8F (NEDC 10-049), 8H (NEDC 10-043), 14A (NEDC 10-044), 14B (NEDC 10-045), and 13A (NEDC 09-095).
FSS-A6	FSS – Fire Scenario Selection	Closed	5-10: NEDC 08-041 Section 4.1.1 identifies the heat release rate profile of electrical cabinets is as recommended per NUREG 6850 Appendix G. The times used are 720 seconds (12 minutes) to reach peak HRR, 480 seconds (8 minutes) at peak HRR, and 720 seconds (12 minutes) for decay. The decay time does not match NUREG 6850 Appendix G.	An update to NEDC 08-041, "Main Control Room Abandonment," revised the fire growth profile to include a decay time of 20 minutes for the analysis.
FSS-D7	FSS – Fire Scenario Selection	Closed	5-11: NEDC 10-022 R0, Review of CNS FP System Impairments, documents the unavailability time for CNS fire protection systems. Section 5.8 of the detailed fire modeling reports (NEDC 09-101 R0 for Fire Compartment RB that contains Fire Zone	NEDC 09-101 includes the corrected reference to NEDC 10-022, "Review of CNS Fire Protection System Impairments."

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			3C) indicates that the results from the review of plant-specific data on detection and suppression system unavailability were implemented in the Fire Modeling Workbook.	
FSS-F1	FSS – Fire Scenario Selection	Closed	5-12: NEDC 09-090 R0, Exposed Structural Steel Impact, determines locations with exposed structural steel, identifies if any high fire hazards are present, and justifies exposed structural steel in high fire hazard locations quantitatively. Section 3.4.4 addresses Fire Zone 13A, the Turbine Operating deck. The only exposed structural steel identified supports the roof. The steel columns around the perimeter of Fire Zone 13A from the floor of the operating deck to the roof are not addressed. In addition, the only postulated fire is associated with oil in Fire Zone 13A. Oil fires on lower elevations are not considered.	An update to Calculation NEDC 09-090, "Exposed Structural Steel Impact," addresses the impact of fire on steel columns around the perimeter of operating deck and the exposure fire hazard caused by oil cascading to elevations below the operating deck.
FSS-F2	FSS – Fire Scenario Selection	Closed	5-13: NEDC 09-090 R0 identifies the 1100 °F failure temperature for exposed structural steel. NUREG 6850 guidance (including errata) for the spill area of unconfined oil spills are used. Calculations were performed to determine the minimum HRR to reach the damage threshold. The NUREG 1805 FDT spreadsheets, approved for use by NRC for fire modeling, were used to evaluate the time, temperature, HRR, and spill areas to result in structural steel damage. The bases for determining that structural steel damage thresholds would not be	An update to Calculation NEDC 09-090, "Exposed Structural Steel Impact," provides documentation and justification for the calculated fire durations, damage criteria, and the associated lumped thermal mass calculations.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			reached is in part based on the short duration (1 to 3 minutes), and that there is not sufficient time to reach the damage thresholds.	
FSS-D9	FSS – Fire Scenario Selection	Closed	5-14: Section 8.2.5 of EPM-DP-FP-001 R1, Detailed Fire Modeling, contains the criteria for assessment of potential smoke damage following the criteria of Appendix T to NUREG 6850. Section 8.2.5 identifies the following: "In general, short term smoke damage will only result for components housed in the same electrical panel as the fire source or in an electrical panel directly connected via a bus duct unless a specific installation feature precludes such damage." Only NEDC 09-093 (Fire Compartment NCS, Fire Zone 13B) and NEDC 09-097 (Fire Compartment YD, Yard) identify the potential for smoke damage per Section 8.2.5 of EPM-DP-FP-001.	The detailed fire modeling calculations NEDC-09-091 through NEDC-09-101, and NEDC 10-043 through NEDC 10-049 have been revised to include a section documenting smoke damage impacts on PRA components.
FSS-A6	FSS – Fire Scenario Selection	Closed	5-15: NEDC 10-001 R0, Main Control Room Analysis, evaluates fire scenarios within the MCR and the likelihood of spread to other locations in the MCR. Per Section 6.8.2 of NEDC 10-001, the Main Control Board, fire would not spread between the 6 panels of the MCB since they are open-backed and there are side panels between each panel. However, the approach used is based on Appendix S of NUREG 6850, and noted as such in the report. Appendix S is applicable to Chapter 11 evaluations of fire propagation to	The latest revision of NEDC 08-041, "Main Control Room Abandonment," addresses fire spread between the panels and provides justification for the Appendix S method implemented in the calculation.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			electrical cabinets, not the MCB.	
FSS-E4	FSS – Fire Scenario Selection	Closed	6-2: Task Plan – Task 7.15 Uncertainty and Sensitivity Analysis identifies that cable routing could be a source of potential inaccuracy. No specific mention of assumed cable routes.	Assumed target location is identified as a source of epistemic uncertainty for each Fire Compartment in the detailed fire modeling calculations NEDC-09-091 through NEDC-09-101, and NEDC 10-043 through NEDC 10-049, under the section "Characterization of Uncertainty for Fire Compartment." Additionally, the fire modeling workbooks included as attachments to the Detailed Fire Modeling Calculations identify where target failures are due to assumed target locations and routing.
FSS-H5	FSS – Fire Scenario Selection	Closed	6-4: Fire model outputs including characterization of uncertainty are provided in NEDC 09-091 through NEDC 09-101. 1. The uncertainty analysis has not been completed for the Main Control Room. The Main Control Room uncertainties that would need consideration, in addition to the input parameters associated with the fuel includes location dependencies for the source fire and grid dependencies on the solution. The FDS input files provided indicate that the number of cells used to approximate the fire dimensions (and thus the plume entrainment characteristics) is two. It should be verified that increasing the solution resolution in the area of the fire does not adversely affect the smoke layer development and the radiant heat fluxes predicted at secondary combustible targets (cable trays). 2. Not all relevant model outputs are provided in the MCR abandonment calculation. The temperature, heat flux	NEDC 08-041, "Main Control Room Abandonment," includes a detailed Uncertainly and Sensitivity Analysis in Appendix A. In addition, NEDC 08-041 includes relevant output data from the numerous FDS simulations and calculations.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			information necessary to make the correct determination as well as the heat release rate modeled in FDS (to demonstrate the fire is correctly simulated and provided indication of oxygen limitation effects) should be provided for all cases in the calculation.	
IGN-A9	IGN – Ignition Frequency	Closed	1-3: Transient fire frequencies are listed in Table C-1, 2, and 3 of the ignition frequency calculation NEDC 08-032 r1 Fire Ignition Frequencies.pdf. All fire areas, other than the drywell, include fire frequencies for transient fires. However, some of the rankings provided in Table C-2 do not appear to follow the guidance provided in NUREG/CR-6850 for maintenance occupancy and storage. The following are examples: 8A - Auxiliary Relay Room is ranked with 1 for maintenance, occupancy, and storage (M/O/S). A ranking of 1 would basically mean that each are prohibited by plant procedures - which is unlikely. These appear to be a 3 for maintenance and occupancy, and a 3 for storage unless the area has special transient controls over most areas (prohibited storage). Areas 8B and 8C are RPS rooms 1A and 1B and are ranked as 3, 1, 1 for M/O/S. It appears that O/S are likely a 3 each, using similar logic to 8A above. No areas in the Control or Reactor Building are listed as a 10 for storage. It appears it is likely there are approved storage areas in these buildings. The walkdown confirmed the following: a) 8H (cable spread room) should probably	Calculation NEDC 08-032, "Fire Ignition Frequencies," was updated to include the following: Fire Zones 8B and 8C were revised to "3" for maintenance, occupancy, and storage influencing factors. Reactor Building and all zones of the ignition frequency calculation were revised based on their combustible control ranking in CNS Procedure 0.7.1. Critical Switchgear Rooms (3A and 3B) were revised to "3" for maintenance and occupancy, and a "1" for storage influencing factors. DC Switchgear Rooms (8G and 8H) were revised to "3" for maintenance, occupancy, and storage influencing factors. Fire Zones 8E and 8F (Battery Rooms) were revised to "3" for maintenance, "1" for occupancy, and "3" for storage influencing factors. Areas of the Turbine Building (13A, 12B, and 11D) were changed to "50" for maintenance influencing factor. Fire Zone 9A (Cable Spreading Room) and Fire Zone 8A (Auxiliary Relay Room) have been designated as transient-free/hot work-free fire zones and a "Very Low" (0.05 or 0.1) ranking has been applied to the influencing factors for these zones. These factors were applied to reflect a significantly

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			have a larger than 1 for maintenance/ occupancy due to amount of traffic and equipment in the area, b) Area 12D had numerous transients stored including an open barrel of lube oil, and should be ranked higher than 3 for storage, c) Areas 3A/3B were easily entered and likely has occupancy to perform surveillance, inspections, etc - and should likely be ranked as a 3, d) other areas such as 8g/8h/8b/8c should be ranked higher for occupancy since no entry restrictions were present: e) battery rooms 8e/f were locked requiring a key, and a ranking of 1 may be appropriate. None of the areas appear to have a 50 ranking for maintenance.	lower-than-average level of the factor. A very low maintenance factor (0.05) was applied to areas where maintenance and hotwork are administratively prohibited during at-power operation. A very low storage factor (0.1) was applied if storage of transient materials is prohibited administratively. These very low factors are not specified in the NUREG/CR-6850 methodology. However, since the administrative restrictions in place at CNS significantly reduce the likelihood of fire ignition in the affected areas, and since the likelihood that these administrative restrictions are violated is judged to be very small considering how these restrictions are applied at CNS, it is concluded that the very low factors are appropriate for use in the CNS Fire PRA analysis.
IGN-A5	IGN – Ignition Frequency	Closed	3-6: The generic ignition frequencies are not weighted by the fraction of time that the plant is at power.	NEDC 08-032, "Fire Ignition Frequencies," was revised to include the Average Criticality Factor of 0.875 from Table B-4 of CNS-PSA-001, "Initiating Event Notebook." This value is used to convert the initiating event frequencies for this PSA update from critical years to calendar years. This criticality factor is considered representative of the continual improvement in plant operation and appropriately represents future operation. Therefore, each of the Bin generic ignition frequencies identified in Table 3-2 has been updated to plant-specific values using the 0.875 Average Criticality Factor multiplier. Table 3-4 documents the weighted Bin frequencies for the CNS fire ignition frequency calculation.
IGN-A7	IGN – Ignition Frequency	Closed	5-4: A separate spreadsheet, not included as part of NEDC 08-032 R1, is used to total the ignition source bins for each fire zone that are then summed up in Tables A-1 and	All tables in Attachments A, B, and C have been corrected in NEDC 08-032, "Fire Ignition Frequencies," by including the information directly from the Microsoft Excel spreadsheet used to

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			A-2.	perform the calculations to prevent typos in reproducing the data in the Microsoft Word document.
IGN-A9	IGN – Ignition Frequency	Closed	1-2: Transient fire frequencies are listed in Table C-1, 2, and 3 of the ignition frequency calculation NEDC 08-032 r1 Fire Ignition Frequencies.pdf. All fire areas, other than the drywell, include fire frequencies for transient fires. The drywell can have a transient fire, given the containment inerting fails and transient fire sources are present in the drywell. Although unlikely, the Fire Section of the standard was originally changed to remove the drywell exception to the standard, meaning the standard words under IGN-A9 are also meant to apply the drywell.	The influencing factors for "Maintenance," "Occupancy," and "Storage" for the Drywell have been changed to "1" for each in NEDC 08-032, "Fire Ignition Frequencies," to identify the potential for a transient fire in the drywell given containment inerting failure.
IGN-A1	IGN – Ignition Frequency	Closed	1-22: Met. NEDC 08-32 Table 3.2 has frequency data that is consistent with EPRI 1011989, except for bins 12, 16a, and 16b which don't match (typos). The correct frequencies are used in the ignition source data sheets. Two frequency bins applicable to PWR plants are excluded.	Table 3.2 has been corrected in NEDC 08-032 "Fire Ignition Frequencies," by including the information directly from the Microsoft Excel spreadsheet used to perform the calculations to prevent typos in reproducing the data in the Microsoft Word document.
IGN-A7	IGN – Ignition Frequency	Closed	1-23: Walkdown confirmed the following discrepancies in the ignition counts: 9A (Cable room): TB-C324 is a large cabinet (6x2x3) that wasn't counted, multiple security cabinets were not counted (appeared to be new), there were 2 AC fans in the area (1 was counted), 3 cabinets were listed as less than 4 switches by were large (6x6x2) and should not be	Walkdown discrepancies have been corrected in NEDC 08-032, "Fire Ignition Frequencies" by the following: 9A TB-C324 has been included in the ignition frequency count as a ventilated electrical cabinet and a fire scenario developed under detailed fire modeling. Multiple security cabinets were added to the ignition

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			screened. 8E- EE-PNL-A and AA2 were listed as robustly secured, but were not. 8H: EE-DSC-250A was not documented (sealed, secured). 3B: Ground fault indicator panel (sealed, secured) is not listed,	frequency calculation as non-ventilated, robustly-secured cabinets. The additional AC fan has been added as a <5 horsepower motor, and therefore, not included in the count for ventilation subsystems. The notes section of the ignition frequency calculation for the 3 (6x6x2) cabinets has been changed to "Non-ventilated, Robustly-Secured" instead of "Less than 4 switches". 8E The notes section of EE-PNL-A and AA2 has been revised to remove "Non-ventilated, Robustly-Secured" (all panels of similar construction have been revised to the same) and fire scenarios have been developed under detailed fire modeling. 8H EE-DSC-250A has been added to the ignition frequency calculation to Bin 15a and the notes section states "Non-ventilated, Robustly-Secured." The detailed fire modeling calculation for Fire Zone 8H includes a fire scenario for EE-DSC-250A, which damages only those PRA targets that terminate at the ignition source as the panel is well sealed, robustly secured, and non-ventilated. 3B Ground fault indicator panels have been added to the ignition frequency calculation as "Non-ventilated, Robustly-Secured – Less than 4 switches".
IGN-A7	IGN – Ignition Frequency	Closed	1-24: The Cable Spread Room (9A) includes a number of cabinets listed as secure, but with a single latching device. FAQ 42 basically says: "Simple twist-	The justification for these panels being Robustly-Secured with three-point twist handle latches has been added to Calculation NEDC 08-032, "Fire Ignition Frequencies."

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			handle style top-and-bottom door latches are not sufficient to contain a fire within a panel." It was not clear whether these doors attach at three locations, and if the three location attachment would prevent the type of problem with door warping discussed in FAQ 42.	
IGN-A4	IGN – Ignition Frequency	Closed	3-5: 1. One criterion for eliminating plant event updates for a particular frequency bin is that the plant event frequency is between the 5 and 95th percentile of the generic frequency. The time interval used to make this determination is 40 yrs; however, the data events are from an 8 year span. This results in a plant event frequency that is 1/5 the intended value. 2. The stated fraction of potentially challenging events in Section 3.4.2 of NEDC-032 is 15/28 but the actual fraction is 11/28.	NEDC 08-032, "Fire Ignition Frequencies," states that the data events were over an 8 year period and the Plant-Specific Bin IEF (Point estimate - /ry) was calculated using the 8-year period to identify if a Bayesian update was necessary. The fraction of potentially challenging events was revised to 10/28 in the updated NEDC 08-032.
CF-A1	CF – Circuit Failure	Closed	1-31: Initial quantification of circuit failures is set to a generic industry value of 0.3 or 0.6. When significant, specific circuit analysis is performed, this is documented in the Task 10 report. Table B of the Task 10 report is derived from the Task 9 report. Circuit failure likelihood is then developed in Table E-1 for each cable based on the generic SO probabilities, and the cable type, failure mode, etc. The specific circuit configuration for cables is included in the consideration. However, the plant specific	The fire scenarios were further reviewed, and additional detailed analysis was performed. This included not only detailed fire modeling and fire human reliability analysis, but also detailed circuit analysis and circuit failure likelihood analysis. Significant fire-induced failures of these scenarios were reviewed. The results of the detailed circuit analysis and circuit failure likelihood analysis are found in Task 9 (Calculation NEDC 09-073) and Task 10 (Calculation NEDC 09-082).

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			analysis was not performed for all significant SO events. For example, in 3B, there are 5 events with FV> 0.01 that contain generic probabilities (0.6 or 0.3). In this area, all cables appear to be in conduit, which would likely result in a lower spurious operation probability for these events (unless SO is inside of an electrical cabinet).	
HR-G1	HRA – Human Reliability Analysis	Closed	1-9: Detailed HRA was performed for most HFEs included in the FPRA. The EPRI HRA Calculator was used for this analysis. The results are included in the HRA notebook, Appendix B. Some HFEs are conservatively applied as 1.0 or using screening values in the FPRA. See table 4-7 for screening HFEs. A review of the CDF results indicates that a number of the significant HFEs are included with screening values in the results. For example, PCV-XHE-FO-AOV is included as a 1.0 for fire area 3B, along with a number of other HFE events. Similar events are included in other significant fire areas.	NEDC 09-083 "Task 7.12 Fire Human Reliability Analysis" documents the completed detailed HRA analysis for those HFEs in the Fire PRA as required. The fire scenarios were further reviewed and additional detailed analysis was performed, if possible. This included not only detailed fire modeling and fire human reliability analysis, but also detailed circuit analysis and circuit failure likelihood analysis. For PCV-XHE-FO-AOV in particular, from the Internal Events HRA Notebook Section 3.43.4: "There are a number of associated HEPs for Containment Venting that deal with local actions outside the control room. These additional local HEPs are set to a failure probability of 0.1 for the current version of the PRA because they are a last resort action; they may not be directed until the pressure is near PCPL. At that time, it is judged the local actions could be too late to accomplish the action before exceeding PCPL. Therefore, a high failure probability of 0.1 is chosen for the model." This operation will occur long after the fire has been put out, so the IE PRA evaluation applies. The use of 0.1 for PCV-XHE-FO-AOV is not applicable in all fire zones due to various conditions, such as fire location preventing the operator manual action. In

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
				these cases, 1.0 is applied.
HRA-A4	HRA – Human Reliability Analysis	Closed	4-22: Although there was a general over view of operational philosophy, there was not a talk-through of each new operator action.	Attachment A of NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis," was updated after further discussion of selected dominant scenarios and their associated operator actions. These actions were discussed in additional operator interviews. Also, a talk-through of actions in Attachment M, "Operator Manual Action Timelines," was conducted.
HRA-C1	HRA – Human Reliability Analysis	Closed	4-25: Currently, HRA document 09-083 does not address the impact of fire on internal events related actions as related to travel path.	During the second interview with operators, effects of fires on travel paths were specifically reviewed, and no significant change in travel time could be determined to occur. No further action is required. This is documented in Attachment A of NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis,"
HRA-A1	HRA – Human Reliability Analysis	Closed	4-26: Although the 09-083 HRA report is of high quality and comprehensive, it is a casualty of the evolving nature of the Cooper project. Several HFE are in the current FQ cutsets that are not addressed in the fire HRA report. In a sample of 19, these four actions were identified as not being addressed: RRSXHE-FI-RRTRIP, FPS-XHE-FO-RHR25A, EAC-XHE-FI-SS1G, EAC-XHE-FI-DGFOTP.	HFEs were reviewed for inclusion as Fire HFEs. In addition, the HFEs RRS-XHE-FI-RRTRIP, FPS-XHE-FO-RHR25A, EAC-XHE-FI-SS1G, and EAC-XHE-FI-DGFOTP have been verified as being addressed. This action was included in NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis,"
HRA-A1	HRA – Human Reliability Analysis	Closed	4-27: Although the 09-083 HRA report is of high quality and comprehensive, the basis for many of the actions is the internal event model. Several HFEs are in the current FQ cutsets that do not have an adequate timing basis or T delay established. This calls the HRA value into question and does not allow for an adequate dependency evaluation. In a sample of 19 actions, these	Timing basis and Tdelay were reviewed for Fire HFEs, and updates were made as necessary to ensure appropriate HFE values and to allow accurate dependency evaluation. This action was included in NEDC 09-083 "Task 7.12 Fire Human Reliability Analysis".

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			two actions were identified as not being addressed: FPS-XHE-FO-DISEL, SWS-XHE-FO-SWPS.	
QU-D6	FQ – Fire Quantification	Closed	7-8: The listing of significant sequences does not include all sequences considered significant (see the PRA Standard definition of significant). The total listed sequences in Section 7.2 of CNS calculation 17712-011, Task 7.14 Fire Risk Quantification, only provide a total of about 65% of CDF. Additionally, the failures generally listed in the descriptions of the significant accident sequences do not help determine significant basic events. For example, for 14A_D-1~F0.cut, the top events are spurious EDG start, spurious EDG breaker closure, RHR pump maintenance, Miscalibration of position switch, RHR pump FTS, Strainer plugging, RHR HX in maintenance, and SW pump in maintenance. Most of these are not described in the accident sequences, so there is no method for determining that these events are considered significant for the FPRA (if used for applications). Due to some limitations (e.g., different values for certain events in different scenarios) of the cutset files, the total CDF/LERF cutset files are not merged. Significant sequences were identified by sequence contributions calculated from ignition frequencies and CCDP's. However, significant basic events have not been documented.	This finding does not affect the results and use of the Fire PRA in fire risk evaluations. The calculation documentation was enhanced to clarify the definition of risk significance, and that all risk significant sequences at the fire damage state combined with CCDP and CLERP level are included in an attachment to the calculation. The CNS Fire PRA is extremely detailed in order to account for sequence dependent basic event values including spurious operations and human interaction.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
QU-A5	FQ – Fire Quantification	Closed	8-3: CNS calculation 17712-011 Task 7.14, Fire Risk Quantification, Section 6.1 and Attachment C discuss the recovery actions. Dependency analysis is documented in CNS Calculation 17712-003 Rev 1, Task 7.12 Fire Human Reliability Analysis. However, the recovery rules treating the dependent HEPs are neither applied nor documented. For example, combination 169 is analyzed, which is in the 8th top cutset in file 3B_4160GHFDS1.CUT: SWS-XHE-FO-WNDML-E_OMA ESC-XHE-FI-CSPAOMA_OMA	HEP dependency was completed and added to the noted calculation with no significant impact on results and insights. The recovery rule file based on the dependency analysis is provided in Attachment C of NEDC 09-085, "Task 14 Fire Risk Quantification." Fire PRA results that incorporate dependent HFEs are provided in Attachment D of NEDC 09-085.
QU-F6	FQ – Fire Quantification	Closed	8-5: The quantitative definition used for significant basic event, significant cutset, and significant accident sequence is not documented. CNS Calculation 17712-011 Task 7.14, Fire Risk Quantification, Attachment B provides a summary of the sequences (measured at the FSF combined with CCDP level) which contribute greater than ~99% to the calculated CDF and LERF results. However, this does not cover all the quantitative definition for significant basic event and significant cutset.	See disposition for SR QU-D6, F&O 7-8.
LE-G3	FQ – Fire Quantification	Closed	8-6: CNS Calculation 17712-011, Task 7.14 Fire Risk Quantification, Section 7 and Attachment B document the significant contributors to LERF. However, the relative contribution of contributors (i.e., plant damage states, accident progression sequences, phenomena, containment challenges, containment failure modes) to	See disposition for SR QU-D6, F&O 7-8.

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
			LERF is not documented.	
QU-E3	UNC – Uncertainty and Sensitivity Analysis	Closed	1-17: No quantitative estimate of CDF/LERF uncertainty intervals was performed for CNS. Additionally, no quantitative estimate[s] for many of the basic event uncertainties is provided. For example, the quantitative EFs for the Ignition Frequencies are not provided. As a result, the quantitative estimate of the overall parameter uncertainty cannot be estimated or quantitatively performed.	First, the approach and results subject to the peer review is summarized, and then an updated discussion is provided. Summary of approach and results: Calculation NEDC-09-086, "Task 7.15 Uncertainty and Sensitivity Analysis," which was the basis for the peer review, provides the uncertainty and sensitivity analyses. The following summarizes the philosophy for addressing the uncertainty quantitatively used in the calculation: • From Section 1, "A complete, quantitative analysis of uncertainties and assumptions is not conducted. This is, for the most part, consistent with the PRA Standard [4] and referenced guidance documents [1, 2, and 3], and consistent with prevailing good practices. The focus is on a qualitative characterization. Where appropriate sensitivity evaluations will be conducted when applying the FPRA." • Table 3, in Section 6 discussed the approach to addressing parameter uncertainties. • From Section 8, Conclusions, "The results of this task are used to support integrated decision making. As has been noted, a quantitative characterization has not been developed as the quantitative results are conservatively biased for key contributors. A better estimate of the mean value for CDF and LERF is estimated to be a factor of 5 to 10 lower than calculated with a 90 percentile range of a factor of 10 on the lower end and 5 on the higher end. Relevant uncertainties will be addressed in the use of the FPRA. The major conservatisms are believed to be fire ignition frequency and the development and progression

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
				of a fire. Both increase the calculated frequency of core damage and large early release. Generic fire frequencies are directly based on assumptions in NUREG/CR-6850 (including FAQ-48 enhancements) estimating the severity and applicability of fires from incomplete fire event reports which lack critical information in the Fire Events Database. In the absence of complete data, the ignition frequencies remain conservative based on fires being assumed to be challenging fires, event duration being assumed the same as the actual fire duration, and lack of detailed damage information. These incomplete fire events provide the basis for the probability of non-suppression values for manual fire fighting. Additionally, the assumptions involving the growth and propagation of a fire including non-realistic peak heat release rates, cable flame spread rates, and cable tray propagation rule sets directly lead to reduction in effectiveness of detection/suppression, and time available for operator action. Given the significant contributors to calculated results, the modeling and quantification of human reliability also can influence the results considerably. This occurs because for many of the significant sequences safe shutdown requires use of alternate shutdown which can involve many coordinated actions, and in some sequences with limited time margin for accomplishing the actions. State-of-the-art methods have been used but there is the potential for central tendency of the results to be either optimistic or pessimistic. The conservatism in fire growth time is expected to more than offset the potential for optimistic results to significantly influence calculated results." • Attachment A to the calculation documented the

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
				approach for each related supporting requirement in the Standard. **Discussion:** As noted, an estimate of the uncertainty interval was provided. However, the peer review team finding is related to the technical basis for this estimate and that each of the technical areas contributing to this estimate was not explicitly addressed. Thus the following is provided: • The start of the Fire PRA CDF and LERF calculations are the fire damage states (FDS) and associated frequencies. More than 800 FDS were developed; combined with the cut-sets associated with each FDS more than 100,000 sequences at the cut-set level were developed. • The FDS include the ignition frequency and severity factor, detection and suppression conditional probabilities, and the fire modeling needed to develop these conditional probabilities. • The cut-sets address data (DA), human reliability (HR), and circuit failure (CF). The DA parameter uncertainties are addressed in the Internal Events PRA and are directly relevant to the Fire PRA. • As noted in the Rev. 0 calculation, "A better estimate of the mean value for CDF and LERF is estimated to be a factor of 5 to 10 lower than calculated with a 90 percentile range of a factor of 10 on the lower end and 5 on the higher end. Relevant uncertainties will be addressed in the use of the FPRA. The major conservatisms are believed to be fire ignition frequency and the development and progression of a fire. Both increase the calculated frequency of core damage and large early release." • To support this estimate, consider as typical for a

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
				sequence which results in core damage or large early release: o Ignition Frequency uncertainties range factor is on the order of 10 based on NUREG/CR-6850. o The estimated conditional probability of an FDS, given a fire ignition, has an uncertainty range factor on the order of 10 on the low side, and 5 on the high side, based on judgment (includes fire development and propagation, and detection and suppression). o The range factor for DA can be approximated by a factor of 3 to 5 based on the Internal Events PRA, and typical uncertainties associated with unavailability and failure rates in recognition that the actual parameter uncertainty depends on the specific cut-sets. o The range factor for HR can be approximated by a factor of 3 to 5 based on the HRA calculation. o The range factor for CF can be approximated by a factor of 4 on the low side and 2 on the high side based on NUREG/CR-6850. o The uncertainty on large early release given a core damage end state has a range factor, again on average, on the order of 3 to 5 based on the Internal Events PRA and other PRA studies. Thus, typical significant CDF sequences can be represented by: CDF1 (where a random failure must occur after the FDS is reached, such as RCIC fails to start) = Ignition Frequency ("-10/+10") * Conditional FDS frequency ("-10/+5") * DA ("-5/+5"). CDF2 (where a human error must occur after the FDS is reached, such as failure to realign a valve

Table V-1 Fire PRA Facts and Observations

SR	Topic	Status	Fact/Observation	Disposition
				after inadvertent operation of the valve due to a spurious signal causes flow diversion) = Ignition Frequency ("-10/+10") * Conditional FDS probability ("-10/+5") * Conditional CF likelihood probability ("-4/+2") * HR failure probability ("-5/+5"). Thus, as provided in the Rev. 0 calculation, a reasonable estimate of the uncertainty interval is minus 10 to plus 5 on the calculated mean value, where the mean is estimated to be on the order of a factor of 5 to 10 lower than calculated.
MU-D1	MU – Maintenance Update	Closed	1-34: The Fire PRA solution process scripts and fire modeling are not currently controlled by the plant procedures.	The Fire PRA database will be controlled as an electronic document in the same way the Internal Events PRA model (CAFTA model) is controlled. See Implementation Item S-3.24 of Attachment S, Table S-3.

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
SY-A2	CNS PRM report NEDC 09-079 Rev. 0 Section 4.2.4 includes some discussion on system models. The internal events PRA fault trees were modified to include the failure modes caused by fires and to add ISLOCA pathways, and IORVs caused by spurious component operation. In addition, the fault trees, while sufficient for use in the internal events PRA, were enhanced to ensure they were sufficient to meet the needs of the fire PRA as well. Significant system model changes have been implemented. However, the documentation is not sufficient for review with respect to the pertinent information listed in SR SY-A2 and other SY SRs. Moreover, the feedwater system model is significantly enhanced in the fire model, which would be appropriate to update the internal events system model to capture all the updated information. As a result, SY-A2 is considered not met.	Current CC: Met All. Calculation NEDC 09-079, "Risk Model Development," provides a list of references used in the development of system modeling, and NEDC 09-078, "Fire PRA Component Selection," lists references used in the component selection process. The references provide the pertinent information referred to by this supporting requirement or point to sources of where this information can be located. References include post-fire response procedures and other operating procedures that represent the as-operated plant. Additionally, the Fire PRA MSAccess Database is referenced which includes the SAP Database ID. The SAP Database is a CNS-specific database that contains the details of applicable as-built resources such as P&IDs, I&C drawings, and other information important for supporting a sufficient review of the criteria for this supporting requirement. The Fire PRA database is an electronic document that will be controlled in the same way as the Internal Events PRA Model (CAFTA Model). See Implementation Item S-3.24 of Attachment S, Table S-3. As noted in the review, the Feedwater system modeling was enhanced to support Fire PRA modeling.
SY-A3	See SY-A2 assessment.	Current CC: Met All. Fire PRA modeling was performed in a manner that maintained the integrity of the Internal Events Model and allows for proper evaluation of both the Internal Events PRA and the Fire PRA. The Internal Events Model is documented separately

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		and was evaluated under the Internal Events Peer Review. The information in SR SY-A3 does not have to be recreated in the development of the Fire PRA Model. However, new components and fire-induced impacts should be considered. The new components are listed in NEDC 09-079. Applicable plant information, such as post-fire operating procedures are referenced in the Fire PRA supporting calculations for component selection (NEDC 09-078), Fire PRA model development (NEDC 09-079), and (NEDC 09-083) Fire Human Reliability Analysis. Additionally, the Fire PRA MSAccess Database is referenced by these calculations which include the SAP Database ID. The SAP Database is a CNS-specific database that contains the details of applicable as-built resources such as P&IDs, I&C drawings, and other information germane to allowing a sufficient review of the criteria for this supporting requirement.
SY-A4	See SY-A2 assessment.	Current CC: Met II/III. The Fire PRA was developed using the CNS Internal Events PRA. Plant walkdowns were performed; however, this was not performed explicitly for SSC review for system analysis, but rather for fire modeling. Licensed operators were interviewed for HRA, and others (licensed operators, system engineers, etc.) were part of an Expert Panel convened for analysis of multiple spurious operation. No changes were made in the Fire PRA model that would invalidate internal events systems analyses or the walkdown and interview results.
SY-A6	The system boundary may be changed due to the updates to the system models in fire PRA, such as the feedwater system model. The addition of instruments may also require the updates to the system boundaries. SY-A6 is considered not met.	Current CC: Met All. The system boundary is defined in the Internal Events documentation which was not revised for Fire

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		PRA modeling. Fire PRA documentation; Calculation NEDC 09-079 identifies new components added to the model.
SY-A12	Some fire impacts are inappropriately modeled crediting beneficial failures. See F&O 4-12 for details. As a result, SY A12 is considered not met.	Current CC: Met All. In accordance with NEDC 09-079, Section 4.4, "...if a passive failure will place a component in an acceptable configuration, modeling is not included because the spurious operation would be similar to modeling a failure as a success." Furthermore the calculation states: "Failures that place components in a successful position are not modeled in the fault tree logic." In regard to the reviewer's example of where this was overlooked, SRV logic in the Fire PRA fault tree has been rebuilt, and nested NOTs have been reviewed and revised, as needed.
SY-A24	Followed internal events model. However, repairs modeled in internal events have not been evaluated for fire PRA model. See SPC-XHE-FO-RCVR, SWS-XHE-FO-RCVR in F&O 4-11.	Current CC: Met All. Table 4-4 was updated to Table 4 of NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis," and these specific repairs modeled in internal events no longer apply and have been removed from Table 4. Disposition of HFEs noted: SPC-XHE-FO-RCVR & SWS-XHE-FO-RCVR - Internal events values are set to 1.0 (always failed). The HEP of 1.0 is retained for the Fire PRA. The Fire PRA did include repair of battery chargers and diesel generator fuel oil transfer pumps. These repairs are included in the fire response procedures, have the needed parts and equipment pre-staged, and include timing assessments. A detailed human reliability analysis was performed to determine the HEP. The fire scenarios were further reviewed and additional detailed analysis was performed. This included not only detailed fire modeling and fire human reliability analysis, but also detailed circuit

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		analysis and circuit failure likelihood analysis.
SY-C1	Significant system model changes have been implemented. However, the documentation is not sufficient for review with respect to the pertinent information listed in SR SY-A2 and other SY SRs. As a result, SY-C1 is considered not met.	Current CC: Met All. Calculation NEDC 09-079, "Risk Model Development" describes how the Fire PRA Model was developed and lists the files that document the modeling. The fault tree modifications implemented as part of the Fire PRA are documented in the Fire PRA Database which is an electronic document that will be controlled in the same way the Internal Events PRA Model (CAFTA Model) is controlled. See Implementation Item S-3.24 of Attachment S, Table S-3. The Fire PRA Database provides the details necessary for reviewing the Fire PRA system analysis. Specific information includes a detailed description of the fault tree enhancements made to the Internal Events Model, and cross-referenced to the SAP Location. The SAP Database is a CNS-specific database that contains the details of applicable as-built resources such as P&IDs, I&C drawings, and other information germane to allowing a sufficient review of the criteria listed in SR SY-A2 and other SRs. Therefore, the logic used for Fire PRA modeling is documented in the Fire PRA calculations and is not required by the Standard to be included with the Internal Events documentation.
SY-C2	Significant system model changes have been implemented. However, the documentation is not sufficient for review with respect to the pertinent information listed in SR SY-A2 and other SY SRs. As a result, SY-C2 is considered not met.	Current CC: Met All. Calculation NEDC 09-079, "Fire Induced Risk Model," documents the process and changes made to develop the Fire PRA Model. Aspects of the types of documentation listed in SY-C2 that differ for Fire PRA Modeling, as compared to Internal Events Modeling, are defined and documented in the Fire PRA Documentation. For example, nomenclature

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		used in Fire PRA Modeling is defined in Section B.2 of NEDC 09-079. Fire PRA Modeling does not represent a change to the Internal Events Model; therefore, it is appropriately not included in the Internal Events documentation. Human failure events (HFEs) added to the Fire PRA Model to represent post-fire human actions are documented by calculation NEDC 09-083, "Fire Human Reliability Analysis."
SY-C3	Significant system model changes have been implemented. However, the documentation is not sufficient for review with respect to the pertinent information listed in SR SY-A2 and other SY SRs. Additionally, plant specific uncertainty due to PRA model changes is not discussed in Table 5 of the uncertainty analysis. As a result, SY-C3 is considered not met.	Current CC: Met All. The Fire PRA system analyses are documented in calculation NEDC 09-079, "Risk Model Development." Calculation NEDC 09-086, "Uncertainty and Sensitivity," provides a summary of assumptions and uncertainties from NEDC 09-079. NEDC 09-079 addresses F&O associated with this task. An expanded discussion was added on assumptions and uncertainties to explicitly address this supporting requirement.
HR-G7	Based on the review of draft model files, CNS methodology cannot meet HR-H3 and QU-C1 requirements for the dependencies among multiple HFEs that potentially impact significant accident sequences/cutsets. The current quantification method does not use higher HEP values in quantification and does not apply recovery file that includes HEP combination events. As a result, HR-G7 is considered not met.	Current CC: Met All. Dependency Analysis was completed in NEDC 09-083, "Task 7.12 Human Reliability Analysis," and is summarized in NEDC 09-085, "Task 7.14 Fire Risk Quantification," as follows: Dependencies among operator actions exist for any of the following parameters that may have a common effect on operator actions including timing, procedures, cues, personnel, and staffing resources. Dependencies arise from post-initiator human error events that may be linked in the quantification process. If these operator actions are not independent, then any sequence cutsets associated with two or more of these dependent operator actions would be incorrect. The dependency analysis was performed in Task 7.12 using the EPRI HRA

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		Calculator. The dependencies were identified and examined in the analysis by identifying combinations of operator actions and then determining their level of dependence.
HR-H3	Based on the review of draft model files, CNS methodology cannot meet HR-H3 and QU-C1 requirements for the dependencies among multiple HFEs that potentially impact significant accident sequences/cutsets. The current quantification method does not use higher HEP values in quantification and does not apply recovery file that includes HEP combination events.	Current CC: Met All. Dependency Analysis was completed in NEDC 09-083, "Task 7.12 Human Reliability Analysis," and is summarized in NEDC 09-085, "Task 7.14 Fire Risk Quantification," as follows: Dependencies among operator actions exist for any of the following parameters that may have a common effect on operator actions including timing, procedures, cues, personnel, and staffing resources. Dependencies arise from post-initiator human error events that may be linked in the quantification process. If these operator actions are not independent, then any sequence cutsets associated with two or more of these dependent operator actions would be incorrect. The dependency analysis was performed in Task 7.12 using the EPRI HRA Calculator. The dependencies were identified and examined in the analysis by identifying combinations of operator actions and then determining their level of dependence.
DA-C2	The majority of component failure probability values, excluding spurious operation, do not need reanalysis for the Fire PRA. However, a query of BE table in the cnsone.rr database shows that several events with a style of 'fire' may be added in the fire models and require DA reanalysis: CRD-SOV-CC-SO140A, CRD-SOV-CC-SO140B, LCS-CKV-LK-18CV, LCS-CKV-LK-19CV, RCI-CKV-LK-18CV, RCI-CKV-LK-19CV, RHR-CKV-LK-26CV, RHR-CKV-LK-27CV. As a result, PRM-B12 and B13 are considered not met. Check valve leakage can be uncovered through surveillance.	Current CC: Met All. With respect to new components added to support Fire PRA modeling; basic event/parameter grouping is from recognized industry sources such as NUREG/CR-6928, "Industry-average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." The basic event Type Codes were established independent of the Fire PRA Model; therefore, no data reanalysis is necessary.
DA-C3	See PRM-B13.	Current CC: Met All.

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		With respect to new components added to support Fire PRA modeling; basic event/parameter grouping is from recognized industry sources such as NUREG/CR-6928, "Industry-average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." The basic event Type Codes were established independent of the Fire PRA Model; therefore, no data reanalysis is necessary.
DA-C9	See PRM-B13.	Current CC: Met I/II. With respect to new components added to support Fire PRA modeling; basic event/parameter grouping is from recognized industry sources such as NUREG/CR-6928, "Industry-average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." The basic event Type Codes were established independent of the Fire PRA Model; therefore, no data reanalysis is necessary.
DA-C10	See PRM-B13.	Current CC: Met II. With respect to new components added to support Fire PRA modeling; basic event/parameter grouping is from recognized industry sources such as NUREG/CR-6928, "Industry-average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." The basic event Type Codes were established independent of the Fire PRA Model; therefore, no data reanalysis is necessary.
DA-D8	See PRM-B13.	Current CC: Met II. With respect to new components added to support Fire PRA modeling; basic event/parameter grouping is from recognized industry sources such as NUREG/CR-6928, "Industry-average Performance for Components and Initiating Events at U.S.

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		Commercial Nuclear Power Plants." The basic event Type Codes were established independent of the Fire PRA Model; therefore, no data reanalysis is necessary.
DA-E1	No documentation of data analysis for basic events added in fire PRA model.	Current CC: Met All. With respect to new components added to support Fire PRA modeling; basic event/parameter grouping is from recognized industry sources such as NUREG/CR-6928, "Industry-average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." The basic event Type Codes were established independent of the Fire PRA Model; therefore, no data reanalysis is necessary.
DA-E3	No documentation of data analysis for basic events added in fire PRA model.	Current CC: Met All. With respect to new components added to support Fire PRA modeling; basic event/parameter grouping is from recognized industry sources such as NUREG/CR-6928, "Industry-average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." The basic event Type Codes were established independent of the Fire PRA Model; therefore, no data reanalysis is necessary.
QU-E3	No quantitative estimate of CDF/LERF uncertainty intervals was performed for CNS.	Current CC: Met II. Calculation NEDC-09-086, "Task 7.15 Uncertainty and Sensitivity Analysis," which was the basis for the peer review, provides the uncertainty and sensitivity analyses. The following summarizes the philosophy for addressing the uncertainty quantitatively used in the calculation: From Section 1, A complete, quantitative analysis of uncertainties and assumptions is not conducted. This is, for the most part, consistent with the PRA Standard and referenced guidance documents, and

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		is consistent with prevailing good practices. The focus is on a qualitative characterization. Where appropriate sensitivity evaluations will be conducted when applying the FPRA. Table 3, in Section 6 discusses the approach to addressing parameter uncertainties. From the conclusions (Section 8), the results of this task are used to support integrated decision making. As has been noted, a quantitative characterization has not been developed as the quantitative results are conservatively biased for key contributors. A better estimate of the mean value for CDF and LERF is estimated to be a factor of 5 to 10 lower than calculated with a 90 percentile range of a factor of 10 on the lower end and 5 on the higher end. Relevant uncertainties will be addressed in the use of the FPRA. The major conservatisms are believed to be fire ignition frequency and the development and progression of a fire. Both increase the calculated frequency of core damage and large early release. Generic fire frequencies are directly based on assumptions in NUREG/CR-6850 (including FAQ-48 enhancements) estimating the severity and applicability of fires from incomplete fire event reports which lack critical information in the Fire Events Database. In the absence of complete data, the ignition frequencies remain conservative based on fires being assumed to be challenging fires, event duration being assumed the same as the actual fire duration, and lack of detailed damage information. These incomplete fire events provide the basis for the probability of non-suppression values for manual fire fighting. Additionally, the assumptions involving the growth and propagation of a fire including non-realistic peak heat release rates, cable flame spread rates, and cable tray propagation rule sets directly lead to reduction in effectiveness of

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		detection/suppression, and time available for operator action. Given the significant contributors to calculated results, the modeling and quantification of human reliability also can influence the results considerably. This occurs because for many of the significant sequences safe shutdown requires use of alternate shutdown which can involve many coordinated actions, and in some sequences with limited time margin for accomplishing the actions. State-of-the-art methods have been used but there is the potential for central tendency of the results to be either optimistic or pessimistic. The conservatism in fire growth time is expected to more than offset the potential for optimistic results to significantly influence calculated results." Attachment A to the calculation documents the approach for each related supporting requirement in the Standard. **Discussion:** As noted, an estimate of the uncertainty interval was provided. However, the peer review team finding is related to the technical basis for this estimate and that each of the technical areas contributing to this estimate was not explicitly addressed. Thus the following is provided: The start of the Fire PRA CDF and LERF calculations are the FDS and associated frequencies. More than 800 FDS were developed; combined with the cut-sets associated with each FDS, more than 100,000 sequences at the cut-set level were developed. The FDS include the ignition frequency and severity factor, detection and suppression conditional probabilities, and the fire modeling needed to develop these conditional probabilities. The cut-sets address data (DA), human reliability (HR), and circuit failure (CF). The DA parameter

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		uncertainties are addressed in the Internal Events PRA and are directly relevant to the Fire PRA. As noted in the calculation, "A better estimate of the mean value for CDF and LERF is estimated to be a factor of 5 to 10 lower than calculated with a 90 percentile range of a factor of 10 on the lower end and 5 on the higher end. Relevant uncertainties will be addressed in the use of the FPRA. The major conservatisms are believed to be fire ignition frequency and the development and progression of a fire. Both increase the calculated frequency of core damage and large early release." To support this estimate, consider as typical for a sequence which results in core damage or large early release: The Ignition Frequency uncertainties range factor is on the order of 10 based on NUREG/CR-6850. The estimated conditional probability of an FDS, given a fire ignition, has an uncertainty range factor on the order of 10 on the low side and 5 on the high side, based on judgment (includes fire development and propagation, and detection and suppression). The range factor for DA can be approximated by a factor of 3 to 5 based on the Internal Events PRA and typical uncertainties associated with unavailability and failure rates in recognition that the actual parameter uncertainty depends on the specific cut-sets. The range factor for HR can be approximated by a factor of 3 to 5 based on the HRA calculation. The range factor for CF can be approximated by a factor of 4 on the low side and 2 on the high side based on NUREG/CR-6850. The uncertainty on large early release given a core

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		damage end state has a range factor, again on average, on the order of 3 to 5 based on the Internal Events PRA and other PRA studies. Thus, typical significant CDF sequences can be represented by: CDF1 (where a random failure must occur after the FDS is reached, such as RCIC fails to start) = Ignition Frequency ("-10/+10") * Conditional FDS frequency ("-10/+5") * DA ("-5/+5") CDF2 (where a human error must occur after the FDS is reached, such as failure to realign a valve after inadvertent operation of the valve due to a spurious signal causes flow diversion) = Ignition Frequency ("-10/+10") * Conditional FDS probability ("-10/+5") * Conditional CF likelihood probability ("-4/+2") * HR failure probability ("-5/+5") Thus, as provided in the calculation, a reasonable estimate of the uncertainty interval is minus 10 to plus 5 on the calculated mean value, where the mean is estimated to be on the order of a factor of 5 to 10 lower than calculated.
LE-C7	Although there is no changes specific for fire, the existing HFEs should be evaluated in the context of fire.	Current CC: Met All. NEDC 09-079, "Task 7.5 Fire-Induced Risk Model," Attachment E (Attachment F in latest revision) is the Level 2 Human Failure Events Review. Those HFEs that are not post-fire operator actions have been denoted as such. Those that are post-fire operator actions make reference to NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis." All Attachment F HFEs were reviewed and LERF HFEs were verified to be included in the HRA report, NEDC 09-083.
PRM-B9	CNS system model changes in the fire PRA models are summarized in multiple reports. However, these changes are considered as temporary until	Current CC: Met All. CNS system model changes in the Fire PRA models

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SR	Topic	Resolution
	the final quantification is complete. A number of supporting requirements under SY-A (SYA2, 3, 4, 6, 12, and 24) are not met as a result. Therefore, PRM-B9 is considered not met. See F&Os 1-15, 2-3, 2-6, 2-13, 4-5, 4-11, and 4-12 for more details.	are summarized in multiple reports and these changes are now considered as final with the final quantification now complete. The purpose of NEDC 09-079 "Task 7.5 Fire-Induced Risk Model" was to develop a risk logic model to enable identification and quantification of all CDF and LERF sequences that could result from a fire initiating event. The internal events model was used as the foundation for the Fire PRA model. The Fire PRA model includes fire-induced impacts on the systems, trains, and components modeled in the internal events fault tree logic. Fire PRA modeling was performed in a manner that maintains the integrity of the internal events fault tree model and allows for proper evaluation of both the Internal Events PRA and the Fire PRA. To evaluate internal events, the fire initiators are all set to the default value of 0.0 and fire events are set to either 0.0 or 1.0 depending on how they input to their parent gate. For example, if a fire event is under an "OR" gate, it would be set to 0.0 for the internal events cases. PRAQuant solves each fire scenario by setting all internal events initiators to 0.0 and setting fire initiators and those basic events representing components impacted by the fire to 1.0. As such, Fire PRA modeling does not represent a change to the internal events modeling and appropriately is not included in the internal events documentation (e.g., system notebooks). Fire-induced failures (or new components) and supporting fault tree logic such as a check-valve internal leakage (ISLOCA logic), spurious operations, and/or new failure modes not included in the internal events modeling are documented in NEDC 09-079 "Task 7.5 Fire-Induced Risk Model", Attachment D "New Components and New Random Failure Modes Added to Support Fire PRA."

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SR	Topic	Resolution
		The Fire PRA model considers the impact of a fire on active and passive failures. For each active failure, a corresponding fire-induced failure is modeled. If a passive failure (i.e., a spurious operation) could place the component in an undesirable configuration, the passive failure is modeled as well. However, if a passive failure places a component in an acceptable configuration, modeling is not included because the spurious operation would be similar to modeling a "failure" as a "success." For example, for a valve Fails-To-Close in the internal events model; the fire model would have a corresponding, Fails-To-Close Due-To-Fire basic event. There would not be a passive failure (Spurious Close) for this valve in this branch of the fault tree, as the success position is "closed." The fire scenarios were further reviewed and additional detailed analysis was performed. This included not only detailed fire modeling and fire human reliability analysis, but also detailed circuit analysis and circuit failure likelihood analysis. The post-fire human reliability analysis considers operator actions (human failure events) as needed for safe-shutdown, including those called out in relevant fire response procedures. The post-fire human actions added to the Fire PRA model include the human performance shaping factors associated with a fire and are not applicable to an internal events non-fire initiator.
PRM-B11	CNS Calculation NEDC 09-083 (Scientech Calculation 17712-003) describes the operator actions included in the model. Included in the documentation are those internal events operator actions not credited in the Fire PRA and the additional human actions from CNS Procedure 5.4POST-FIRE included in the Fire PRA. The HRA Calculator was used to determine the HEPs for both existing internal events human actions and new fire response human actions. NEDC 09-078 R0 (Scientech Calc 17712-001), FIRE PRA COMPONENT	Current CC: Met All. NEDC-09-083, "Task 7.12 Fire Human Reliability Analysis," describes the operator actions included in the model. Included in the documentation are those internal events operator actions not credited in the Fire PRA and the additional human actions from CNS Procedure 5.4POST-FIRE included in the Fire

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
	SELECTION, Attachments E and F include a review of instrumentation and Manual Action Review, which covers the operator interface dependencies across systems or trains, where applicable. However, the Level 2 HFEs were not evaluated adequately. See F&Os 4-9 and 4-10 for more details. As a result, PRM-B11 is considered not met.	PRA. The HRA Calculator was used to determine the HEPs for both existing internal events human actions and new fire response human actions. NEDC 09-078, "Tasks 7.2 Component Selection," Attachments E and F include a review of instrumentation and Manual Action Review, which covers the operator interface dependencies across systems or trains, where applicable. Existing EOP operator actions were identified from a review of the Internal Events HRA analysis. The fire impacts due to fire damage to instrumentation are identified as part of NEDC 09-078, "Tasks 7.2 Component Selection," and NEDC 09-075, "Task 7.3 Cable Selection/Location." For any existing operator action where the instrumentation cables were not traced the HFE was set to 1.0. It was assumed that these instruments would be unavailable for every fire and with no instrumentation available for diagnosis the HEP is 1.0. Table 5 in NEDC-09-083, "Task 7.12 Fire Human Reliability Analysis," shows the HFEs retained in the Fire PRA and the instrumentation required for diagnosis for each HFE. NEDC 09-079, "Task 7.5 Fire-Induced Risk Model," Attachment E (Attachment F in latest revision), is the Level 2 Human Failure Events Review. Those HFEs that are not post-fire operator actions have been denoted as such. Those that are post-fire operator actions make reference to NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis." The four HFEs identified in F&O 4-9 are not in the HRA report because they are Level 2 events, not LERF events. All Attachment F HFEs were reviewed and LERF HFEs were verified to be included in the HRA report, NEDC 09-083.
PRM-B13	This refers to component failure probabilities. Based on the CNS self assessment, no component failure probability values, excluding spurious	Current CC: Met All. NEDC 09-079, "Risk Model Development,"

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
	operation, needed reanalysis for the Fire PRA. However, a query of BE table in the cnsone.rr database shows that several events with a style of 'fire' may be added in the fire models and require DA reanalysis: CRD-SOVCC-SO140A, CRD-SOV-CC-SO140B, LCS-CKV-LK-18CV, LCS-CKV-LK-19CV, RCI-CKV-LK-18CV, RCI-CKV-LK-19CV, RHRCKV-LK-26CV, RHR-CKV-LK-27CV, EAC-DG1-OVERLOAD, EAC-DG2-OVERLOAD. On the other hand, the expanded feedwater model and other additional system model changes for the fire PRA models may warrant more DA reanalysis. Human error probabilities were re-evaluated given a fire in Task 7.12 Calculation NEDC 09-083 (Scientech Calculation 17712-003). Spurious operations probabilities are documented in Task 7.10 Calculation NEDC 09-082 (Scientech Calculation 17712-007). PRM-B12 is considered met because the basic events were identified and B13 is considered not met because the data reanalysis was not performed.	documents new components and new failure modes that were added to the Fire PRA Model. Each new component and/or new failure mode was reviewed for System Analysis and/or Data Analysis, as appropriate to assess supporting level requirements. Component random failure probabilities use the values existing in the Internal Events parameter file. The new basic events evaluated are not fire-induced failures; therefore, the values used in the quantification of the Fire PRA Model are appropriate as a change would constitute a change to the Internal Events Model which is outside the scope of the Fire PRA. Regarding other Fire PRA Modeling changes such as fire-induced spurious operations and other fire-induced failures, the data was taken from NUREG/CR-6850 and fire initiator frequencies, respectively. As such, no data reanalysis is warranted.
FSS-A4	EPM-DP-FP-001 Rev. 1 provides the methodology for target identification. Target damage sets are provided in each fire compartment/zone detailed fire modeling calc (NEDC 09-091 through NEDC 09-101). Target damage sets for compartments modeled as full room burnout are provided in FPRA Level 1 failure report. However, a review of the top cutsets indicates there are opportunities for improvement. For example, the 3B and 3A zone failure evaluations would benefit from creating fire scenarios which would save the bus duct for the emergency transformer and in the case of 3B the cross power cables.	Current CC: Met All. Detailed fire modeling calculations have been developed for all fire zones deemed risk significant which include Fire Zones 3A (NEDC 10-046), 3B (NEDC 10-047), 8E (NEDC 10-043), 8F (NEDC 10-049), 8H (NEDC 10-043), 14A (NEDC 10-044), 14B (NEDC 10-045), and 13A (NEDC 09-095). These updates made to the Fire PRA in response to F&O 2-21 and 4-18 have incorporated combinations of target sets for each of these unscreened risk significant areas.
FSS-A5	EPM-DP-FP-001 Rev. 1 provides the methodology for target identification. Target damage sets are documented in each fire compartment/zone detailed fire modeling calc (NEDC 09-091 through NEDC 09-101). However, a review of the top cutsets indicates there are opportunities for improvement. For example, the 3B and 3A zone failure evaluations would	Current CC: Met I/II. Detailed fire modeling calculations have been developed for all fire zones deemed risk significant which include Fire Zones 3A (NEDC 10-046), 3B (NEDC 10-047), 8E (NEDC 10-043), 8F (NEDC 10-

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
	benefit from creating fire scenarios which would save the bus duct for the emergency transformer and in the case of 3B the cross power cables.	049), 8H (NEDC 10-043), 14A (NEDC 10-044), 14B (NEDC 10-045), and 13A (NEDC 09-095). These updates made to the Fire PRA in response to F&O 4-18 have incorporated combinations of target sets and combinations of ignition sources for each of these unscreened risk significant areas, and this SR is now considered to meet Cat I/II.
FSS-D1	The fire modeling tools selected are addressed in CNS documents R1906-711-01, R1906-07-011b-001, and EPM-DPFP-001. The fire modeling tools are adequate when used within their limitations. However, there are no specific limits beyond which the output from the fire modeling tools become invalid. Two cases where this arises is with hot gas layer effects coupled with localized fire exposure effects (plume-layer/thermal radiation-layer) and when the postulated flame height exceeds the ceiling height. In the latter case, the radiant heat flux model would be invalid. The smoke detection model appears to be critical for application of the severity factor and non-suppression probabilities; however, this model has not been through a formal verification and validation process.	Current CC: Met All. Updates made to the Fire PRA in response to F&Os 3-1, 3-9, 3-12, and 3-13 justify that the appropriate fire modeling tools are used within their limitations through a formal verification and validation process.
FSS-D2	CNS documents R1906-711-01, R1906-07-011b-001, and EPM-DP-FP-001. The detailed fire model tools are validated in CNS document R1906-711-01 and the analysis bases are provided.	Current CC: Met All. Updates made to the Fire PRA in response to F&Os 3-1, 3-9, and 3-12, and 3-13 justify that the appropriate fire modeling tools are used within their limitations through a formal verification and validation process.
IGN-A5	Ignition frequencies are calculated on a reactor year basis in NEDC 08-032; however, they are not weighted by the fraction of time that the plant is at power. This can lead to an overestimate on the order of ten percent for the frequencies.	Current CC: Met All. The Ignition Frequency Calculation NEDC 08-032 was updated to include the Average Criticality Factor from CNS-PSA-001, "Initiating Event Notebook." This value is used to convert the initiating event frequencies for this PSA update from critical to calendar years. This criticality factor is considered representative of the continual improvement in plant operation and appropriately represents future operation. Each of the Bin generic ignition frequencies identified in NEDC 08-032 has been updated to reflect plant-specific values weighted by

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		the fraction of time that the plant is at power.
IGN-B1	The total ignition frequencies are provided for the PAUs and for the fire zones in NEDC 08-032 Tables A-1 and A-2 of Appendix A. The frequencies for the specific ignition source bins in each PAU/fire zone used for other tasks are not provided in this document but they are contained in the ignition source data sheets. These frequencies should be documented in a report or calculation.	Current CC: Met All. The Ignition Frequency Calculation NEDC 08-032 was updated to include the information directly from the Microsoft Excel spreadsheet used to perform the calculations to prevent typos in reproducing the data in the Microsoft Word document.
HRA-D2	Calculation NEDC 09-083 (Scientech Calculation 17712-003) "Post-Fire Human Reliability Analysis" addresses this issue. Recovery actions are only incorporated if they are existing EOP actions or if they are Fire Response actions. Existing EOP actions are identified in Table 4-1. Table 4-2 lists Fire Response HFEs Included In the FPRA. HFE mapping and treatment are documented in tables 4-3a/b/c/d, and table 4-4 through 4-9. Detailed HFE analysis is documented in Attachment B. Operator interviews are summarized in Section 6.4 and described in Attachment A. However, since HR-H3 is considered not met due to dependency analysis not being complete, this SR is considered not met.	Current CC: Met All. Dependency Analysis was completed in NEDC 09-083, "Task 7.12 Human Reliability Analysis," and is summarized in NEDC 09-085 "Task 7.14 Fire Risk Quantification," as follows: Dependencies among operator actions exist for any of the following parameters that may have a common effect on operator actions including timing, procedures, cues, personnel, and staffing resources. Dependencies arise from post-initiator human error events that may be linked in the quantification process. If these operator actions are not independent, then any sequence cutsets associated with two or more of these dependent operator actions would be incorrect. The dependency analysis was performed in Task 7.12 using the EPRI HRA Calculator. The dependencies were identified and examined in the analysis by identifying combinations of operator actions and then determining their level of dependence.
UNC-A1	No quantitative estimated of parameter uncertainty was performed. See F&O 1-17. Qualitative evaluations of assumptions and uncertainty is provided in the uncertainty analysis NEDC 09-086 FPRA Uncertainty and Sensitivity r0.pdf.	Current CC: Met All. See SR QU-E3
QU-D7	The FPRA does not include a development of a single cutset report or a list of importance measures for SSCs, Operator Actions or FPRA basic events.	Current CC: Met All. Results were reviewed in detail and determined to

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		make logical sense. This finding does not affect the results and use of the Fire PRA in fire risk evaluations. The calculation documentation was enhanced to clarify the definition of risk significance, and that all risk significant sequences at the fire damage state combined with CCDP and CLERP level are included in an attachment to the calculation. The CNS Fire PRA is extremely detailed in order to account for sequence dependent basic event values including spurious operations and human interaction.
QU-F3	Significant initiating events and accident sequences have been documented in the fire quantification report Calculation 17712-011, Task 7.14 Fire Risk Quantification. A detailed description of significant accident sequences has been documented. However, due to the lack of single merged cutset files, significant basic events have not been documented.	Current CC: Met I to II/III. Capability Category I is acceptable for this application as the actual results are not influenced by not achieving Capability Category II. The contributors at any level can be determined from a review of the cut-sets, including basic events. Importance ranking at the basis event level were not developed. The significant contributors at the sequence level are discussed.
QU-F6	The quantitative definition used for significant basic event, significant cutset, and significant accident sequence is not documented.	Current CC: Met All. This finding does not affect the results and use of the Fire PRA in fire risk evaluations. NEDC 09-085, "Task 7.14 Fire Risk Quantification," was enhanced to clarify the definition of risk significance, and that all risk significant sequences at the fire damage state combined with CCDP and CLERP level are included in an attachment to the calculation. The CNS Fire PRA is extremely detailed in order to account for sequence dependent basic event values, including spurious operations and human interaction.
LE-G6	The quantitative definition used for significant basic event, significant cutset, and significant accident sequence is not documented.	Current CC: Met All. This finding does not affect the results and use of the Fire PRA in fire risk evaluations. NEDC 09-085, "Task 7.14 Fire Risk Quantification," documentation

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SR	Topic	Resolution
		was enhanced to clarify the definition of risk significance and that all risk significant sequences at the fire damage state combined with CCDP and CLERP level are included in an attachment to the calculation. The CNS Fire PRA is extremely detailed in order to account for sequence dependent basic event values including spurious operations and human interaction.
HR-E3	Operator interviews are documented in Attachment A of the HRA notebook. This includes talk-through of some of the FPRA actions, including walk-through on the simulator. However, a talk-through of all HEPs was not performed.	Current CC: Met II/III. NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis," was updated after further discussion of selected dominant scenarios and their associated operator actions. These actions were discussed in additional operator interviews. Also, a talk-through of actions was conducted.
HR-E4	Operator interviews are documented in Attachment A of the HRA notebook. This includes talk-through of some of the FPRA actions, including walk-through on the simulator. However, a talk-through of all HEPs was not performed.	Current CC: Met II/III. NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis," was updated after further discussion of selected dominant scenarios and their associated operator actions. These actions were discussed in additional operator interviews. Also, a talk-through of actions was conducted.
HR-G5	Operator action timing was based on walkthroughs, or talk-through, including simulator observations. Internal events PRA HEP actions used the internal events timing. For new actions, the actions were timed by the Appendix R feasibility analysis, as documented in Appendix M of the Appendix R analysis. However, not all HEPs were talked through with operations.	Current CC: Met II. NEDC 09-083 "Task 7.12 Fire Human Reliability Analysis" was updated after further discussion of selected dominant scenarios and their associated operator actions. These actions were discussed in additional operator interviews. Also, a talk through of actions was conducted.
DA-D1	See PRM-B13.	Current CC: Met II. With respect to new components added to support Fire PRA modeling, basic event/parameter grouping is from recognized industry sources such as NUREG/CR-6928, "Industry-average Performance

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SR	Topic	Resolution
		for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." The basic event Type Codes were established independent of the Fire PRA Model; therefore, no data reanalysis is necessary.
PP-B2	The justification of partitioning between fire zone barriers to prevent full room burn-up not provided in some areas. These include 8H/8E, 8F/8E, 11A/11B, and 14A/14C.	Current CC: Met II/III. Plant Boundary Definition and Partitioning Calculation NEDC 10-004 was updated to address fire compartments with multiple fire zones in which the fire zone boundaries may have been credited and whole room burnout approach taken to "screen" fire zones from detailed fire modeling. A review of the boundaries of the "screened" fire zones was performed via plant walkdown in order to confirm that these barriers are substantial enough to preclude fire spread to adjacent fire zones within the fire compartment. Detailed assessment of these barriers is provided in Multi-Compartment Analysis.
FSS-C1	An effective one point heat release rate model is used in the detailed fire modeling analyses (NEDC-091 - NEDC-101), except for MCR abandonment calculation (R1906-07-011b-001), which uses a fifteen point heat release rate model. The one point model approach typically identifies a threshold fire size and evaluates the time to damage given a 98th percentile growing fire.	Current CC: Met II. The HRR distribution for each scenario has been discretized into two points in the Detailed Fire Modeling Calculations. The first point corresponds to the minimum HRR required to damage the nearest target, and its fire severity factor (SF) represents the fraction of fires that will damage only the ignition source itself. The second point corresponds to the 98^{th} percentile HRR, and its SF represents the fraction of fires that will damage all targets within the Zone of Influence (ZOI) of the 98^{th} percentile HRR, excluding the fraction of fires that will damage only the ignition source itself.
FSS-D3	EPM-DP-FP-001 Rev. 1 provides the methodology detailed fire modeling of potentially risk significant compartments. Compartment/zone detailed fire modeling are provided in calculations NEDC 09-091 through NEDC 09-101 and NEDC 10-001 Rev. 0. A review of the top cutsets indicates there are opportunities for improvement through the use of detailed fire modeling. For	Current CC: Met II. Detailed fire modeling calculations have been developed for all fire zones deemed risk significant which include Fire Zones 3A (NEDC 10-046), 3B

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SR	Topic	Resolution
	example, the 3B and 3A zone failure evaluations would benefit from creating fire scenarios which would save the bus duct for the emergency transformer and in the case of 3B the cross power cables.	(NEDC 10-047), 8E (NEDC 10-043), 8F (NEDC 10-049), 8H (NEDC 10-043), 14A (NEDC 10-044), 14B (NEDC 10-045), and 13A (NEDC 09-095). These updates made to the Fire PRA in response to F&O 4-18 have incorporated detailed fire modeling scenario development for each of these unscreened risk significant areas and this SR is now considered to meet Cat II.
FSS-E3	The uncertainty parameters are characterized in Section 7.3 of the detailed fire modeling calculations NEDC 09-091 through NEDC 09-101. Mean values and distributions for these parameters are not established.	Current CC: Met I. A consensus approach for meeting Capability Category II is not available. Prevailing good practice addresses this supporting requirement qualitatively, and thus is intended to meet Capability Category I. A reasonable "qualitative" characterization of the conditional probability of a FDS given a fire is a 90% range of -10 to +5 on the calculated point estimate. This approach is sufficient for fire risk evaluations. Where appropriate sensitivity evaluations are considered.
IGN-A10	The characterization of uncertainty intervals is qualitatively discussed in the uncertainty analysis NEDC 09-086 FPRA Uncertainty and Sensitivity r0.pdf. A quantitative or statistical uncertainty estimate for each areas or scenario is not provided.	Current CC: Met II. See SR QU-E3.
CF-A1	Initial quantification of circuit failures is set to a generic industry value of 0.3 or 0.6. When significant, specific circuit analysis is performed, this is documented in the Task 10 report. Table B of the Task 10 report is derived from the Task 9 report. Circuit failure likelihood is then developed in Table E-1 for each cable based on the generic SO probabilities, and the cable type, failure mode, etc. The specific circuit configuration for cables is included in the consideration. However, the plant specific analysis was not performed for all significant SO events. For example, in 3B, there are 5 events with FV> 0.01 that contain generic probabilities (0.6 or 0.3). In this area, all cables appear to be in conduit, which would likely result in a lower spurious operation probability for these events (unless SO is inside of an electrical cabinet). As a result, CC I is considered met, while CC II is considered not met.	Current CC: Met II/III. The fire scenarios were further reviewed and additional detailed analysis was performed. This included not only detailed fire modeling and fire human reliability analysis, but also detailed circuit analysis and circuit failure likelihood analysis. Significant fire-induced failures of these scenarios were reviewed. The results of the detailed circuit analysis and circuit failure likelihood analysis are found in Task 9 (Calculation NEDC 09-073) and Task 10 (Calculation

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SR	Topic	Resolution
		NEDC 09-082).
HRA-A4	Although there was a general over view of operational philosophy, there was not a talk-through of each new operator action.	Current CC: Met II/III. NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis," was updated after further discussion of selected dominant scenarios and their associated operator actions. These actions were discussed in additional operator interviews. Also, a talk-through of actions was conducted.
HRA-C1	Detailed HRA was quantified in most cases for HEPs, accounting for fire-related effects. All SRs under HR-G were considered met other than the dependency SR HR-G6. Several F&Os were identified on these referenced SRs.	Current CC: Met II. NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis," documents the completed detailed HRA analysis for those HFEs in the FPRA as required. Consistency check of Fire PRA HFEs was completed and is documented in NEDC 09-083 "Task 7.12 Fire Human Reliability Analysis." This consistency check included such items as timing, use of appropriate HRA analysis method, scenario descriptions, procedures, and performance shaping factors. The fire scenarios were further reviewed and additional detailed analysis was performed if possible. This included not only detailed fire modeling and fire human reliability analysis, but also detailed circuit analysis and circuit failure likelihood analysis. A Dependency Analysis was completed in NEDC 09-083, "Task 7.12 Human Reliability Analysis," and is summarized in NEDC 09-085, "Task 7.14 Fire Risk Quantification," as follows: Dependencies among operator actions exist for any of the following parameters that may have a common effect on operator actions, including timing, procedures, cues, personnel, and staffing resources. Dependencies arise from post-initiator human error events that may be linked in the quantification

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SR	Topic	Resolution
		process. If these operator actions are not independent, then any sequence cutsets associated with two or more of these dependent operator actions would be incorrect. The Dependency Analysis was performed in Task 7.12 using the EPRI HRA Calculator. The dependencies were identified and examined in the analysis by identifying combinations of operator actions and then determining their level of dependence. During the second interview with operators, effects of fires on travel paths were specifically reviewed and no significant change in travel time could be determined to occur. No further action is required. This is documented in NEDC 09-083, "Task 7.12 Fire Human Reliability Analysis."
QU-D6	Section 7.2 of the Quantification Analysis provides a discussion of all significant accident sequences in the FPRA. Discussion involves verification that the results are reasonable and accurate. Multi-compartment scenarios are reviewed in Section 7.3 and MCB scenarios are reviewed in Section 7.4. The FPRA model relies heavily on the internal events model logic, which was reviewed for consistency, including operational consistency. Operational consistency for fire includes the verification for scenarios that the equipment damaged is located in the compartment (cables or components). Additionally, review of HFEs is performed as part of the HRA for the FPRA. However, the FPRA does not include a development of a single cutset report or a list of importance measures for SSCs, Operator Actions or FPRA basic events. Additionally, the listing of the significant accident sequences does not list all significant accident sequences or significant SSCs as required by this SR (CC II).	Current CC: Met I to II/III. Capability Category I is acceptable for this application as the actual results are not influenced by not achieving Capability Category II. The contributors at any level can be determined from a review of the cut-sets, including basic events. Importance ranking at the basis event level were not developed. The significant contributors at the sequence level are discussed.
LE-G3	CNS Calculation 17712-011, Task 7.14 Fire Risk Quantification, Section 7 and Attachment B document the significant contributors to LERF. However, the relative contribution of contributors (i.e., plant damage states, accident progression sequences, phenomena, containment challenges, containment failure modes) to LERF is not documented.	Current CC: Met I to II/III. Capability Category I is acceptable for this application as the actual results are not influenced by not achieving Capability Category II. The contributors at any level can be determined from a review of the cut-sets, including basic events. Importance ranking at the basis event level were not developed. The

Table V-2 Fire PRA - Category I Summary

SR	Topic	Resolution
		significant contributors at the sequence level are discussed.

ATTACHMENT W

Fire PRA Insights

26 Pages

W.1 Fire PRA Overall Risk Insights

Risk insights were documented as part of the development of the Cooper Nuclear Station (CNS) Fire Probabilistic Risk Assessment (FPRA). The calculated fire core damage frequency/large early release frequency (CDF/LERF) were derived using NUREG/CR-6850 methodology for FPRA development. The results are useful in understanding risk and risk contributors on a relative basis among the areas of the plant. The risk insights generated were useful in identifying areas where specific contributors might be reduced via plant modification. A description of significant risk sequences associated with sequences above 1% contribution to the calculated Fire CDF was prepared for the purposes of gaining these insights and developing an understanding of the risk significance of multiple spurious operation (MSO) combinations. These sequences and insights are provided in Table W-1 for the Post NFPA 805 plant model.

Using the definition of "significant" from the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard (for the term *significant accident sequence*) the fire initiating events that sum to 95% of the collective CDF or those whose contribution is more than 1% of the total Fire CDF are considered to represent the significant fire scenarios. Approximately 900 fire damage states (FDS) were quantified in the intra-compartment analyses, including the Control Room FDS. Approximately 1300 scenarios were addressed in the multi-compartment analysis (MCA). The contributors to the calculated total CDF are as follows (note that contributors to LERF have similar insights): ~25 FDS contribute > 1%; to achieve 95% contribution more than 200 FDS must be considered, with the 200th ranking FDS having a CDF contribution of < 0.1%.

W.2 Risk Change Due to NFPA 805 Transition

In accordance with the guidance in Regulatory Position 2.2.4.2 of Regulatory Guide (RG) 1.205 Revision 1:

> *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.*

Table W-2 provides the risk increases and decreases on a fire area basis. The transition to National Fire Protection Association (NFPA) 805 resulted in a collective calculated risk decrease (CDF/LERF). This risk decrease was achieved by modifications to resolve selected VFDR and targeted, risk-informed modifications beyond those that would result in compliance (Changes Beyond Compliance). An example of a risk-informed modification is providing for both offsite power and emergency power from a diesel generator (DG) where power solely from the DG would result in compliance.

There is a net decrease in total CDF and LERF associated with this application. Therefore, these changes meet the RG 1.174 acceptance guidelines.

RG 1.174 does not require calculation of total CDF and LERF if the increases are below the delta CDF and delta LERF of 1E-06/yr and 1E-07/yr respectively. However, it does recommend that if there is an indication that the CDF is 'considerably higher' than 1E-04/yr or if LERF is 'considerably higher' than 1E-05/yr then the focus should be on finding ways to decrease rather than increase CDF or LERF. Although there is a net risk decrease, an estimate of the total calculated CDF and LERF is provided below for perspective.

The total calculated Fire CDF and LERF (Post NFPA 805), including multi-compartment scenarios, are 5.2E-5/year and 1.2E-5/year, respectively. These calculated values are estimated to be conservative by a factor of 5 to 10 as review of uncertainties associated with Fire PRA tasks indicates a conservative bias of the calculated mean values. Thus, better estimates of CDF and LERF are <~1E-5/year and <~2E-6/year, respectively. Internal Events PRA including Internal Flooding have total calculated values for CDF and LERF of 7.0E-06/year and 2.3E-06/year, respectively. The External Events CDF and LERF were not quantitatively developed in the Individual Plant Examination – External Events (IPEEE). External events other than seismic screened, based primarily on Standard Review Plan (SRP) guidance, and thus their total risk is on the order of or less than 1E-6/year for both CDF and LERF. For seismic events, a seismic margins analysis (SMA) was performed as part of the IPEEE. The risk is estimated to be on the order of 1E-5/year for CDF and less for LERF based on screening analyses conducted by NRC in support of Generic Issue 199. An assessment of seismic-fire interactions following the methodology in NUREG/CR-6850 and using information from the IPEEE concluded that no significant seismic-fire interactions exist. In addition, no specific vulnerabilities to seismically induced ignition sources were identified, and installed fire suppression features are likely to be available and remain operational after a seismic event.

Given these factors, the following criteria for total delta risk are judged to be appropriate: delta CDF of 1E-5/year as the total CDF is reasonably estimated to be below 1E-4/year, and delta LERF of 1E-6/year as the total LERF is reasonably estimated to be below 1E-5/year.

Most importantly, as noted above, the calculated delta risk is negative. That is, safety was improved beyond the level of a compliant plant, as a result of risk-informed modifications such as making offsite power available for two fire areas, installing incipient detection for two panels, establishing enhanced transient and hot work controlled zones, and other changes as noted in Attachment S.

W.3 Recovery Actions

Recovery Actions (RA) were considered in the FPRA. The risk of RA is included in the delta risk results (see Table W-2). As shown, this delta risk is more than offset by risk reductions due to the risk-informed changes identified as a part of the evaluation.

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF

Scenario	Description	CDF Contribution (%)	Risk Insights	IF	CCDP	CDF (per year)
14A D-1 F0	This is a fire of the Division 1 Diesel Generator (DG1) in Zone 14A. In addition to DG1, cables for the Division 1 diesel generator output breaker EG1 are impacted by the fire. Power to Bus 1G is automatically available from Bus 1B and the Division 2 Diesel Generator (DG2). It is assumed that the fire can cause a spurious start of DG1 and spurious closure of breaker EG1 before the fire destroys DG1. This could cause asynchronous loading of the DG1 to Bus 1F, which would be aligned to offsite power via breaker 1FS and the Emergency Station Service Transformer. This renders Bus 1F non-recoverable. High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) are available for high pressure RPV level control. Core Spray Train B and the LPCI Mode of Residual Heat Removal (RHR) Train B with Pump D are available for low pressure Reactor Pressure Vessel (RPV) level control if needed. RHR Train B is available for decay heat removal first in Suppression Pool Cooling (SPC) mode and subsequently in the Shutdown Cooling (SDC) mode of operation.	7.8	Dominant contributors are asynchronous loading of DG1 to Bus 1F and test/maintenance of RHR Pump D.	2.01E-03	1.92E-03	3.85E-06

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
11F TS1	This is a transient fire (TS1) located in Turbine Building Controlled corridor at Elevation 882'. Cables for the supply breakers to Bus 1A and Bus 1B, cables to the Bus 1F breaker 1FS, cables to the Bus 1G breaker 1GS, and cables for the DG2 output breaker EG2 are impacted by the fire. The Emergency Station Service Transformer and associated cables are not impacted by the fire with the exception of breakers 1FS and 1GS. All power from non-safety Buses 1A and 1B is lost and cannot be recovered. Cable X10 for EE-CB-4160F-1FS (Emergency Transformer Supply to Bus 1F) and EE-CB-4160G-1GS (Emergency Transformer Supply to Bus 1G) will only cause spuriously-opens or fails-to-close functional failures. The failure mode spuriously-closes is not impacted by the fire. Given a fire, offsite power is lost, the diesel generators receive a START signal, and breakers 1FS and 1GS receive CLOSE signals. Because of cable X10, breakers 1FS and 1GS, however, will not close. DG1 will load automatically to Bus 1F. Given field actions, the Emergency Transformer can also provide power to both	3.6	Dominant contributors are random failures of DG1, Start-Up Station Service Transformer, and RHR Pump A.	1.19E-04	1.48E-02	1.77E-06

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	Bus 1F and 1G, and DG2 can be loaded onto Bus 1G, but these actions were not credited for this zone.					
13B BD03	This is a fire of the bus duct (BD03) located in the Non-Critical Switchgear Room. The 2000 amp bus ducts to Buses 1A and 1B, cables for the Bus 1A breaker 1AE, Bus 1B, cables for the Bus 1G breakers 1GB, 1GE, and 1GS, cables for the Bus 1G undervoltage circuit, and cables for the DG2 output breaker EG2 are impacted by the fire. The Emergency Station Service Transformer and associated cables are not impacted by the fire with the exception of breaker 1GS. Given a fire, all power from non-safety Buses 1A and 1B is lost and cannot be recovered, the diesel generators receive a START signal, and breakers 1FS and 1GS receive CLOSE signals. A manual action from the Control Room is needed to open breaker 1FS because the Bus 1G is connected to the faulted Bus 1B (via breakers 1BG and 1GB) and the Emergency Transformer is connected to Bus 1G (via breaker 1GS). Once breaker 1FS is open, Bus 1F is loaded from DG1. Given field actions, Bus 1G can also be	3.3	Dominant contributors are the test/maintenance unavailabilities of DG1 and RHR Pump 1A.	3.58E-05	4.61E-02	1.65E-06

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	Description	CDF Contribution (%)	Risk Insights	IF	CCDP	CDF (per year)
	recovered. Operator field actions to align Bus 1G to either DG2 or the Emergency Station Service Transformer are included when determining the risk for scenarios from Zone 13B.					
8G CHG125B FD2	Scenario 8G CHG125B FD2 is a fire of 125VDC Battery Charger 1B located in Direct Current (DC) Switchgear Room 1B. The fire causes a loss of all Division 2 DC power. Scenario 8G CHG125B FD2 also impacts the supply and exhaust fan of Train A of the Essential Control Building Ventilation system. This system may be required to support Division 1 DC power 1A in ~9 hours and is not a significant contributor to risk. Given a fire, the reactor trips and Condensate/Feedwater is assumed to be lost. All pumps supported by Bus 1G are not available without field actions. HPCI relies on Division 2 DC power and is not available. The Core Spray Train B and RHR Train B flowpaths, and RHR Pump C, are lost. RCIC is available to maintain RPV level. Only RHR Pump A is available for decay heat removal in the SPC mode.	1.8	Risk is dominated by random failures of RHR Train A.	6.55E-05	1.38E-02	9.06E-07
8G	This scenario is a fire of 250VDC Battery	1.8	The effects and insights are the	6.55E-05	1.38E-02	9.06E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
CHG250B FD2	Charger 1B located in DC Switchgear Room 1B.		same as for 8G CHG125B FD2, except for the impacts on ventilation.			
22A ZONE FAILURE	22A ZONE FAILURE is a full zone burn-up of the Augmented Radwaste Building basement. This is a non-critical area that has no essential safe shutdown equipment. This zone and similar non-critical areas are modeled at the level of ZONE FAILURE; i.e., full zone burnup. All safe shutdown systems are available after the fire. The Fire PRA assumes that a fire in this area will result in a reactor trip and loss of Condensate/Feedwater. No other Fire PRA systems are impacted. Because of the nature of this area, however, it is entirely possible that the plant would continue to operate and remain on line, thereby resulting in a much lower CDF for a fire in this zone.	1.8	The conditional core damage probability (CCDP) and conditional large early release probability (CLERP) for this zone are the baseline, e.g., plant transient, no Condensate/Feedwater, all safety systems available.	1.18E-02	7.54E-05	8.92E-07
10A ZONE FAILURE	Scenario 10A ZONE FAILURE is a full zone burn-up of the Computer Room and is part of the Control Room Envelope, Fire Area CB-D. For fires in Fire Area CB-D, the fire response procedures 5.4POST-FIRE and 5.4FIRE-S/D credit Control Room abandonment and use Alternate Shutdown (ASD) to bring the plant to a safe stable condition. The calculated CCDP for a full zone burnup of Zone 10A is 6.6E-04. For this FDS, operators would have sufficient command and control from the Control	1.7	A fire in this room can potentially cause a hot short that could prevent an automatic scram. This, however, was not a significant contributor to risk. The dominant contributors to risk for this scenario are manual actions for RPV depressurization, Emergency Core Cooling System (ECCS) initiation, early SPC, and tripping the Reactor Recirculation pumps at the	1.25E-03	6.56E-04	8.18E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	Room so that they would not shutdown using ASD.		pump breakers.			
14B D-2 F0	This is a fire of DG2 in Zone 14B. In addition to DG2, cables for the DG2 output breaker EG2 are impacted by the fire. Power to Bus 1F is automatically available from Bus 1A and DG1. It is assumed that the fire can cause a spurious start of DG2 and spurious closure of breaker EG2 before the fire destroys the DG. This could cause asynchronous loading of DG2 to Bus 1G which would be aligned to offsite power via breaker 1GS and the Emergency Station Service Transformer. This renders Bus 1G non-recoverable. HPCI and RCIC are available for high pressure RPV level control. Core Spray Train A and the LPCI Mode of RHR Train B are available for low pressure RPV level control if needed. RHR Train A is available for decay heat removal first in SPC mode and subsequently in the SDC mode of operation.	1.6	Dominant contributors are asynchronous loading of DG1 to Bus 1F, test/maintenance of RHR Pump A, and failure by the operators to align the hard pipe vent for decay heat removal.	2.01E-03	4.03E-04	8.08E-07
8A PNL 9-41	This scenario is a fire of Panel 9-41 (Inboard Isolation Valve Relay Rack) in the	1.6	In addition to the loss of RHR, the risk is dominated by the	1.85E-05	4.33E-02	8.02E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	Auxiliary Relay Room. A fire in this panel causes a loss of all four RHR pumps, HPCI, and many of the inboard isolation valves. RCIC and Core Spray are not impacted and are available for mitigation. Given a fire a plant transient occurs and Condensate/Feedwater are assumed to be lost. RCIC receives a start signal to maintain RPV level. With no SPC, RCIC is not a long term success and the operators must depressurize and initiate Core Spray.		failure of the operators to depressurize the RPV to allow Core Spray to maintain RPV level and failure of the operators to align the hard pipe vent for decay heat removal. The use of ASD for this scenario was not credited.			
8A PNL 9-42	This scenario is a fire of Panel 9-42 (Outboard Isolation Valve Relay Rack) in the Auxiliary Relay Room. A fire in this panel causes a loss of all four RHR pumps and many of the outboard isolation valves. HPCI, RCIC, and Core Spray are not impacted and are available for mitigation. Given a fire a plant transient occurs and Condensate/Feedwater are assumed to be lost. RCIC receives a start signal to maintain RPV level. With no SPC, HPCI and RCIC are not a long term success and the operators must depressurize and initiate Core Spray.	1.6	In addition to the loss of RHR, the risk is dominated by the failure of the operators to depressurize the RPV to allow Core Spray to maintain RPV level and failure of the operators to align the hard pipe vent for decay heat removal. The use of ASD for this scenario was not credited.	1.85E-05	4.27E-02	7.90E-07
8H CHG250A	Scenario 8H CHG250A FD2 is a fire of 250VDC Battery Charger 1A located in DC	1.6	In addition to the loss of Bus 1F loads, risk is dominated by	7.55E-05	1.03E-02	7.81E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
FD2	Switchgear Room 1A. This scenario causes a loss of DC Panel AA1 (non-safety DC power), AA3 (Bus 1F breakers, RCIC Starter Rack), NBPP (No Break Power Panel), 250VDC Starter Rack Div. 1, and the RCIC 250VDC Starter Rack. Given a fire, the reactor trips and Condensate/Feedwater is lost. RCIC is not available and the pumps supported by Bus 1F are lost. HPCI is available to maintain RPV level. Only RHR Pump D is available for decay heat removal in SPC.		random failures of RHR Pump D.			
9A RPSP1B F1	The scenario is a fire of Panel RPSPP1B (Reactor Protection System Power Panel 1), located in the Cable Spreading Room. This scenario is a fire that propagates beyond RPSPP1B and impacts FDS1 targets. Bus 1G and associated loads are lost, HPCI is lost, and Motor Control Center (MCC) LX and MCC TX are lost. MCC LX provides power to the 125VDC and 250VDC 1A battery chargers. MCC TX provides power to the 125VDC and 250VDC 1B Battery Chargers. This results in all DC power being lost except for the short term provided by the batteries. Manual shutdown from the ASD Panel for	1.4	The calculated CCDP for a fire in Panel RPSPP1B is 1.0. Given a fire in this panel and based on 5.4POST-FIRE, operators would move command and control to the ASD Room and use 5.4FIRE-S/D to safely shutdown. Failure of the operators to safely shutdown with ASD is 0.1 giving a CCDP of 0.1 and a CDF of 6.92E-07/year.	6.92E-06	1.00E-01	6.92E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	all fires in Zone 9A is credited in the fire response Procedure 5.4POST-FIRE. The systems available when using ASD are HPCI for early RPV level control, DG2 for AC power, Safety Relief Valves (SRVs) 71E, 71F, and 71G for RPV depressurization, and RHR Train B for decay heat removal first in SPC mode and subsequently in SDC cooling mode of operation.					
9A RTD CAB B	This is a fire of RTD Cabinet B located in the Cable Spreading Room. A fire of this panel causes a loss of HPCI, RCIC, and both trains of Core Spray. RHR is available for RPV level control and SPC. Given a fire, a plant transient would be experienced and an assumed loss of Condensate/Feedwater. With no high pressure system available, the operators must depressurize the RPV and align RHR for RPV level control. They must also align RHR in SPC to provide decay heat removal.	1.3	The risk is dominated by the loss of HPCI, RCIC, and Core Spray with the failure of the operators to depressurize the RPV and align RHR for RPV level control and SPC.	1.85E-05	3.54E-02	6.55E-07
8G SWG125B-FD0	This is a fire of 125VDC Switchgear 1B with equipment that terminates at the switchgear being impacted. 125VDC Switchgear 1B is located in the DC Switchgear Room 1B, 903'6" Elevation of the Control Building. 125VDC Battery 1B, 125VDC Battery	1.3	In addition to the loss of 125VDC, the risk is dominated by random failures of RHR Train A.	4.48E-05	1.38E-02	6.20E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF

Scenario	Description	CDF Contribution (%)	Risk Insights	IF	CCDP	CDF (per year)
	Charger 1B, 125VDC Switchgear 1B, 125VDC Starter Rack 1B, 125VDC RCIC Starter Rack, and 125VDC Reactor Building Starter Rack are all lost. Given a fire, a plant transient would be experienced and an assumed loss of Condensate/Feedwater. RCIC is lost. However, HPCI is available for RPV level control. All Division 2 systems are also available.					
2A-1 MCC K	This scenario is a fire of MCC K. MCC K is the critical MCC supplying Division 1 loads such as Reactor Equipment Cooling (REC) Pumps 1A and 1B, Standby Liquid Control (SLC) Pump 1A, power feed to MCC Q, and the normal power feeds to MCC R and MCC RA. Loss of MCC K would result in loss of the components noted above. The RCIC turbine trip and throttle valve is also failed, rendering RCIC unavailable. The Northeast and Northwest Quad fan coil units are also lost. Given a fire, a plant transient would be experienced and an assumed loss of Condensate/Feedwater. Although both HPCI and RCIC receive start signals, only HPCI would be available to maintain RPV level. RHR Train A is supported by MCC Q and would not be available for SPC.	1.2	Risk is dominated by HPCI and Service Water Pump B random failures and failure of the operators to depressurize the RPV and initiate SPC.	1.67E-04	3.61E-03	6.02E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	However, RHR Train B is available for mitigation.					
3A 4160F FDS0	Scenario 3A 4160F FDS0 is a fire in Bus 1F impacting the FDS0 targets. Zone 3A differs from its counterpart Zone 3B in that cables for Bus 1G breakers do not pass through Zone 3A. Offsite power to Bus 1G from the Emergency Station Service Transformer is, however, lost because the Emergency Transformer 2000 amp bus duct is failed by a fire in Zone 3A. HPCI and RCIC are not failed by a fire in this zone, but do require SPC for success (manual action RHR-XHE-FO-RHRE). Train A of RHR is supported by Bus 1F and is lost. RHR Pump B is powered from Bus 1F and is lost. Even though RHR Pump C is powered from Bus 1G, its flow path is through RHR Train A valves and is not available for mitigation. The remaining RHR Pump D via the Train B flowpath is not affected by the fire, is available, and can be used for RPV level control and SPC. Note that automatic control of valve RHR-MOV-MO16B is lost. When in SPC mode, this is acceptable. To protect the pump, operators may open MO16B if the valve spuriously closes. However, in LPCI or SDC modes, MO16B must be operated from MCC Y.	1.2	Dominant contributors are failure of operators to locally open primary containment vent valve PC-MOV-233MV as part of hard pipe vent alignment coupled with random failures of RHR Pump D.	1.90E-04	3.01E-03	5.72E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	Core Spray Pump A is powered from Bus 1F and is lost. Core Spray Pump B receives power from Bus 1G and is available for low pressure RPV level control. All four RHR Service Water (RHRSW) Booster Pumps are lost but operators may allow the Service Water system to "windmill" the RHRSW pump impellers maintaining Service Water flow to the RHR Heat Exchangers (see Implementation Item S-3.11 of Attachment S, Table S-3). A manual action to align MCC R to MCC S, which is backed by Bus 1G, was included in the FPRA model based on instructions provided in 5.4POST-FIRE for Fire Area RB-J. MCC R loads include several REC valves. Available frontline systems are: HPCI, RCIC, RHR Train B flowpath with Pump D and manual control of valve MO16B, Core Spray Train B, and Bus 1G.					
2C ASD ROOM	Scenario 2C ASD ROOM is a fire in the Alternate Shutdown Room above the Reactor Building SE stairway on 913' Elevation. This room contains the ASD panels for HPCI, RHR, ADS, and REC. Available frontline systems are: RCIC, ADS, Core Spray Train A, SDC Train A, SPC	1.1	This scenario is dominated by the failure of operators to depressurize the RPV and loss of HPCI.	1.12E-04	5.07E-03	5.67E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	Train A, REC Train A ,Service Water Train A, and both divisions of AC and DC power. An exclusion analysis was performed for Zone 2A-1 ASD ROOM scenario. This analysis determined that Condensate/Feedwater and their supporting systems are not failed by a fire in Zone 2A-1 ASD Room. The exclusion analysis included a review of the instrument air distribution system for this zone. By not assuming that Condensate/Feedwater is failed for a fire in Zone 2A-1 ASD Room, the CCDP decreased from 1.68E-02 to 5.07E-03 with a CDF of 5.67E-07.					
8A PNL 9-32-MCR	Zone 8A is the Auxiliary Relay Room on Elevation 903'-6" in the Control Building. Panel 9-32 is the Train A relay panel. Because many systems receive signals from 4 channels both Train A and Train B cables are associated with Panel 9-32. A fire in this panel will fail the control power to both the Train A and Train B ECCS logic circuits. The installation of incipient detection in Panel 9-32 is included in this scenario. Detailed circuit analysis has shown that the operators can operate a train of equipment from the Control Room if pre-NFPA 805 procedures are changed and if the operators are alerted to the specific location	1.1	CCDP/CLERP for safely shutting down from the Control Room is conservatively assumed to be on the order of 1.0E-02. Combined with the scenario frequency, this gives a CDF/LERF of 5.39E-07/year. Section V.2 provides additional details on incipient detection.	5.55E-05	9.70E-03	5.39E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	of the potential fire. This scenario is based on that case where operators respond correctly to the incipient detection alarm. Note that the incipient detection does not reduce the fire frequency or severity; it just provides an alert to fire location and allows the operators to take action from the Control Room. Failure of the operators to respond correctly to the incipient detection alarms results in Control Room abandonment and is addressed in scenario 8A PNL 9-32 with CDF and LERF values of 1.65E-07/year. Given a fire, the plant experiences a transient and it is assumed that Condensate/Feedwater are lost. An incipient detection alarm is received in the Control Room and operators verify fire location. Automatic actuation of ECCS is lost. However, the operators take manual control of the mitigating systems from the Control Room. The probability that operators successfully respond to fire in Panel 9-32 and shutdown from the Control Room is 9.7E-01.					
8A PNL 9-33-MCR	Panel 9-33 is the Train B relay panel. Like its counterpart Panel 9-32, both Train A and Train B cables are associated with Panel 9-33. A fire in this panel will fail the control power to both the Train A and Train B	1.1	See above discussion for 8A PNL 9-32-MCR.	5.55E-05	9.70E-03	5.39E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	ECCS logic circuits.					
9A PN-PL2-F1	This scenario is a fire of Panel LRP-PNL-PL2 "Foxboro Panel PL2 Div II" located in the center of Zone 9A, the Cable Spreading Room at the 918' Elevation of the Contol Building. The fire propagates beyond the panel and impacts FDS1 targets. These targets include RPS Power Panel RPSPP1A, HPCI, and RCIC. Only the RHR SDC suction valves are impacted for RHR. For fires in Fire Area CB-D, the fire response Procedures 5.4POST-FIRE and 5.4FIRE-S/D credit Control Room abandonment and use ASD to bring the plant to a safe stable condition. The calculated CCDP for Scenario 9A PN-PL2-F1 is 3.5E-02. For this FDS, operators would have sufficient command and control from the Control Room that they would not shutdown using ASD. Given a fire, a plant transient would be experienced and an assumed loss of Condensate/Feedwater. With no high pressure system available, the operators must depressurize the RPV and align Core Spray Train B or RHR for RPV level control. They must also align RHR in SPC to provide decay heat removal.	1.1	The scenario risk is dominated by the loss of RCIC and HPCI coupled with failure of the operators to depressurize the RPV and initiate low pressure injection.	1.51E-05	3.54E-02	5.33E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
9A CCP1A-FD1	Scenario 9A CCP1A FDS1 is a fire in Critical Control Panel EE-PNL-CCP1A in the Cable Spreading Room on Elevation 918' of the Control Building. This fire propagates beyond the panel impacting FDS1 targets. The FDS1 targets include Core Spray Train A flow and pressure instrumentation, the inboard MSIVs, RCIC power supply, and Service Water valve SW-MOV-MO89A. Critical Control Panel EE-PNL-CCP1B is impacted by the fire in this scenario. Service Water valve MO89B is initially lost because of CCP1B. For fires in Fire Area CB-D, the fire response Procedures 5.4POST-FIRE and 5.4FIRE-S/D credit Control Room abandonment and use ASD to bring the plant to a safe stable condition. The calculated CCDP for Scenario 9A CCP1A FDS1 is 3.5E-02. For this FDS, operators would have sufficient command and control from the Control Room that they would not shutdown using ASD. Given a fire, a plant transient would be experienced and an assumed loss of Condensate/Feedwater. HPCI is available for high pressure RPV level control. If needed, the operators can depressurize and use Core Spray Train B and both trains	1.1	Dominant contributors for this scenario are failure to depressurize using the SRVs and initiate low pressure RPV level control.	1.51E-05	3.47E-02	5.24E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	of RHR for low pressure RPV level control. With both Service Water MO89A and MO89B valves impacted by the fire, the operators would align the hard pipe vent for decay heat removal.					
9A PNL-DCA	Scenario 9A PNL-DCA is a fire in panel EE-PNL-DCA which causes loss of instrument power to CCP1A with the same equipment being impacted.	1.1	See above discussion for panel CCP1A.	1.51E-05	3.47E-02	5.24E-07
1E ZONE FAILURE	Scenario 1E ZONE FAILURE is a full zone burnup of the HPCI Pump Room at 859' Elevation of the Reactor Building. This fire results in the failure of HPCI. RHR SDC suction outboard isolation valve RHR-MOV-MO17 is also failed. All other systems are available; RCIC, Core Spray Train A and B, RHR Train A and B, Service Water Train A and B, all SRVs, and offsite power to Bus 1F and Bus 1G. Condensate and Feedwater were assumed not to fail for this scenario and are available. Given a fire, a plant transient would be experienced. RCIC is available for high pressure RPV level control. If needed, the operators can depressurize and use Core Spray Train B and both trains of RHR for low pressure RPV level control. Both trains of RHR are also available for SPC for decay	1.0	Dominant contributors for both scenarios are failure to trip the Reactor Recirculation pumps in the field and initiate ECCS given a small LOCA.	7.08E-04	7.13E-04	5.05E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	heat removal. An exclusion analysis was performed for Zone 1E ZONE FAILURE. This analysis determined that Condensate/Feedwater and their supporting systems are not failed by a fire in Zone 1E. The exclusion analysis included a review of the instrument air distribution system for this zone. By not assuming that Condensate/Feedwater is failed for a fire in Zone 1E, the CCDP decreased from 1.24E-03 to 7.23E-04 with a CDF of 5.05E-07.					
9A PN-PL1-F1	This scenario is a fire of Panel LRP-PNL-PL1 "Foxboro Panel PL1 Div I" located in the center of Zone 9A, the Cable Spreading Room at the 918' Elevation of the Control Building. The fire propagates beyond the panel and impacts FDS1 targets. These targets include Critical Control Panel CCP1B, RPS Power Panel RPSPP1A, HPCI, and RCIC. Only the RHR SDC suction valves are impacted for RHR. For fires in Fire Area CB-D, the fire response Procedures 5.4POST-FIRE and 5.4FIRE-S/D credit Control Room abandonment and use ASD to bring the plant to a safe stable condition. The calculated CCDP for Scenario 9A PN-PL1-	1.0	The scenario risk is dominated by the loss of RCIC and HPCI coupled with failure of the operators to depressurize the RPV and initiate low pressure injection.	1.41E-05	3.57E-02	5.04E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	Description	CDF Contribution (%)	Risk Insights	IF	CCDP	CDF (per year)
	F1 is 3.6E-02. For this FDS, operators would have sufficient command and control from the Control Room that they would not shutdown using ASD. Given a fire, a plant transient would be experienced and an assumed loss of Condensate/Feedwater. With no high pressure system available, the operators must depressurize the RPV and align Core Spray or RHR for RPV level control. They must also align RHR Train A in SPC to provide decay heat removal. The loss of CCP1B would result in Service Water SW-MOV-MO89B not being available.					
8H SWG125A-FD0	This is a fire of 125VDC Switchgear 1A with equipment that terminates at the switchgear being impacted. 125VDC Switchgear 1A is located in the DC Switchgear Room 1A, 903' Elevation of the Control Building. 125VDC Battery 1A, 125VDC Battery Charger 1A, 125VDC Switchgear 1A, 125VDC RCIC Starter Rack, and 125VDC Distribution Panel are all initially lost. Given a fire, a plant transient would be experienced and an assumed loss of Condensate/Feedwater. RCIC is lost; however, HPCI is available for RPV level control. All Division 2 systems are also	1.0	In addition to the loss of 125VDC, the risk is dominated by test/maintenance of RHR Pump D.	4.60E-05	1.09E-02	5.00E-07

Table W-1 Individual Scenarios 1% and Above of Total Fire CDF						
Scenario	**Description**	**CDF Contribution (%)**	**Risk Insights**	**IF**	**CCDP**	**CDF (per year)**
	available.					

Table W-2 CNS Fire Area Risk Summary

Fire Area	Fire Area Description	NFPA 805 Basis	Fire Area CDF	Fire Area LERF	VFDR(s) (Yes/ No)	RA(s) (Yes/ No)	Fire Risk Eval Delta CDF	Fire Risk Eval Delta LERF	Additional Risk of RAs
CB-A	RHR SW Booster Pump and Service Air Compressor Areas Emerg Condensate Storage Tank Area RPS Room 1A Seal Water Pump Area and Corridor	4.2.4.2	6.89E-07	7.03E-08	Yes	Yes	1.48E-07	4.62E-08	1.48E-07 4.62E-08
CB-A-1	Battery Room 1A DC Swgr Room 1A	4.2.4.2	3.43E-06	1.14E-07	Yes	Yes	9.49E-08	4.68E-08	9.49E-08 4.68E-08
CB-B	Battery Room 1B DC Swgr Room 1B	4.2.4.2	4.61E-06	1.85E-07	Yes	Yes	1.54E-07	6.94E-08	1.54E-07 6.94E-08
CB-C	RPS Room 1B	4.2.4.2	1.74E-07	7.83E-09	Yes	Yes	ε	ε	NA
CB-D	Computer Room Control Room and SAS Corridor Aux Relay Room Cable Spreading Room Cable Expansion Room	4.2.4.2	1.37E-05	4.20E-06	Yes	Yes	-1.53E-05	-1.44E-05	-1.53E-05 -1.44E-05
DG-A	Div. 1 Diesel Generator	4.2.3.2	5.09E-06	9.18E-08	No	No	NA	NA	NA
DG-B	Div. 2 Diesel Generator	4.2.3.2	1.16E-06	9.00E-08	No	No	NA	NA	NA
IS-A	SW Pump Area Circ Water Pump and Traveling Screen Area	4.2.4.2	2.14E-07	2.61E-08	Yes	Yes	2.36E-08	4.43E-10	2.36E-08 4.43E-10
RB-A	RCIC and CS A Pump Room	4.2.4.2	2.52E-07	1.04E-08	Yes	Yes	7.90E-08	2.29E-09	7.90E-08 2.29E-09
RB-B	Core Spray B Pump Room Hydraulic Drive Pump Area	4.2.4.2	1.25E-07	1.15E-08	Yes	Yes	1.25E-07	1.15E-08	1.25E-07 1.15E-08
RB-CF	RHR Pump Rm 1A and 1C CRD Units-North 903' 6" South Corridor RHR HX-1A	4.2.4.2	1.67E-06	7.15E-08	Yes	Yes	7.90E-07	5.12E-08	7.90E-07 5.12E-08

Table W-2 CNS Fire Area Risk Summary

Fire Area	Fire Area Description	NFPA 805 Basis	Fire Area CDF	Fire Area LERF	VFDR(s) (Yes/ No)	RA(s) (Yes/ No)	Fire Risk Eval Delta CDF	Fire Risk Eval Delta LERF	Additional Risk of RAs
RB-DI	RHR Pump Room 1B and 1D HPCI Pump Room 903' 6" South Corridor CRD Units South RHR HX-1B	4.2.4.2	2.86E-06	5.05E-07	Yes	Yes	9.50E-07	2.00E-08	9.50E-07 2.00E-08
RB-E	Suppression Pool Area	4.2.4.2	2.15E-07	5.14E-09	Yes	Yes	2.18E-09	2.44E-09	2.18E-09 2.44E-09
RB-FN	Rx Bldg 903' 6" NE Corner	4.2.4.2	1.84E-06	1.83E-08	Yes	Yes	-1.24E-07	-7.03E-10	-1.24E-07 -7.03E10
RB-J	SWGR Room 1F	4.2.4.2	1.25E-06	2.58E-07	Yes	Yes	1.43E-07	5.65E-08	1.43E-07 5.65E-08
RB-K	SWGR Room 1G	4.2.4.2	1.64E-06	2.64E-07	Yes	Yes	-2.08E-06	-2.15E-06	-2.08E-06 -2.15E-06
RB-M	RWCU Recirc Pumps and Corridor	4.2.4.2	4.46E-07	2.21E-08	Yes	Yes	-6.32E-07	-2.76E-09	-6.32E-07 -2.76E-09
RB-N	RHR HX-1B Regenerative HX Areas RWCU Recirc Pumps and Corridor	4.2.4.2	2.63E-07	2.48E-08	Yes	Yes	6.43E-08	5.70E-10	6.43E-08 5.70E-10
RB-P	RB Elevator and accessway Area RB HVAC Areas Fuel Pool, HX, CRD Repair Room, and Raw Water Cleanup Areas Reactor MG Set Oil Pump Area	4.2.4.2	9.02E-08	1.97E-08	Yes	Yes	4.63E-08	1.72E-08	4.63E-08 1.72E-08
RB-T	SBLC Pump Tank and Accessway (Zone 5A) and Refueling Floor (Zone 6)	4.2.3.2	3.80E-08	6.53E-09	No	No	NA	NA	NA
RB-V	Rx MG Set Area	4.2.4.2	3.15E-08	7.19E-09	Yes	Yes	1.88E-08	6.31E-09	1.88E-08 6.31E-09
TB-A	Turbine Building	4.2.4.2	9.56E-06	3.87E-06	Yes	Yes	6.77E-06	3.33E-06	6.77E-06 3.33E-06
TB-C	Steam Tunnel	4.2.4.2	1.09E-08	1.23E-09	Yes	Yes	ε	ε	NA

Table W-2 CNS Fire Area Risk Summary

Fire Area	Fire Area Description	NFPA 805 Basis	Fire Area CDF	Fire Area LERF	VFDR(s) (Yes/ No)	RA(s) (Yes/ No)	Fire Risk Eval Delta CDF	Fire Risk Eval Delta LERF	Additional Risk of RAs
Yard	Yard Outside of Buildings	4.2.3.2	1.35E-06	6.30E-07	No	No	NA	NA	NA
	Total		**5.07E-05**	**1.05E-05**			**-8.71E-06**	**-1.29E-05**	**-8.71E-06** **-1.29E-05**

ATTACHMENT 1

License Condition 2.C.(4) and Technical Specification
Page Markups

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2419 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 241, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Cooper Nuclear Station Safeguards Plan," submitted by letter dated May 17, 2006.

NPPD shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NPPD CSP was approved by License Amendment No. 238.

(4) Fire Protection

Insert 1

~~The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Cooper Nuclear Station (CNS) Updated Safety Analysis Report and as approved in the Safety Evaluations dated November 29, 1977; May 23, 1979; November 21, 1980; April 29, 1983; April 16, 1984; June 1, 1984; January 3, 1985; August 21, 1985; April 10, 1986; September 9, 1986; November 7, 1988; February 3, 1989; August 15, 1995; and July 31, 1998, subject to the following provision:~~

~~The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.~~

Insert 1

NPPD shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated __________ (and supplements dated _________) and as approved in the safety evaluation report dated ___________ (and supplements dated ____________). Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) <u>Risk-Informed Changes that May Be Made Without Prior NRC Approval</u>

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 CNS. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1 10^{-7}/year (yr) for CDF and less than 1 10^{-8}/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) <u>Other Changes that May Be Made Without Prior NRC Approval</u>

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

   Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A

qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the 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 report dated ________ to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by (c)2. below, risk-informed changes to NPPD'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 (b)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) as provided in Table S-2 of the Cooper Nuclear Station License Amendment Request dated April 27, 2012, prior to startup from the first refueling outage greater than 12 months following the issuance of the License Amendment.

3. The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance program for radioactive effluent and radiological environmental monitoring; and

d. ~~Fire Protection Program implementation; and~~ Not Used

e. All programs specified in Specification 5.5.

ATTACHMENT 2

Clean, Retyped License and Technical Specification Pages

<u>NEBRASKA PUBLIC POWER DISTRICT</u>

<u>DOCKET NO. 50-298</u>

<u>(COOPER NUCLEAR STATION)</u>

<u>FACILITY OPERATING LICENSE</u>

Renewed License No. DPR-46

1. The Nuclear Regulatory Commission (the Commission), having previously made the findings set forth in License No. DPR-46, has now found that:

   A. The application for renewed Facility Operating License No. DPR-46 filed by the Nebraska Public Power District (NPPD, the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Cooper Nuclear Station (facility) has been substantially completed in conformity with Provisional Construction Permit No. CPPR-42 and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. The licensee is technically and financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public;

   H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of renewed Facility Operating License No. DPR-46 (subject to the conditions for protection of the environment set forth herein) is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied;

   I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by the license will be in accordance with the Commission's regulations in

10 CFR Parts 30, 40, and 70, including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31; and

J. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.

2. Facility Operating License No. DPR-46 is superseded by Renewed Facility Operating License No. DPR-46, hereby issued to the Nebraska Public Power District, to read as follows:

   A. This renewed operating license applies to the Cooper Nuclear Station, a boiling water nuclear reactor and associated equipment (the facility), owned by the Nebraska Public Power District. The facility is located near Brownville in Nemaha County, Nebraska, and Atchison County, Missouri, and is described in the "Final Safety Analysis Report" (Amendment 7) as supplemented and amended (Amendments 8 through 30), and the Environmental Report as supplemented and amended (Supplements 1 through 6).

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Nebraska Public Power District:

      (1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location near Brownville in Nemaha County, Nebraska, and Atchison County, Missouri, in accordance with the procedures and limitations set forth in this renewed license;

      (2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended, and the licensee's filings dated June 20, 1975 and September 22, 1975;

      (3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time any byproduct, source and special nuclear materials as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

      (4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear materials without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2419 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Cooper Nuclear Station Safeguards Plan," submitted by letter dated May 17, 2006.

NPPD shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NPPD CSP was approved by License Amendment No. 238.

(4) Fire Protection

NPPD shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated __________ (and supplements dated _________) and as approved in the safety evaluation report dated ___________ (and supplements 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.

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2419 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Cooper Nuclear Station Safeguards Plan," submitted by letter dated May 17, 2006.

NPPD shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NPPD CSP was approved by License Amendment No. 238.

(4) Fire Protection

NPPD shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated ___________ (and supplements dated _________) and as approved in the safety evaluation report dated ____________ (and supplements dated _____________). Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at CNS. Acceptable methods to assess the risk of the change may include methods that have been used in the peer- reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense- in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-8}/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) Other Changes that May Be Made Without Prior NRC Approval

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

   Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the 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 report dated ________ to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) <u>Transition License Conditions</u>

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by (c)2. below, risk-informed changes to NPPD'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 (b)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) as provided in Table S-2 of the Cooper Nuclear Station License Amendment Request dated April 27, 2012, prior to startup from the first refueling outage greater than 12 months following the issuance of the License Amendment.

3. The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

(5) <u>Additional Conditions</u>

The Additional Conditions contained in Appendix C, as revised through Amendment No. 178, are hereby incorporated into this license. Nebraska Public Power District shall operate the facility in accordance with the Additional Conditions.

(6) Deleted

(7) <u>Mitigation Strategy License Condition</u>

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
   1. Pre-defined coordinated fire response strategy and guidance
   2. Assessment of mutual aid fire fighting assets
   3. Designated staging areas for equipment and materials
   4. Command and control
   5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(8) The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

(9) Upon implementation of Amendment No. 230 adopting TSTF-448-A, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.4.4, in accordance with Specification 5.5.13.c.(i), the assessment of CRE habitability as required by Specification 5.5.13.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.13.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.4.4, in accordance with Specification 5.5.13.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from July 12, 2004, the date of the most recent successful tracer gas test. (The tracer gas test was stated to have been performed in July, 2004, in the September 30, 2004 letter response to Generic Letter 2003-01).

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.13.c.(ii), shall be within the next 9 months.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.13.d, shall be within 18 months, plus the 138 days allowed by SR 3.0.2, as measured from May 4, 2007, the date of the most recent successful pressure measurement test.

D. (Not Used)

E. The Updated Safety Analysis Report (USAR) supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the USAR required by 10 CFR 50.71(e)(4), as appropriate, following the issuance of this renewed operating license. Until this update is complete, the licensee may not make changes to the information in the supplement. Following incorporation of the supplement into the USAR, the need for Commission approval of any changes will be governed by 10 CFR 50.59.

F. The USAR supplement, as revised, describes certain future activities to be completed prior to and/or during the period of extended operation. The licensee shall complete these activities in accordance with Appendix A of NUREG-1944, "Safety Evaluation Report Related to the License Renewal of Cooper Nuclear Station," dated October 2010, as supplemented by letters from the licensee to the U.S. Nuclear Regulatory Commission (NRC) dated November 15 and 18, 2010. The licensee shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

G. This license is effective as of the date of issuance and shall expire at midnight, January 18, 2034.

FOR THE NUCLEAR REGULATORY COMMISSION

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendices A&B - Technical Specifications
Appendix C - Additional Conditions

Date of Issuance: November 29, 2010

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance program for radioactive effluent and radiological environmental monitoring; and

d. Not Used

e. All programs specified in Specification 5.5.