Xcel Energy®

Prairie Island Nuclear Generating Plant
1717 Wakonade Drive East
Welch, MN 55089

May 28, 2015

L-PI-15-041
10 CFR 50.90
10 CFR 50.48(c)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
Renewed License Nos. DPR-42 and DPR-60

License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors – Response to Request for Additional Information (TAC Nos. ME9734 and ME9735)

References:

1. NSPM letter, J.P. Sorensen to NRC Document Control Desk, *License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors,* L-PI-12-089, dated September 28, 2012 (ADAMS Accession No. ML12278A405).

2. NSPM letter, S. Sharp to NRC Document Control Desk, *Supplement to License Amendment Request to Adopt NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactors,* L-PI-14-045, dated April 30, 2014 (ADAMS Nos. ML14125A106 and ML14125A149).

3. NRC email, T. Beltz to S. Chesnutt, *Prairie Island Nuclear Generating Plant, Units 1 and 2 - NFPA 805 Requests for Additional Information and Response Timeline (TAC Nos. ME9734 and ME9735),* dated March 30, 2015 (ADAMS Accession No. ML15089A157).

In Reference 1, the Northern States Power Company, a Minnesota Corporation (NSPM) doing business as Xcel Energy requested approval from the Nuclear Regulatory Commission (NRC) to transition the fire protection licensing basis for the Prairie Island Nuclear Generating Plant (PINGP) to 10 CFR 50.48(c), National Fire Protection Association Standard 805 (NFPA 805). Supplemental information was provided in letters dated November 8, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12314A144) and December 18, 2012 (ADAMS Accession No. ML12354A464).

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In Reference 2, NSPM submitted a revised Fire Probabilistic Risk Assessment (PRA) in a supplement to the subject License Amendment Request (LAR). In Reference 3, the NRC staff provided requests for additional information (RAIs) regarding this request and also provided a timeline and due dates for submitting responses within 60, 90, or 120 days after an on-site Audit that was conducted March 23-25, 2015.

Enclosure 1 provides NSPM's responses to all of the 60-day RAIs (due by May 29, 2015) and one 90-day RAI included in Reference 3.

Enclosure 2 provides Licensee Identified Changes to the LAR (Reference 2) that are not directly associated with RAI responses. This enclosure includes clarifications and changes to LAR Attachment S, Modifications and Implementation Items, that are based on NSPM reviews subsequent to the submittal of Reference 2. These changes will be included in a revision to Attachment S to be submitted with the final RAI response letter.

Enclosure 3 provides a revision to Approval Request 1 in LAR Attachment L, "NFPA 805 Chapter 3 Requirements for Approval," in support of the response to FPE RAI 05. This revised Approval Request 1 supersedes the Approval Request 1 submitted in Reference 2.

Enclosure 4 provides a revision to Request 1 in LAR Attachment T, "Clarifications of Prior NRC Approvals," in support of the response to SSA RAI 02. This revised Request 1 supersedes the Request 1 submitted in Reference 2.

This letter is submitted in accordance with 10 CFR 50.90. The additional information provided in this letter does not impact the conclusions of the No Significant Hazards Evaluation or Environmental Considerations Evaluation presented in Reference 2.

In accordance with 10 CFR 50.91, NSPM is notifying the State of Minnesota of this additional information by transmitting a copy of this letter to the designated State Official.

If there are any questions or if additional information is needed, please contact Gene Eckholt at 651-267-1742.

Summary of Commitments

This letter contains no new commitments and makes no revisions to any existing commitments.

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

Executed on **MAY 28 2015**

Kevin Davison

Kevin Davison
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosures (4)

cc: Administrator, Region III, USNRC
NRR Project Manager, PINGP, USNRC
Resident Inspector, PINGP, USNRC
State of Minnesota

Response to Requests for Additional Information (RAIs)
Regarding the License Amendment Request to
Adopt National Fire Protection Association (NFPA) Standard 805
at Prairie Island Nuclear Generating Plant Units 1 and 2

Responses to 60-Day RAIs

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Introduction

This Enclosure provides additional information from the Northern States Power Company, a Minnesota corporation (hereafter "NSPM") doing business as Xcel Energy, in support of a License Amendment Request (LAR) to transition the Fire Protection Licensing Basis for the Prairie Island Nuclear Generating Plant (PINGP) to National Fire Protection Association Standard 805 (NFPA 805).

NSPM submitted an NFPA 805 LAR for PINGP in a letter dated September 28, 2012 (ADAMS Accession No. ML12278A405). Supplemental information was submitted in letters dated November 8, and December 18, 2012 (ADAMS Accession Nos. ML12314A144 and ML12354A464, respectfully). In a letter dated April 30, 2014, NSPM submitted a revised Fire Probabilistic Risk Assessment (PRA) in a supplement to the subject LAR (ADAMS Accession Nos. ML14125A106 and ML14125A149). This 2014 Supplement included the entire LAR and was designated "Revision 1;" this is the version referred to in discussions involving "the LAR."

After an onsite audit conducted during the week of March 23, 2015, the NRC staff provided requests for additional information (RAIs) and a response timeline in an email dated March 30, 2015 (ADAMS Accession No. ML15089A157). The timeline provided due dates for 60, 90, and 120 day RAI responses.

This Enclosure provides NSPM's responses to all of the 60-day RAIs which are due by May 29, 2015, and one 90-day RAI, as follows:

- Fire Protection Engineering (FPE) RAI 01, 02, 03*, 04, 05, 06
  (*note that 03 was originally a 90-day RAI)
- Safe Shutdown Analysis (SSA) RAI 01, 02, 03, 04, 05, 06, 08, 09, 10
- Fire Modeling (FM) RAI 04, 05, 06
- Radioactive Release (RR) RAI 01, 02, 03
- Probabilistic Risk Assessment (PRA) RAI 01.a, 01.b, 01.c, 01.d, 01.f, 02.b, 02.c, 04, 05, 06, 09, 10, 11, 13, 14, 15, 19

Each of the RAI questions is quoted in italics and each question is then followed by the NSPM response in normal font. Referenced documents are identified at the end of each RAI response.

Acronyms and Abbreviations

ADAMS	Agencywide Document Access and Management System
ADT	Auxiliary Drain Tank
AFW	Auxiliary Feedwater
AFWP	Auxiliary Feedwater Pump
AHJ	Agency Having Jurisdiction
ANS	American Nuclear Society
AOP	Abnormal Operating Procedure
APEO	Auxiliary Plant Equipment Operator
AR	Action Request
ARP	Alarm Response Procedure
ASD	Alternate Shutdown
ASME	American Society of Mechanical Engineers
BA	Boric Acid
BAST	Boric Acid Storage Tank
BTP	Branch Technical Position
CAAB	Common Area of the Auxiliary Building
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CE	Combustion Engineering
CFAST	Consolidated Fire and Smoke Transport
CFR	Code of Federal Regulations
CLERP	Conditional Large Early Release Probability
CLG WTR	Cooling Water
CO2	Carbon Dioxide
CR	Control Room
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
DDFP	Diesel-Driven Fire Pump
DID	Defense in Depth
DAW	Dry Active Waste
DC	Direct Current
EC	Engineering Change
EOP	Emergency Operating Procedure
EPRI	Electrical Power Research Institute
EPA	Environmental Protection Agency
ERFBS	Electrical Raceway Fire Barrier System
ES	Equipment Selection
FA	Fire Area
FAQ	Frequently Asked Question
FDS	Fire Dynamics Simulator
FDT	Fire Dynamics Tool
FM	Fire Modeling
F&O	Fact & Observation
FPA	Foote, Pagni, and Alvares
FPE	Fire Protection Engineering (or Engineer)
FPEE	Fire Protection Engineering Evaluation
FPRA	Fire Probabilistic Risk Assessment

FQ	Fire Quantification
FRANX	Fire Risk Analysis Software Tool
FRE	Fire Risk Evaluation
HEAF	High Energy Arcing Fault
HEP	Human Error Probability
HEPA	High Efficiency Particulate Air
HFE	Human Failure Event
HRA	Human Reliability Analysis
HRE	Higher Risk Evolution
HRR	Heat Release Rate
HSD	Hot Shutdown
HUT	Holdup Tank
IFG	Instrument Failure Guide
IGN	Ignition
KSF	Key Safety Function
LAR	License Amendment Request
LERF	Large Early Release Frequency
LLRW	Low Level Radioactive Waste
MAAP	Modular Accident Analysis Program
MCR	Main Control Room
MDFP	Motor-Driven Fire Pump
MQH	McCaffrey, Quintiere, and Harkleroad
N/A	Not Applicable
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NLO	Non-Licensed Operator
NPO	Non-Power Operations
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSCA	Nuclear Safety Capability Assessment
NSP	Northern States Power
NSP	Non-Suppression Probability
NSPM	Northern States Power - Minnesota
NSSS	Nuclear Steam Supply System
NUREG	NRC Publication
NUREG/CR	NUREG/Contractor Report
OCA	Owner Controlled Area
ODCM	Offsite Dose Calculation Manual
PA	Plant Attendant
PAU	Physical Analysis Unit
PCS	Primary Control Station
P&ID	Piping and Instrumentation Diagram
PI	Prairie Island
PINGP	Prairie Island Nuclear Generating Plant
PNS	Probability of Non-Suppression
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RA	Recovery Action
RAI	Request for Additional Information
RCA	Radiologically Controlled Area

RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RMU	Reactor Makeup
RO	Reactor Operator
RR	Radioactive Release
RWST	Refueling Water Storage Tank
SAFE GENESIS	Database used to develop PINGP analytical model
SCA	Single Compartment Analysis
SCBA	Self Contained Breathing Apparatus
SER	Safety Evaluation Report
SF	Severity Factor
SFP	Spent Fuel Pool
SFPE	Society of Fire Protection Engineers
SR	Supporting Requirement
SSA	Safe Shutdown Analysis
SSC	Structures, Systems, and Components
TBD	Technical Basis Document
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
V&V	Verification and Validation
VCT	Volume Control Tank
VDC	Volts Direct Current
VEWFDS	Very Early Warning Fire Detection System
VFDR	Variance From Deterministic Requirements
WCAP	Westinghouse Commercial Atomic Power
X/Q	Chi/Q – Relative Concentration
ZOI	Zone of Influence

RAI Responses – Fire Protection Engineering (FPE)

FPE RAI 01

NFPA 805, Section 3.4.1(a), requires that a fully-staffed and qualified fire brigade comply with NFPA standards NFPA 600, NFPA 1500, and NFPA 1582, as applicable. In the Enclosure to your August 20, 2014, license amendment request (hereafter referred to as the LAR), Attachment A, Section 3.4.1(a), it states that the compliance strategy as "Complies via Previous Approval." However, the compliance basis does not provide excerpts from the past licensee submittal(s) and associated U.S. Nuclear Regulatory Commission (NRC) documentation that substantiates the previous approval. The NRC endorsed guidance in NEI 04-02, Appendix B, Section B-1, states that "When claiming previous approval, excerpts from the NRC documents that provided the formal approval shall be included in documentation, as well as appropriate excerpts from licensee's submittals." Additionally, Section B.1 states "for each reference document that is referenced as part of the transition review, provide sufficient documentation to provide traceability back to the original submittal and approval.

Please provide, as appropriate, information such as revision number, date, and section/page number in order to make the statements as clear as possible to facilitate reviews and long term configuration management."

Additionally,

[This RAI includes Subparts a and b, as shown below along with NSPM responses]

NRC Request (FPE RAI 01.a):

a. *The Technical Specification discussion in the Compliance Basis only addresses minimum staffing and does not address the fire brigade capabilities as addressed in the NFPA standards.*

   Please provide additional details to substantiate the compliance strategy that the NRC has previously approved the fire brigade for PINGP.

NSPM Response (FPE RAI 01.a):

a. Correspondence between Northern States Power (NSP) and the NRC that explains the NRC's previous approval of the PINGP fire brigade includes the following:

   <u>NSP letter dated January 9, 1979 (Reference F-1):</u> NSP submitted information regarding fire brigade training that included the following excerpts:

   PF-44 Fire Brigade Practice Sessions

   Staff Position: Practice sessions should be held for fire brigade members on the proper method of fighting various types of fires that could occur in a nuclear power plant considering such factors as the magnitude of the fire, and the complexity and difficulty of fire fighting. These sessions should be designed to provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous working conditions. The sessions should be in addition to the scheduled fire brigade training sessions and fire drills and should

include firefighting strategies, (i.e., simple plans showing fire fighting equipment locations, entry and egress points, ventilation, communications and emergency lighting locations and controls). These practice sessions should be provided at regular intervals, but not exceeding a one year interval for each fire brigade member.

Licensee's Response

The licensee will respond to this position by 3-2-79. [The response is provided in an excerpt from the March 9, 1979 letter below (Reference F-2).]

PF-6 Fire Brigade Training

d. Fire Drills should be performed at regular intervals but not to exceed three months for each brigade.

Licensee's Response

d. With our six (6) fire brigades, this would require a minimum of 24 drills a year. We consider this excessive and are concerned it could lead to perfunctory performance on the part of those participating in the drills as well as an unwarranted state of confusion for others at the facility. We believe one drill per month (2 drills per brigade) is the maximum.

NSP letter dated March 9, 1979 (Reference F-2): NSP submitted additional information regarding fire brigade training that included the following excerpt:

PF-44 Fire Brigade Practice Sessions

[See Staff Position in the January 9, 1979 letter above (Reference F-1)]

Licensees Response

We agree to schedule fire brigade practice sessions as described above at least annually. We cannot provide assurance, however, that 100% of the fire brigade members will be available to attend each scheduled practice session. Due to vacation, sickness, offsite training, and unexpected schedule changes, some fire brigade members may miss a session. Because of the amount of preparation and planning that goes into a practice session, it is impractical to schedule makeup sessions. We therefore propose to require, at least 85% of all fire brigade members to attend each practice session.

NSP letter dated May 2, 1979 (Reference F-3): NSP submitted additional information regarding fire brigade staffing that included the following excerpt:

J. Nuclear Plant Fire Brigades and Fire Brigade Support Teams

A Fire Brigade of five (5) persons will be on-site at all times. In addition, a Fire Brigade Support Team will be on-site at all times to bring the minimum number of persons responding to any fire to five. The Fire Brigade Support Team may be drawn from the site security force. The Support Team assists the Fire Brigade by providing communications, bringing equipment to the scene, renewing air breathing bottles, and providing other support.

Each Fire Brigade will have an appointed leader. This leader will not be the Shift Supervisor (the Unit No. 1 Shift Supervisor at Prairie Island).

NRC SER dated September 6, 1979 (Reference F-4): The NRC's previous approval of the fire brigade for PINGP is provided in a Safety Evaluation Report (SER) dated September 6, 1979. This SER was included in a letter from A. Schwencer (NRC) to L.O. Mayer (NSP) identified as Reference 6.28 in the LAR (Reference F-4).

Sections of the SER that apply to the Fire Brigade include the following excerpts:

Section 6.1, Fire Protection Organization, page 6-1:

The licensee's fire protection organization contains the organizational responsibilities and lines of communication between the various positions through the use of organizational charts and functional descriptions of each position's responsibilities. Upper level offsite management positions are designated which have management responsibility for the formulation, implementation, and assessment of the effectiveness of the nuclear plant fire protection program. The results of these assessments are reported to the upper level management position responsible for fire protection with recommendations for improvements or corrective actions as deemed necessary.

The organizational responsibilities are delineated through onsite management positions for design, installation, testing, maintenance, modification, and review of fire protection systems and for fire brigade training.

A fire brigade size has been agreed upon. Qualification requirements have been established for fire brigade members, and the position responsible for formulating and implementing the fire protection program. Satisfactory completion of a physical examination including periodic screening for performing strenuous activity is required for all fire brigade members.

We find that the fire protection organization conforms to the provisions of Appendix A to BTP 9.5-1 and is, therefore, acceptable.

Section 6.2, Fire Brigade Training, page 6-2:

The fire brigade training program consists of an initial classroom instruction program followed by periodic classroom instruction, practice in firefighting and fire drills. Practice sessions are held for fire brigade members on the proper method of fighting various types of fires that could occur in a nuclear plant considering such factors as the magnitude of the fire and the complexity and difficulty of firefighting. Records of training are provided and are available for review including drill critiques.

We find that the fire brigade training conforms to the provisions of Appendix A to BTP 9.5-1 and is, therefore, acceptable.

Section 6.5, Firefighting Procedures, page 6-2

The licensee has provided an adequate description of its current firefighting procedures, those under development and those planned to be developed in the

near future. Firefighting procedures/plans are established to cover such items as notification of a fire, fire emergency procedures, coordination of firefighting activities with offsite fire departments, strategies for fighting fires in all safety-related areas and areas presenting a hazard to safety-related equipment. Provisions have been made for including offsite firefighting organizations in fire brigade drills and training as required.

We find that the control of firefighting procedures conforms to the provisions of Appendix A to BTP 9.5-1 and is, therefore, acceptable.

NRC letter dated September 6,1979 (Reference F-4): In addition, the 1979 NRC letter (Reference F-4) included the following statements regarding resolutions of fire brigade issues.

4. Technical Specification 6.1.C.6 requires a fire brigade of three members rather than the five members that the staff had requested. We have completed our review of this matter and have concluded that a five man brigade is necessary. Your staff has agreed to this change and the Technical Specifications have been changed in these amendments to read "five" rather than "three" fire brigade members.

5. Figure 6.1-1 does not have the specific delineation of fire protection responsibility and Figure 6.1-2 does not have specific delineation of operator and fire protection training responsibilities under the Training Supervisor as requested by the staff. We have completed our review and find that Figure 6.1-1 and Figure 6.1-2 should be modified as requested. Your staff has agreed to these changes and they have been incorporated in these amendments.

The SER concluded that the PINGP Fire Brigade capabilities conform to the provisions of Appendix A to BTP 9.5.1 and are therefore acceptable.

NRC Request (FPE RAI 01.b):

b. The licensee has stated that the fire brigade has been reviewed against NFPA 600; however there is no statement of compliance with regard to the requirements of NFPA 600.

Please provide a more detailed description of compliance with NFPA 600.

NSPM Response (FPE RAI 01.b):

b. NSPM performed a detailed review of NFPA 600, Standard on Industrial Fire Brigades, as documented in Fire Protection Engineering Evaluation FPEE-11-031, NFPA 600 – 2000 Code Compliance Report, Revision 1, and did not identify any deviations from the Code.

Based on this review, the PINGP Fire Brigade complies with NFPA 600, 2000 edition, and therefore, Section 3.4.1(a) of NFPA 805.

References

F-1 NSP letter, L.O. Mayer to Director of Nuclear Reactor Regulation (NRC), Corrected Pages for NSP Initial Response to NRC Staff Evaluation of Fire Protection Program, dated January 9, 1979.

F-2 NSP letter, L.O. Mayer to Director of Nuclear Reactor Regulation (NRC), NRC Staff Evaluation of Fire Protection Program, dated March 9, 1979.

F-3 NSP letter, L.O. Mayer to Director of Nuclear Reactor Regulation (NRC), Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance, dated May 2, 1979.

F-4 Letter from A. Schwencer (NRC) to L.O. Mayer (NSP), Issuance of License Amendment Nos. 39 (DPR-42, Unit 1) and 33 (DPR-60, Unit 2), and Related Fire Protection Safety Evaluation Report, dated September 6, 1979.

FPE RAI 02

NFPA 805, Section 3.4.1(c), requires that the fire brigade leader and at least two brigade members have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance criteria.

As described in the Compliance Basis for Section 3.4.1(c) in Attachment A of the LAR, NSPM has incorporated the NFPA 805 requirement in its procedures; however, it does not summarize the training and knowledge level required of the non-licensed operators that are assigned to the fire brigade.

Please provide additional detail regarding the training that is provided to the fire brigade members that addresses their ability to assess the effects of fire and fire suppressants on nuclear safety performance criteria.

NSPM Response (FPE RAI 02):

NFPA 805, Chapter 3, Section 3.4.1(c) requires certain fire brigade members to have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance criteria.

The guidance in FAQ 13-0069, Revision 3, states that one acceptable approach to meeting the requirements of NFPA 805 Section 3.4.1(c) is to provide a Fire Brigade leader that is a licensed reactor operator or has equivalent knowledge of plant systems and at least two brigade members that have training at the level of Non-Licensed Operator (NLO) or have equivalent knowledge of plant systems.

All fire brigade members are trained NLOs. The NLOs are split into two classifications based on their level of experience. The Plant Attendant (PA) reports to the plant and performs the fire brigade and operator/ watchstander duty after receiving initial training. The Auxiliary Plant Equipment Operator (APEO) is more experienced than the PA because, in addition to the initial

training, they have also completed their apprenticeship and journeyperson and fire brigade leader training. The fire brigade may be made up of both PAs and APEOs but only APEOs perform the duty of the Fire Brigade Leader. All fire brigade members receive training on fire control.

All NLOs receive training on all systems in their initial qualifications prior to reporting to the plant as a PA. NLOs rotate between watchstanding positions allowing them to gain experience in all of the systems.

NLOs are evaluated on each system lesson plan objective during initial and continuing training per Systematic Approach to Training processes. A review of the standard learning objectives in system training materials was conducted. The following learning objectives apply to NLOs and Reactor Operators (RO)s:

- Describe the System Flow Path and Major Components for:
    - Normal Operations
    - Abnormal Operations
    - Emergency Operations
- System integration with respect to other associated systems.
- Precautions and limitations pertinent to the system, technical specifications, and Updated Safety Analysis Report design criteria.

All NLOs and ROs receive training on all plant systems including system integration with respect to other associated systems, precautions and limitations pertinent to the systems and equipment required for abnormal and emergency operations, providing the members of the fire brigade with sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance criteria. In addition to the training regarding nuclear safety systems, the Fire Brigade Leader has additional experience with all of the systems during their apprenticeship at the various watchstanding positions. The PINGP training program for NLOs is consistent with the guidance in FAQ 13-0069, Revision 3, for equivalent knowledge of plant systems."

Based on the above, Fire Brigade training complies with NFPA 805 section 3.4.1(c).

FPE RAI 03

NFPA 805, Section 3.10.1, requires that 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 applicable NFPA codes. Attachment A, Section 3.10.1, of the LAR identifies an implementation item for code deviations that require a modification to resolve non-compliances associated with unprotected beam pockets and system supervision, and states that this modification is identified in Table S-2.

Please identify the specific plant modification item to which this applies.

NSPM Response (FPE RAI 03):

The plant modification described in LAR Attachment A, Section 3.10.1, Gaseous Suppression Systems, is the Proposed Modification for FA18 described in Item # 8 in Table S-2. This

modification will resolve the non-compliance with NFPA 72E (Automatic Fire Detectors) associated with unprotected beam pockets and system supervision. This modification also modifies the Ionization Fire Detection system to provide two zones of coverage in the Relay Room and P250 Computer Room, and also modifies the CO2 fire suppression system to actuate if both Ionization zones detect a fire in lieu of heat detectors.

The NFPA code issue listed in Attachment A, Section 3.10.1 under "Items for Implementation" was identified in the NFPA 72E code compliance evaluation, which is not currently referenced in Section 3.10.1 under "Plant Documentation." Section 3.10.1 should be revised to reflect evaluation FPEE-11-048, Revision 0, Code Compliance Review, NFPA 72E Automatic Fire Detectors, 1974, 1982, 1987, Detector Spacing and Location for Plant Areas/Systems Not Addressed in FPP-5, Revision 2.

In addition, the "Compliance Statement" column should also include the compliance basis "Complies with Use of Existing Engineering Equivalency Evaluation." The "Compliance Basis" should be revised to include applicable sections from FPEE-11-048 to demonstrate compliance with the requirements section 3.10.1.

FPE RAI 04

NFPA 805, Section 2.4.3.3, states that the use of the Fire Risk Evaluation performance-based approach requires that "The PSA [probabilistic safety assessment] approach, methods and data shall be acceptable to the AHJ [authority having jurisdiction, which is the NRC]."

Attachment S, Table S-2, Plant Modification Item 5, identifies the installation of a Very Early Warning Fire Detection System (VEWFDS) to monitor the conditions inside low voltage cabinets located in fire compartment FA 18 to reduce the likelihood of fire propagation outside the cabinets.

Please provide a more detailed description of the proposed modification including:

[This RAI includes Subparts a through g, as shown below along with NSPM responses]

NRC Request (FPE RAI 04.a):

a. *The NFPA code(s) of record, proposed installation configuration (inside cabinets or area-wide, common piping or individual cabinet piping), and the use of equipment manufacturers recommendations regarding design, installation, and piping.*

NSPM Response (FPE RAI 04.a):

a. The proposed modification identified in Table S-2 Item 5 is to install an air aspirating incipient detection system inside low voltage (less than or equal to 250V) cabinets in the Relay Room, Fire Area 18. This modification is still in the design phase, however, the following is being planned at this time. Common piping will be used for groupings of cabinets. The system will be designed to meet the sensitivity criteria in NFPA 76. The system will alarm at the existing fire alarm control panel in the Control Room and will be designed, installed, and maintained in accordance with NFPA 72 and manufacturer's recommendations. The applicable codes at the time of design will be used. The

guidance in FAQ 08-0046 Closure Memo dated 11/23/2009 (ML093220426), along with NRC White Paper: Very Early Warning Detection Systems Rev. 1, dated 7/22/2014, will be followed for system design.

NRC Request (FPE RAI 04.b):

b. The acceptance testing, sensitivity and setpoint control(s), alarm response procedures and training, and routine inspection, testing, and maintenance that will be implemented to credit the new incipient detection system.

NSPM Response (FPE RAI 04.b):

b. Factory and Site Acceptance testing will address transport time (maximum of 60 seconds from most remote sampling port), and sensitivity thresholds for alert (0.2% per foot obscuration) and alarm (1.0% per foot obscuration) at each sampling port. Alarm Response Procedures and associated training will be revised and/or developed to describe actions to be taken in response to alert and alarm signals. Inspection, testing, and maintenance procedures will be developed in accordance with equipment manufacturer's recommendations and NFPA 72. Training will be developed with the assistance of manufacturer's recommendations for plant operations, inspection, testing and maintenance groups on the air aspirating incipient detection system.

NRC Request (FPE RAI 04.c):

c. The configuration and design control process that will control and maintain the setpoints for both alert and alarm functions from the VEWFDS.

NSPM Response (FPE RAI 04.c):

c. Configuration and design control process that will control and maintain the setpoints for both alert and alarm functions from the VEWFDS will be in accordance with the site modification process and procedures.

NRC Request (FPE RAI 04.d):

d. The instructions that will be required of the first responders until the degrading component is repaired, the cabinet is de-energized, or the alarm is satisfactorily reset.

NSPM Response (FPE RAI 04.d):

d. Alarm Response Procedures for fire alarms and associated training will be revised and/or developed to describe actions to be taken in response to alert and alarm signals. These actions are still being developed and will be identified as part of the modification process.

NRC Request (FPE RAI 04.e):

e. The credit taken in the Fire PRA [probabilistic risk assessment] for the VEWFDS in assessing the risk in the fire areas where the system is credited.

NSPM Response (FPE RAI 04.e):

e. Credit for incipient fire detection system in the Fire PRA is modeled as a multiplicative factor to the fire ignition frequency. The credited multiplication factor is 0.02 and is derived using the methods in NUREG/CR-6850 Supplement 1.

NRC Request (FPE RAI 04.f):

f. Whether this installation and the credit that will be taken will be in accordance with each of the method elements, limitations and criteria of NUREG/CR-6850, Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements," Chapter 13, and Frequently Asked Question (FAQ) 08-0046, including the closeout memorandum.

NSPM Response (FPE RAI 04.f):

f. Credit was taken in accordance with limitations and criteria of NUREG/CR-6850, Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements," Chapter 13, and Frequently Asked Question (FAQ) 08-0046, including the closeout memorandum.

NRC Request (FPE RAI 04.g):

g. Justification for any deviations from NUREG/CR-6850, Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements," Chapter 13, and FAQ 08-0046, including the closeout memorandum.

NSPM Response (FPE RAI 04.g):

g. No deviations from NUREG/CR-6850, Supplement 1, exist within the credit for modeling of the VEWFDS.

FPE RAI 05

NFPA 805, Section 3.5.16, requires "The fire protection water supply system shall be dedicated for fire protection use only." In Attachment L of the LAR, Approval Request 1, the licensee identifies the fire protection water supply system at PINGP may periodically be utilized to supply water for non-fire protection purposes. The LAR states the fire water system can be aligned for screen wash system use and "emergency uses," and as such it does not meet the requirement or allowed exceptions.

a. *The Approval Request only addresses use of fire water for the screen wash in the "Basis for the Request," and does not address the bases for aligning the fire water for "other emergency uses."*

Please provide a description of "other emergency uses" that may be in addition to screen wash purposes.

For all non-fire use uses, in accordance with 10 [CFR] 50.48(c)(2)(vii), include a discussion on potential impacts to the nuclear safety and radiological release performance criteria. In addition, describe how safety margin is maintained and discuss how the three elements of defense-in-depth are met.

NSPM Response (FPE RAI 05):

a. As described in the RAI, Attachment L to the LAR includes a request for approval to use fire protection water for non-fire protection purposes, including screenwash and "other emergency uses." NSPM would like to clarify that "other emergency uses" should be replaced by "spent fuel pool makeup." Attachment L has been revised to reflect this change and is included in Enclosure 3 to this letter. The revised Attachment L addresses potential impacts to the nuclear safety and radiological release performance criteria, and also includes a discussion regarding safety margin and the three elements (echelons) of defense-in-depth.

FPE RAI 06

NFPA 805, Section 3.6.1, Standpipe and Hose Stations, requires that "For all power block buildings, Class III Standpipe and hose stations shall be installed in accordance with NFPA 14, Standard for the Installation of Standpipe, Private Hydrants and Hose Systems."

In Attachment A of the LAR, Section 3.6.1, Standpipe and Hose Stations, lists NFPA 14 code compliance evaluations. The summary statement in the LAR for the code compliance evaluations identifies the need to resolve one unacceptable deviation (NFPA 14-1986) and two outstanding Action Requests (NFPA 14-1969) resulting from the NFPA 14 reviews. These open items listed in the LAR do not appear to have implementation items identified in Attachment S.

Please provide more detail with regard to the deviation and two outstanding items, including the identification of implementation items required for these items, as appropriate or necessary to meet NFPA 805. If implementation items for these code deviations are deemed unnecessary to meet NFPA 805, provide additional justification.

NSPM Response (FPE RAI 06):

The deviation from NFPA 14-1986 identified in Attachment A of the LAR is that the pipe sizes in the D5/D6 Standpipe and Hose Station System do not meet the minimum size requirements in Table 2-1.1, based on total pipe length. As described in FPEE 11-050, this deviation will be resolved by performance of a hydraulic calculation. Performance of this hydraulic calculation is being tracked by AR 01470080-02, and its completion will be verified through a new Implementation Item in Table S-3, item #64, as follows:

Item 64:
Update code compliance reviews to document resolution of identified open items.

The two outstanding items for NFPA 14-1969 identified in Attachment A of the LAR are as follows:

1) Section 5.10, Hangers. An additional hanger was determined necessary to prevent excessive movement of a standpipe, as described in FPEE NFPA 14-1969. This hanger is identified in Engineering Change (EC) 18011, and the installation will be completed as part of a new Proposed Modification #40 that will be added to Table S-2 of the LAR. .

2) Section 5.11, Gauges. Fire Protection standpipes in the older portions of the PINGP plant were not installed with pressure gauges at the top of the hose station risers. This includes the Screenhouse, Turbine Building, Auxiliary Building, and Old Administration Building. The NFPA 14-1969 code compliance review will be updated to include the resolution of this open item as part of the proposed new Implementation Item #64 in Table S-3, "Update code compliance reviews to document resolution of identified open items."

RAI Responses – Safe Shutdown Analysis (SSA)

SSA RAI 01

NFPA 805, Section 2.4.2, Nuclear Safety Capability Assessment, requires licensees to perform a nuclear safety capability assessment (NSCA). Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," endorsed the guidance in NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Chapter 3, as one acceptable approach to perform an NSCA. In Attachment B of the LAR, the alignment basis for NEI 00-01, Attribute 3.1.1.4, is "Not in Alignment [but Prior NRC Approval]." The licensee stated that the Fire PRA employs the use of a hot shutdown (HSD) panel in conjunction with other controls when required to abandon the Control Room (CR) and that the HSD panels and controls do not meet the definition of a primary control station (PCS) as defined by RG 1.205. Additionally, the licensee stated that actions to enable the HSD panel and establish required controls are listed in the PRA Alternative Shutdown Notebook.

[This RAI includes Subparts a through e, as shown below along with NSPM responses]

NRC Request (SSA RAI 01.a):

a. *RG 1.205 Section C.2.4 describes two cases where operator actions taken outside the main control room may be considered as taking place in the PCS: (a) controls for a system or component specifically installed to meet the "dedicated shutdown" option in Section III.G. 3 of Appendix R and (b) controls for some systems and components that have been modified to meet the "alternative shutdown" option in Section III.G.3.*

   Please provide a discussion of the HSD panel and what attributes of PINGP design and procedures do not meet the definition of PCS as defined in RG 1.205.

NSPM Response (SSA RAI 01.a):

a. The PINGP Hot Shutdown Panels were determined to not meet the definition of a Primary Control Station using the criteria in Regulatory Guide 1.205 and FAQ 07-0030. Regulatory Guide 1.205, Section 2.4 Recovery Actions states: "The staff has identified two cases where operator actions taken outside the main control room may be considered as taking place at a primary control station. These two cases involve dedicated shutdown or alternative shutdown controls, which have been reviewed and approved by the NRC." FAQ 07-0030 describes that the Primary Control Station should have the requisite system and component controls, plant parameter indications and communications so that the operator can adequately and safely monitor and control the plant using the alternative shutdown equipment.

   The PINGP Hot Shutdown Panels have not been reviewed and approved by the NRC. These panels were installed in 1972 before the plant began operation and they were designed for General Design Criteria 19, Control Room. Generic Letter 81-12 states that any modifications that the licensee plans in order to meet the requirements of Section III.G.3 of Appendix R must be reviewed and approved by the NRC. PINGP stated in a letter to the NRC dated June 30, 1982 (Reference S-1), that since no modifications were required for Alternate Shutdown, no plans were submitted to the NRC for review.

The PINGP Hot Shutdown Panel does not meet the definition of a primary control station because:

- It has not been reviewed and approved by the NRC, and
- The Hot Shutdown Panel contains required indication, but it does not contain the required system and component controls, plant parameter indications, and communications so that the operator can adequately and safely monitor and control the plant using alternative shutdown equipment.

NRC Request (SSA RAI 01.b):

b. *As described in FAQ 07-0030, "Establishing Recovery Actions," Revision 5, if a licensee proposes to make modification(s) to their previously approved strategy, it may obtain NRC approval of a new PCS strategy. NRC approval of the new PCS strategy can be obtained by providing the information required in FAQ 07-0030 for either 1) Option 1, to design and install a primary control station(s) in accordance with the guidance and requirements of the existing Fire Protection licensing, or 2) Option 2, to develop the design and analyze the primary control station(s) using the performance-based approach and provide the necessary evaluation.*

If the actions to enable the HSD panel and establish the required controls are credited in the performance based analysis in the PRA Alternative shutdown notebook as PCS actions, then provide a detailed description of the modification to the dedicated or alternative shutdown strategy sufficient for the staff to verify that the strategy meets the attributes provided in Section C.2.4 [i.e., electrical independence, command and control, instrumentation, actions necessary to enable (if required), etc.].

NSPM Response (SSA RAI 01.b):

b. Although the Hot Shutdown Panels in the Auxiliary Feedwater (AFW) Pump rooms are the primary command and control center after abandoning the control room due to fire (see response to subpart c below for more detail), the Hot Shutdown Panels are not credited as the Primary Control Station for the purpose of determining Recovery Actions. All actions taken outside the control room after abandoning due to fire are considered to be Recovery Actions and the additional risk of those actions is included in the Fire PRA Analysis.

NRC Request (SSA RAI 01.c):

c. *For a fire requiring CR abandonment, please describe the actions necessary to enable the HSD panel and/or PCS to establish required controls, including required local/remote indications.*

Clarify if these actions are identified in LAR Attachment G.

NSPM Response (SSA RAI 01.c):

c. After the decision to abandon the control room has been made by the Unit 1 Shift Supervisor, operators make the announcements, don an SCBA if necessary, and travel

to the Hot Shutdown Panels in the AFW Pump room (per procedure F5 Appendix B, "Control Room Evacuation (Fire)"). The Hot Shutdown panels are credited for process monitoring indication only. There are no switch manipulations required to activate, or isolate the process monitoring circuits at Hot Shutdown Panels as the process monitoring indication is always active and isolated from the effects of fire in the Control Room, and Relay and Cable Spreading Room via always-active instrumentation isolation devices. Therefore, no actions are identified in Attachment G.

NRC Request (SSA RAI 01.d):

d. When CR abandonment is necessary, please describe the command and control structure of the operating crew, including locations, communications and coordination required between the decision makers (SROs) and field operators performing recovery actions.

For those actions that are symptom-driven, discuss the instrumentation/indications used as cues to determine that the action is required.

NSPM Response (SSA RAI 01.d):

d. The command and control structure of the operating crew is described in PINGP Procedure F5 Appendix B, Control Room Evacuation (Fire). When the decision is made to abandon, operators take individual attachments to the procedure and travel to various parts of the plant to perform local actions to support achieving and maintaining Mode 3 Hot Standby. The last step of their attachment directs them to the AFW Pump room to report to the Unit 1 Shift Supervisor. Face-to-face communication is credited for required communication.

Procedure F5 Appendix B, states symptoms that may necessitate the need to implement alternate shutdown as evidenced by: "a loss of Control Room control of critical plant functions which cannot be adequately addressed by ARP, AOP, IFG or EOP response actions." There are no specific symptoms identified that would force abandonment; the decision is left to the judgment of the Unit 1 Shift Supervisor. This discretion is appropriate given the risk associated with abandoning all control from the Control Room to rely solely upon one train of Safe Shutdown Equipment outside of the Control Room.

The table directly below lists the symptom-driven actions and their respective cues (indication):

Required Indication	Action Driven by indication
1LI-433C 2LI-433C	Charging pump speed is adjusted using level indication, obtained from 1LI-433C and 2LI-433C for the Unit 1 and Unit 2 pressurizers respectively, to maintain pressurizer level as necessary.

Required Indication	Action Driven by indication
1LI-487A 2LI-487A FI-18032 FI-18035	TDAFW pump discharge flow is adjusted by manually throttling MV-32238 (11 AFW TO 11 SG MV) for Unit 1 or MV-32246 (22 AFW TO 21 SG MV) for Unit 2, >200 gpm as indicated by FI-18032 and FI-18035 for the 11 and 22 TDAFW pumps respectively, to maintain S/G level, obtained from 1LI-487A and 2LI-487A for 11 and 21 S/Gs respectively, as necessary.
55001	IF 55001 indicates a loss of cooling water pressure with D1 running, then stop #1 Emergency diesel generator.
11330 (11 CLG WTR Strainer Inlet PI) 11395 (12 CLG WTR Strainer Inlet PI) 11397 (21 CLG WTR Strainer Inlet PI) 11625 (22 CLG WTR Strainer Inlet PI) 11370 (11 CLG WTR Strainer Outlet PI) 11396 (12 CLG WTR Strainer Outlet PI) 11523 (21 CLG WTR Strainer Outlet PI) 11626 (22 CLG WTR Strainer Outlet PI)	Check the differential pressure across the 11, 12, 21, and 22 cooling water strainers via their respective inlet and outlet pressure indicators. IF the DP across the affected strainer is greater than 4 psid, and it is not backwashing, THEN attempt to manually operate the strainer motor per F5 App. B Attachment C step BB.
PI-11054 PI-11081	Switchover to Clg Wtr from CST for AFWP suctions will be necessary when AFWP suction pressure approaches 3 psi. IF it becomes necessary to switchover the AFWP suction to Clg Wtr, THEN perform the following: 1. OPEN the following MCC Breakers: a. 1A1-A2, 11 TD AFW PMP SUCT CLG WTR SPLY MV-32025 b. 2A2-A3, CLG WTR TO 22 TD AFW PMP SUCT MV-32030 2. Locally OPEN the following MVs: a. MV-32025, 11 TD AFW PMP SUCT CL SPLY MV b. MV-32030, 22 TD AFW PMP SUCT CL SPLY MV

Other actions not listed which are driven by local indication include: manually re-positioning valves due to local valve indication, locally starting/stopping motors due to local indications, and locally re-positioning breakers due to local breaker indication.

NRC Request (SSA RAI 01.e.i):

e. *In the alignment basis for Attribute 3.1.1.4 of NEI 00-01, the licensee stated that prior to abandoning the CR, action is taken on the control board to close the power-operated relief valves (PORVs), and additional actions are taken at a switch panel (to be installed as modification item 27 in Attachment S of the LAR) to isolate excess letdown, head vents, pressurizer vents, the pressurizer PORV, and pressurizer heaters.*

(i) *Please clarify if the actions at the new switch panel are required to be successful in order to meet the nuclear safety performance criteria of Section 1.5.1(b) associated with RCS inventory and pressure control. If so, describe the methodology for ensuring that manual actuation of these switches is successful and discuss how the manual actuation aligns with Attribute 3.1.1.10 of NEI 00-01.*

NSPM Response (SSA RAI 01.e.i):

(i) NSPM would like to clarify that the action described in the Alignment Basis for Attribute 3.1.1.4 in Attachment B is to close the PORV block valves (MV-32195, MV-32196, MV-32197, and MV-32198) on the main control board. The action to isolate the PORV control valves (CV-31231, CV-31232, CV-31233, and CV-31234) at the new isolation panel installed as part of Modification Item #27 in Attachment S, Table S-2, is not credited in the Nuclear Safety Performance Criteria; this modification is for risk reduction purposes only. The action to de-energize the PORV control valves outside the control room (at DC switch panels in the Battery Rooms) is the credited action to assure that the PORVs are closed and remain closed, thus meeting the requirements of NSPC 1.5.1(b).

The new isolation switches are being installed to reduce Fire PRA risk only, and are not credited by the NSCA; and therefore, the provisions of Attribute 3.1.1.10 of NEI 00-01 do not apply. PINGP is transitioning the exemption that allows closure of the PORV block valves from the control room prior to abandonment and de-energizing the PORV control valves outside the control room at the DC switch panels to ensure the PORV relief path is isolated (described in LAR Attachment K, "Appendix R Exemption, Control Room, Use of repair to remove fuses (III.G.1 criteria), Fire Area 13;" and as clarified in Attachment T of the PINGP LAR.

NRC Request (SSA RAI 01.e.ii):

(ii) *Please provide a description of the switch design and location. Include in the discussion how the switch provides the desired electrical isolation for the subject components.*

NSPM Response (SSA RAI 01.e.ii):

(ii) The design function of the PORV switch is to preclude or isolate (mitigate) hot shorts from causing spurious opening of the PORVs. This will be accomplished by breaking the circuit power connection of the solenoid valve that is not affected by a fire in the area. The design will be such that hot shorts at this switch in the control room would not be able to cause spurious operation of the components. This modification is for risk reduction only, and is not credited in the NSCA. This modification, Table S-2, item #27, is still in the design phase.

References:

S-1 NSP Letter, D. Musolf to Director, Office of Nuclear Reactor Regulation, "Fire Protection Safe Shutdown Analysis and Compliance with Section III.G of 10 CFR Part 50, Appendix R, Including Requests for Relief," dated June 30, 1982 (NRC PDR No. 8207060240 820630).

SSA RAI 02

The Executive Summary in the LAR states that the PINGP transition process was performed in accordance with RG 1.205, Revision 1. Regulatory Position 2.3.2, Previously NRC-Approved Alternatives to NFPA 805, Section 4.2.3, Deterministic Requirements, in RG 1.205, Revision 1, states that in accordance with NFPA 805, Section 2.2.7, licensees may use existing exemptions ... to demonstrate compliance with the specific deterministic fire protection design requirements in Chapter 4 of NFPA 805, provided the NRC staff determines that the licensee has acceptably addressed the continued validity of any exemption ... in effect at the time of the NFPA 805 license amendment application and that the exemption ... does not involve a recovery action as defined in NFPA 805, Section 1.6.52, that is used to demonstrate the availability of a success path for the nuclear safety performance criteria.

In Attachment K of the LAR, the licensee stated that the previously approved exemption to allow manual removal of fuses from the PORV control circuit in the event of a fire in the Control Room (CR) (Fire Area 13) will be transitioned in NFPA 805 as clarified in LAR Attachment T, to allow opening of disconnect switches installed subsequent to the exemption approval, in lieu of pulling fuses. Clarify which action, or both, is credited for disabling the PORV control circuits.

Please describe the procedural steps and the feasibility analysis performed for these actions.

NSPM Response (SSA RAI 02):

To clarify which action PINGP is crediting to disable PORV control circuits, NSPM is revising Attachment T to the LAR, which clarified a previously approved exemption. The requested clarification will allow for the opening of disconnect switches in lieu of the original exemption action of pulling fuses as a follow-on action taken outside of the Control Room, to prevent fire-induced spurious opening of the PORVs. The manual operation to open disconnect switches, demonstrated by PINGP to be feasible and reliable, is simpler than pulling fuses and is equivalent in intent and function. A revision to Attachment T to describe this change is included in Enclosure 4 to this letter.

Table S-2 Modification Item #27, to install isolation switches for the PORVs, etc. is credited in the Fire PRA for risk reduction purposes only, and as such, will not eliminate the need to take the follow-on action of opening the disconnect switches for fire scenarios involving a Control Room evacuation.

The procedural steps for assuring PORV isolation during a Control Room fire are as follows:

PINGP Procedure F5 Appendix B, Control Room Evacuation (Fire)

- Attachment C U1 RO Actions: Step F. CLOSE pressurizer PORV block valves. (Performed in the control room after reactor trip, turbine trip, RCP trip, MFW trip, and MSIV closure)
- Attachment D U2 RO Actions: Step F. CLOSE pressurizer PORV block valves. (Performed in the control room after reactor trip, turbine trip, RCP trip, MFW trip, and MSIV closure)
- Attachment B U2 Shift Supervisor Actions:
  - o Step A. Don SCBA
  - o Step B. Proceed to 11 Battery Room (Unit 1, Train A)

- o Step C. Turn off switch PNL 11-8 which powers PNL 171 which powers CV-31232 U1 PZR PORV A
- o Step D. Proceed to 12 Battery Room (Unit 1, Train B)
- o Step E. Turn off switch PNL 12-8 which power PNL 181 which powers CV-31231 U1 PZR PORV B
- o Step F. Proceed to 22 Battery Room (Unit 2, Train B)
- o Step G. Turn off switch PNL 22-10 which powers PNL 281 which powers CV-31233 U2 PZR PORV B
- o Step H. Proceed to 21 Battery Room (Unit 2, Train A)
- o Step I. Turn off switch PNL 21-10 which powers PNL 271 which powers CV-31234 U2 PZR PORV A

The feasibility of these actions has been evaluated by existing PINGP Calculation, GEN-PI-055, "Manual Action Feasibility." Appendix D to this calculation provides a timeline to perform these actions as described in PINGP procedure F5 Appendix B, "Control Room Evacuation (Fire)." These actions (from Attachments B, C, and D, as listed above) have been walked down by multiple operating crews to validate the timing, and are included in regular training:

- The Attachment B, U2 Shift Supervisor steps A-I are completed in 9 minutes.
- The Attachment C, U1 RO action to close Unit 1 PORV block valves is completed in 1 minute.
- The Attachment D, U2 RO action to close Unit 2 PORV block valves is completed in 1 minute.

LAR Attachment S, Table S-3 Item 53 was inadvertently deleted in the 2014 LAR Supplement; this item will be re-instated and revised as follows, to update GEN-PI-055 to reflect PINGP's transition to NFPA 805, and to document how the feasibility criteria of FAQ 07-0030 is met:

> Item 53:
> Update GEN-PI-055, "10CFR50 Appendix R Manual Action Feasibility Study," to reflect *PINGP's* transition to NFPA 805, including addition of new recovery actions, actions to maintain safe and stable conditions, and to document how the criteria, as defined by FAQ 07-0030, are met.

SSA RAI 03

NFPA 805, Section 2.4.2, Nuclear Safety Capability Analysis, requires licensees to perform a nuclear safety capability analysis (NSCA). In RG 1.205, the NRC staff states that one acceptable approach to perform the NSCA is to follow the guidance in NEI 00-01, Chapter 3. Attribute 3.2.1.5 of NEI 00-01 states that instrument circuits (e.g., resistance temperature detectors, thermocouples, pressure transmitters, and flow transmitters) should be assumed to fail upscale, midscale, or downscale as a result of fire damage, whichever is worse, and that an instrument performing a control function is assumed to provide an undesired signal to the control circuit. In Attachment B of the LAR, Section 3.2.1.5, the licensee stated that PINGP assumes that instrumentation circuits fail in their worst-case positions when damaged by the fire unless an analysis was performed to show that the failure mode is incredible.

Please describe the analysis method used to determine the failure mode is incredible and provide an example of the types of instruments that would be analyzed using this method.

NSPM Response (SSA RAI 03):

In general, PINGP assumed that instrumentation circuits involving shielded twisted pair cables will fail in the worst-case mode (upscale, downscale or midrange); however, in a very few cases, additional analysis was performed in the PINGP Non-Power/NSCA Operations Review for NFPA 805, EC 20612, to determine that the worst-case failure mode is incredible, and to identify the true fire-induced failure mode. A description of the methodology and evaluation criteria, along with an example circuit, is provided below.

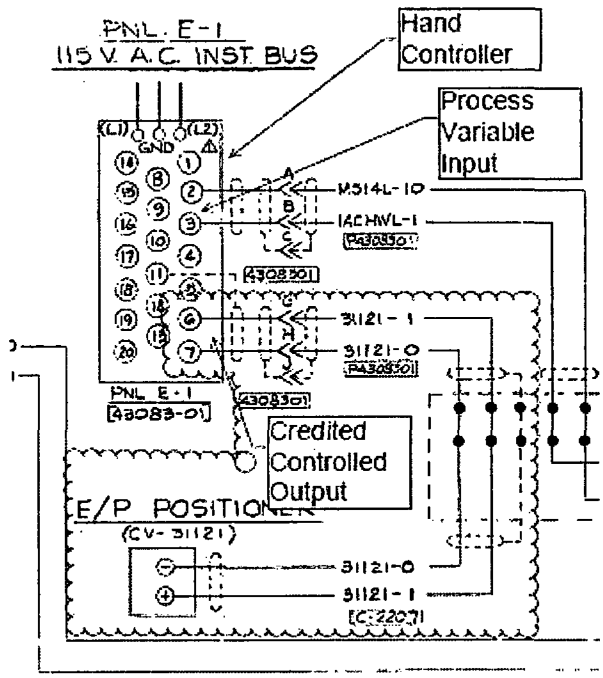
A fire that directly affects cabling in instrumentation circuits can produce open circuits, conductor-to-conductor bolted or resistive hot shorts (inter-cable or intra-cable), and shorts to ground. As stated in the RAI, these fire-induced faults may drive the process variable upscale, downscale, or midscale. To determine that the worst-case failure mode is incredible, instrument loops were evaluated and determined to meet all of the following criteria:

1. Fire-induced faults that cause an open circuit on the target conductors cannot fail the credited function / component in an undesired position/state. This criteria is the fundamental building block (others to follow) for the concept that a loss of signal to the credited function / component shall not be capable of causing an undesired consequence.

2. The circuit must be capable of being positively isolated from any signals (interlocks, process variables, etc.), that could cause an undesired output or positioning. For emphasis, manual isolation of the circuit is required so that the circuit may be placed in its lowest output (equivalent to fail-safe) state. The circuit CANNOT require manual adjustment beyond its lowest output state, and cannot rely upon, nor can it be affected by any automatic functions, interlocks, etc. Additionally, by requiring that the circuit be in its lowest state, leakage current away from the circuit and wire-to-wire resistive shorting is assured to not be of concern. This criteria is in keeping with the guidelines provided in NUREG/CR-7150, Volume 1, Section 5.0.

3. Open circuits, conductor-to-conductor bolted or resistive hot shorts (inter-cable or intra-cable), and shorts to ground must be demonstrated to have no effect on the credited portion of the circuit, or be demonstrated to not be credible (e.g., due to the physical design, credited state of the circuit, etc.).

    a. Example: In the partial schematic directly below, an open circuit on the conductors between the hand controller and the E/P (clouded conductors), cannot cause the valve to fail in the undesired position.



4. The device performing the isolation and controlled level output (e.g., hand controller), cannot be located in the area where the fire is occurring. This device must be capable of positively isolating any process variable inputs and interlocks that may inadvertently cause an undesired output or positioning.

5. Components exposed to the fire (e.g., cable, valve, E/P, positioner), cannot be credited to provide outputs or positions that differ from the fail-safe (low/no-signal) output or position. Additionally, the fire-exposed components must not contain electronic systems that are susceptible to producing sustained output spiking, or are capable of independently generating an output signal that may be out of proportion to the input signal.

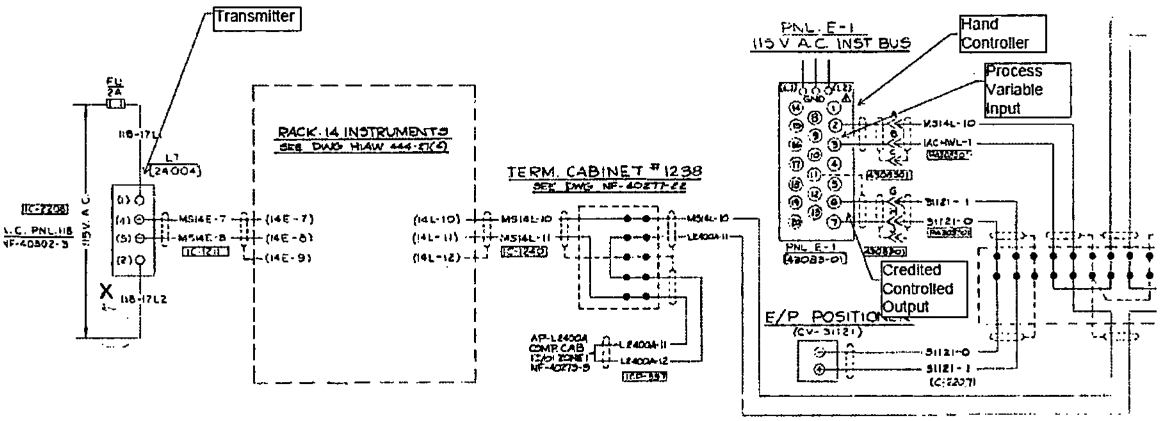
   a. Example: An E/P (I/P) that operates on the principle that a magnetic coil positions a mechanical beam that regulates a mechanical air relay (force-balance), would be acceptable as the coil is not capable of generating a sustained signal greater than it's input. An E/P that operates on the principle that an analog or digital input signal is converted to an output signal via a microcontroller in the E/P, would not be an acceptable configuration as the output is capable of being driven independently of the input.

   b. Additionally, the component must not have a fail / hold last state function (sometimes referred to as a lock-up function).

6. For the fire area of concern (location of the fire), any given fault(s) cannot cause an isolation of the loop regulating device (e.g., transmitter, controller), such that the loop power supply can apply full current to the credited portion of the circuit. This criterion is fully dependent upon the specific design of the loop circuit.

   a. Example: As can be seen from the drawing directly below, the power supply to the credited controlled output is integral to the hand controller itself (all other portions of the

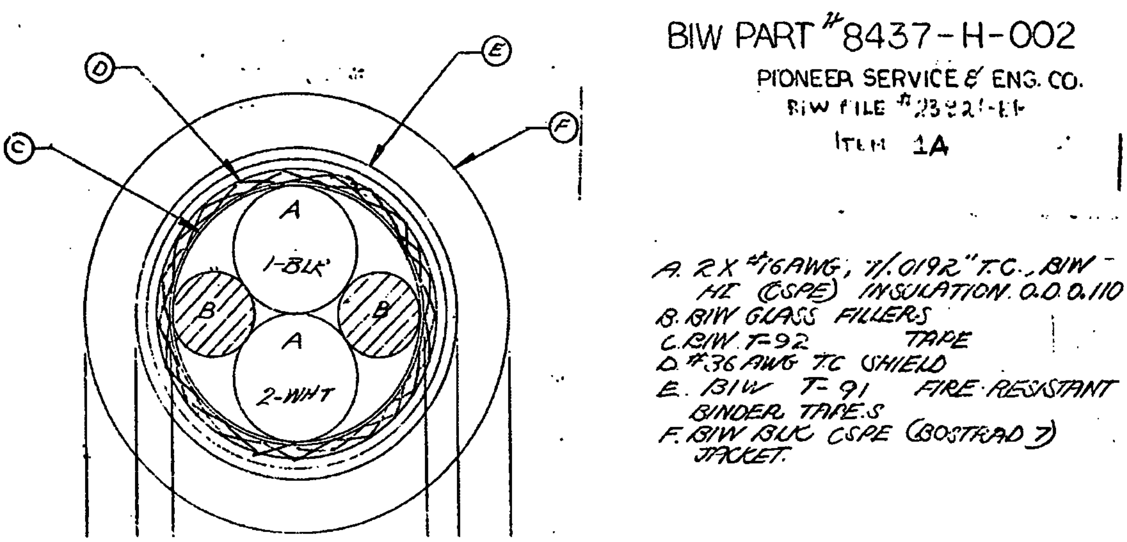
circuit will be isolated in manual mode); therefore, when the hand controller is taken to manual, cable faults between the hand controller and the E/P cannot isolate the hand controller (regulating portion of the credited circuit) from the power supply.



7. The portion of the circuit exposed to the fire cannot contain more than a single twisted pair. This criteria prevents intra-cable shorts from occurring that could increase the signal independent of the hand controller from wire to wire shorts and internal re-referencing of the shield.

8. The credited cable shield must be of a braided type (no foil). This criteria, coupled with the criteria as described below, prevents inter-cable faults from falsely increasing the signal on the circuit.

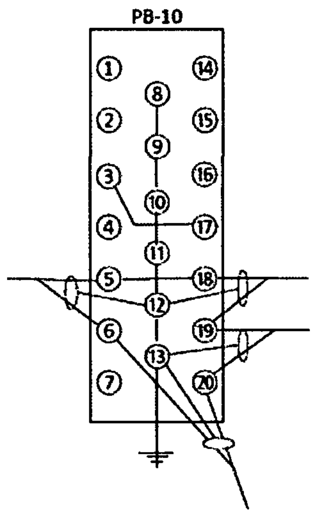
   a. Example: The cable shown in the drawing directly below is a typical single pair, shielded twisted pair cable with a braided shield.



9. The shield must be continuous and must be tied to ground. This criteria, coupled with the criteria described above, prevents inter-cable faults from falsely increasing the signal on the

credited portion of the control circuit. Open circuiting of the cable due to cleaving (e.g. falling debris) may cause a loss of ground on the shield conductor; however, this would also cause a loss of reference of the internal conductors and thus, re-referencing of a cleaved (ungrounded shield) would not be able to cause an undesired effect to the circuit. Additionally, the circumferential design of a braided shield conductor aids in its ability to remain in continuity in the case of a partial cleave (i.e., where only one internal conductor is cleaved).

a. Example: The block diagram directly below demonstrates a typical shield termination scheme for the plant. Shields are continuous and tied to ground.



Conclusion:

The as-built circuit must meet all of the above criteria for a fault (fault consequence) on an instrument loop to be considered "incredible." This method is in keeping with the guidance provided in NUREG/CR-7150, Section 5.0.

LAR Attachment S, Table S-3 Item 51 will be updated to include addition of this specific methodology to Engineering Manual 3.4.3, "Safe Shutdown Circuit Analysis." Item 51 will be changed as follows:

Item 51:
Revise EM 3.4.3, "Safe Shutdown Circuit Analysis" to incorporate applicable details of vendor document EPM-DP-EP-004 as well as the detailed methodology for analyzing shielded twisted pair instrumentation and controls circuits as referenced in EC 20612, "PINGP Non-Power/ NSCA Operations Review for NFPA 805."

SSA RAI 04

NFPA 805, Section 1.3.1, Nuclear Safety Goal, states: "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."

NFPA 805, Section 1.5.1(d), Vital Auxiliaries, states: "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."

In Section 4.2.1.2 of the LAR, the licensee stated that CR temperature will remain below equipment limits for up to 36 hours with actions taken only within the CR itself. A portable fan may be used to maintain temperatures below equipment limits indefinitely. If required, the portable fan will be powered by a designated welding receptacle or a 480-VAC [volts alternating current] portable generator located outside the building.

Please provide the following additional information:

[This RAI includes Subparts a through f, as shown below along with NSPM responses]

NRC Request (SSA RAI 04.a):

a. *Describe the steps taken to maintain CR temperature below equipment limits for 36 hours.*

NSPM Response (SSA RAI 04.a):

a. The steps to maintain CR temperature below equipment limits for 36 hours are defined in C37.9 AOP1, "Loss of Control Room Cooling," and include opening doors / vent paths from the Control Room. The evaluation identified in LAR Section 4.2.1.2 (EC 23925) validated that the required actions specified in C37.9 AOP1 (opening doors / vent paths) will maintain control room temperature below the critical temperature limit for 36 hours.

NRC Request (SSA RAI 04.b):

b. *Clarify if the use of the portable fan and the associated welding receptacle or portable generator is credited for achieving and maintaining safe and stable conditions. If so, provide the justification for excluding this recovery action in Attachment G of the LAR.*

NSPM Response (SSA RAI 04.b):

b. As stated in LAR Section 4.2.1.2, "A PINGP thermal-hydraulic analysis was performed for a mission time of 24 hours to assure that safe and stable conditions can be achieved within that time period." The portable fan is used at approximately 36 hours, which is after safe and stable conditions have been achieved (within 24 hours), and therefore, the action to power the portable fan is not considered to be a recovery action.

NRC Request (SSA RAI 04.c):

c. *Provide additional details on the storage and usage locations of the 480-VAC portable generator and its potential impact, if any, on the NSCA structures, systems, and components (SSCs) that are in the vicinity of these locations.*

NSPM Response (SSA RAI 04.c):

c. The portable generator will be stored and operated outside the Power Block as defined in Attachment I to the LAR, and therefore, it will not be operated in the immediate vicinity of NSCA SSCs. Details regarding connection of the portable generator to the portable fan will be determined as part of Implementation Item #63 (Table S-3), as revised in the response to SSA RAI 04.d.

NRC Request (SSA RAI 04.d):

d. *Describe the type and quantity of fuel associated with the portable generator and the availability and location(s) of sufficient fuel sources to support maintaining safe and stable conditions for the time period required.*

NSPM Response (SSA RAI 04.d):

d. The portable generator will be fueled with diesel fuel. Attachment S, Table S-3, Item #63 will be revised to have procedural controls for the operation, maintenance, storage, and refueling of the portable generator to support maintaining safe and stable conditions. The revision will be as follows:

   Item 63:
   Provide <u>procedural guidance</u> to connect a <u>diesel powered</u> portable generator <u>located outside the power block</u> to power a temporary fan for the Main Control Room to maintain safe and stable <u>conditions. Additionally, procedural guidance shall be provided for the operation, maintenance, storage, and refueling of the portable generator, and for training and drills.</u>

NRC Request (SSA RAI 04.e):

e. *Justification that refueling the portable generator does not present a fire exposure hazard to NSCA SSCs.*

NSPM Response (SSA RAI 04.e):

e. The portable generator will be located and operated outside the Power Block; therefore re-fueling will be performed outside, and not in the vicinity of NSCA SSCs.

NRC Request (SSA RAI 04.f):

f. *A summary of the procedure guidance for the use of the portable generator to power the portable fan, and how this action aligns with each of the feasibility criteria of FAQ 07-0030 (i.e., training, procedures, drills, etc.).*

NSPM Response (SSA RAI 04.f):

f. Procedural guidance for use of the portable generator to power the portable fan is still under development and will be finalized per the modification process. At this time, guidance is planned to include the following (note that if power is available, the portable fan will be powered from a welding receptacle):
   1. Place a power cart* on the turbine deck (if the portable generator will be needed).
   2. Place the fan to take suction from the Control Room and direct it into the Records Room.
   3. Plug the fan into the power cart (or the welding receptacle, as noted above).
   4. Move the portable generator to a suitable service location.
   5. Connect the power cart to the portable diesel generator.
   6. Start and operate the portable generator, including refueling instructions.
   7. Start and operate the portable fan and power cart.

   * Note that the power cart has not been designed as of this response; however, it is expected that the power cart will contain the properly sized fused disconnect, adequately sized cabling and connections to connect the portable fan to the generator.

   Procedural guidance will also be developed for training and drills.

   These actions, taken to maintain safe and stable conditions, will be evaluated for feasibility against the criteria of FAQ 07-0030 as described by the reinstated and revised LAR Attachment S, Table S-3 Item #53, as described in the response to SSA RAI 02.

SSA RAI 05

NFPA 805, Section 1.3.1, Nuclear Safety Goal, states: "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."

NFPA 805, Section 1.4.1, Nuclear Safety Objectives, states: "In the event of a fire during any operational mode and plant configuration, the plant shall be as follows:

(1) Reactivity Control. Capable of rapidly achieving and maintaining subcritical conditions

(2) Fuel Cooling. Capable of achieving and maintaining decay heat removal and inventory control functions

(3) Fission Product Boundary. Capable of preventing fuel clad damage so that the primary containment boundary is not challenged."

In Section 4.3.2 and Attachment D of the LAR, the licensee provided the results of the evaluation process for Non-Power Operations (NPO) analysis.

Please provide additional details as follows:

[This RAI includes Subparts a, b, and c, as shown below along with NSPM responses]

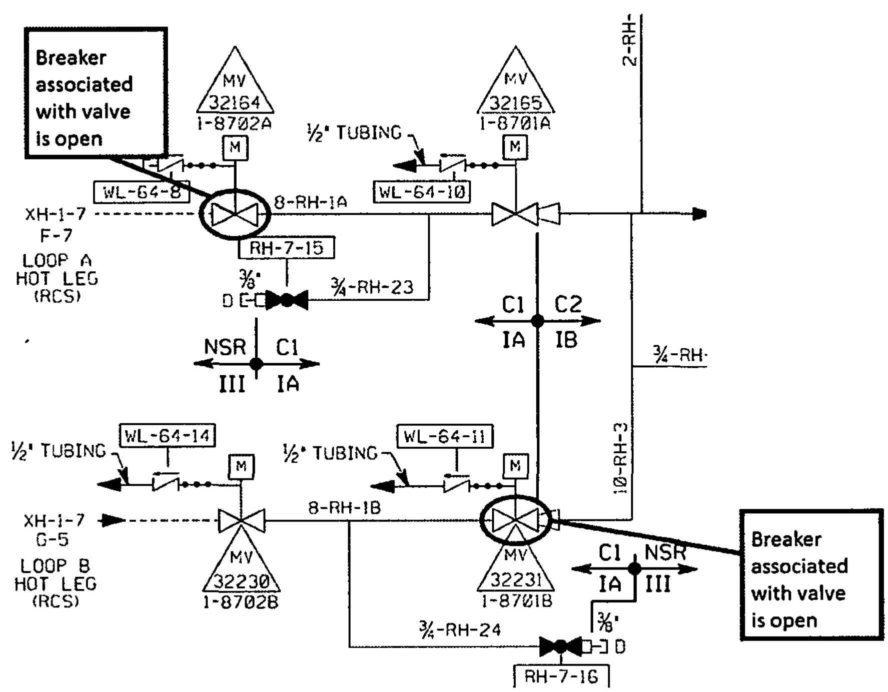
NRC Request (SSA RAI 05.a):

a. *During NPO modes, spurious actuation of valves can have a significant impact on the ability to maintain decay heat removal and inventory control.*

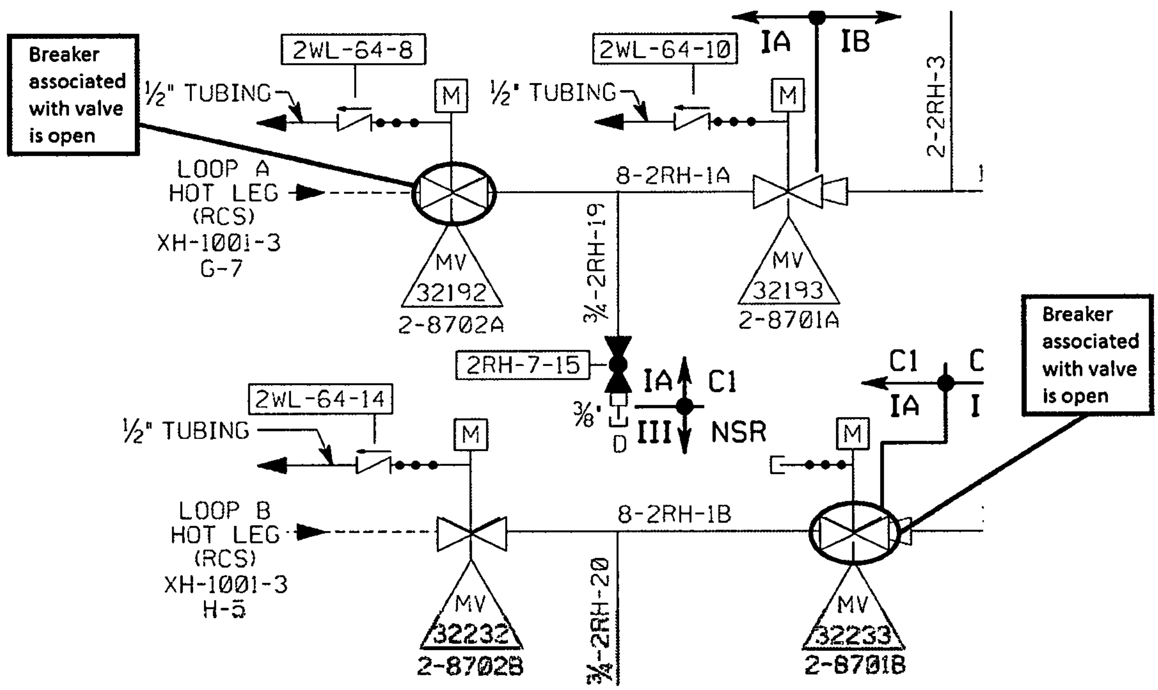
   Provide a description of any actions being credited to minimize the impact of fire-induced spurious actuations on power operated valves (e.g., air-operated valves and motor-operated valves) during NPO (e.g., pre-fire rack-out, actuation of/or pinning of valves, and isolation of air supplies).

NSPM Response (SSA RAI 05.a):

a. During NPO modes, there are four RHR suction valves in each unit where spurious actuations could impact decay heat removal, as shown in the diagrams below. A pre-fire action is taken prior to entering higher risk evolutions to tag open the breakers associated with two of the four RHR suction valves, per procedures 1C1.3-M5 and 2C1.3-M5, "Unit 1 (2) Shutdown to Mode 5", for each unit, MV-32164 (Unit 1), MV-32231 (Unit 1), MV-32192 (Unit 2), and MV-32233 (Unit 2) during shutdown cooling, mode 5, to minimize the impact of fire-induced spurious actuations of RHR suction valves from the RCS Hot Leg. During higher risk evolutions PINGP will use administrative controls listed on page D-10 of the LAR, to minimize the risk of potential fire damage that could impact operation of the following RHR suction valves: MV-32165, MV-32230, MV-32193, and MV-32232.



Unit 1 RHR Suction Valves



Unit 2 RHR Suction Valves

NRC Request (SSA RAI 05.b):

b. During normal outage evolutions, certain credited NPO equipment will have to be removed from service.

Describe the types of compensatory actions that will be used during such equipment down-time and how are they determined to be adequate.

NSPM Response (SSA RAI 05.b):

b. Updates to procedure 5AWI 15.6.1, 'Shutdown Safety Assessment' will ensure that NPO credited equipment is not removed from service during High Risk Evolutions (HRE) without adequate compensatory measures. Plant procedures will provide guidelines and identify compensatory actions that can be taken when fire safe shutdown components are out of service. Hot Work and Safe Shutdown Safety Assessment procedures will be revised during NFPA 805 implementation (Table S-3, Items 37 and 40), and the following types of compensatory actions will be added / retained for fire risk mitigation: Hot Work Restrictions, Transient Combustible Controls, Access Limitations, Automatic Detection and Suppression Systems, Fire Watch Patrols, etc.

Additional Key Safety Function (KSF) pinch points introduced by removal of credited equipment from service will be identified through administrative procedures, shutdown risk management, and work control. Also in the unlikely event that such equipment is deliberately removed from service coincident with a planned or emergent HRE, the

plant's Fire Protection Team will consider appropriate contingency measures to reduce fire risk at the additional locations.

NRC Request (SSA RAI 05.c):

c. In Attachment D of the LAR, the licensee states that operator actions taken to mitigate the loss of a Key Safety function (KSF) are credited in the NPO analysis contained within PINGP Engineering Evaluation EC-20612, "Non-Power/NSCA Operations Review for NFPA 805."

Describe the operator actions credited to maintain KSF and the feasibility analysis performed for these actions.

NSPM Response (SSA RAI 05.c):

c. Types of actions credited to maintain KSF include the following: de-energizing power operated valves to preclude or mitigate spurious operation, adding procedural actions to open a redundant flow path, re-powering an Instrument Bus from the alternate panel, and closing manual valves to isolate flow diversion paths.

The feasibility analysis contained in GEN-PI-055 will be revised as shown in the reinstated and revised S-3 table item #53, as described in the response to SSA RAI 02.

SSA RAI 06

NFPA 805, Section 1.3.1, Nuclear Safety Goal, states: "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."

In Section 4.2.1.2 of the LAR, the licensee stated that the determination of the final state of the safe and stable conditions will be based upon the extent of the fire damage, the inventory remaining in the Refueling Water Storage Tank (RWST), the ability to provide makeup to the RWST, and the ability to re-establish inventory in the Condensate Storage Tank (CST) or realignment of Auxiliary Feedwater (AFW) to its alternate source (cooling water system).

Please provide the additional information to support the review of the NFPA 805 licensing bases for maintaining the fuel safe and stable:

[This RAI includes Subparts a and b, as shown below along with NSPM responses]

NRC Request (SSA RAI 06.a):

a. The licensee stated that the PINGP thermal-hydraulic analysis was performed for a mission time of 24 hours to assure that safe and [stable] conditions can be achieved within that time period.

Provide a qualitative risk assessment for extending the mission time beyond 24 hours and implementing operator actions to establish makeup to the RWST, the CST and/or realignment of AFW to its alternate source.

NSPM Response (SSA RAI 06.a):

a. The qualitative risk assessment for actions to maintain safe and stable conditions is as follows:

The actions to refill the RWST, CST, and re-align AFW to the alternate cooling water source have a low risk associated with them. It is assumed that every fire causes a trip of both Unit 1 and 2, and a loss of Main Feedwater. It is likely that many fires would not cause a reactor trip and loss of Main Feedwater, which reduces the likelihood that these actions would need to be performed. The actions to makeup to the RWST, the CST, and realign AFW suction are well described in the existing plant procedures; there are multiple sources available; and the operators are familiar with these actions. After 24 hours, additional personnel will be available from the Emergency Plan, as well as additional resources (e.g., power, vehicles, equipment, etc.) to support performing these actions. Even if the CST were to be depleted, the AFW pumps can be re-aligned to take suction from the Mississippi River, providing a virtually unlimited supply of water. The AFW re-alignment can be performed from the control room, or locally by opening the cooling water supply valve without having to stop the AFW pump. There is CST level indication and indication for AFW suction pressure to provide diverse indication.

Because the Reactor Coolant Pump Seals have been replaced with seals that are not initially subject to excessive leakage, there is substantial time available to make up to the RWST. The single and multiple spurious combinations that could cause RWST or CST drain down events have been addressed by the thermal-hydraulic analysis described in EC 20736, "Reactivity Control," and EC 20738, "Decay Heat Removal" (References 6.56 and 6.57 in the LAR). Therefore the dependency on the RWST to make-up to the RCS is minimized. There are redundant RWST level indicators and low level annunciators in the control room.

Beyond time frames of 24 hours, existing plant procedures specific to the situation can be implemented to establish system alignments and to use equipment in ways differing from normal plant practice or training. Such procedures and equipment address such a wide spectrum of possibilities that it is not considered useful to develop all possible recovery contingencies. Additional guidance for options beyond those described in the plant AOPs and EOPs is provided by the PINGP document ASB, "Alternative Sources of Power, Water and Air Book," which may be used in formulating strategies in an event that has placed the plant in a condition that goes beyond the normal and emergency use of operating procedures and equipment.

Therefore the risk associated with these actions is low and PINGP can maintain safe and stable conditions indefinitely.

NRC Request (SSA RAI 06.b):

b. *The licensee also stated that operator actions are performed to align makeup/alternate water sources to the RWST beyond 38 hours, and to the CST or AFW pumps beyond 20 hours.*

Provide the details of the makeup sources and the process for aligning these sources, including a discussion of the feasibility analysis performed for these actions.

NSPM Response (SSA RAI 06.b):

b. It is noted that the preferred method to maintain the plant(s) in a shutdown condition with the required Keff level will be to use the Boric Acid Storage Tanks (BASTs) with the Reactor Makeup (RMU) Tanks to blend to the Volume Control Tank (VCT).

Methods and Sources by which Replenishment of the RWST May be Accomplished:
The RWST can be replenished from several different sources if and when required:

1. The RWST can be refilled from the RMU tanks and BASTs using the BA Blender per procedure 1C12.5, "Unit 1 Boron Concentration Control," or procedure 2C12.5, "Unit 2 Boron Concentration Control"
2. Water can be transferred from the other unit's RWST per procedure C16, "Spent Fuel Cooling System"
3. Water can be transferred from the CVCS Holdup tank(s) per procedure C12.8, "CVCS Holdup, Monitor, and Concentrates Holdup Tanks"
4. Water can be transferred from the CVCS Monitor tank(s) per procedure C12.8, "CVCS Holdup, Monitor, and Concentrates Holdup Tanks"

Methods and Sources by which Replenishment of the CST May be Accomplished:
The CST can be replenished from several different sources by unit, unit-to-unit, and by various methods if and when required using procedure 1C28.1 AOP2, "Loss of Condensate Supply to Aux Feedwater Pump Suction" or 2C28.1 AOP2, "Loss of Condensate Supply to Aux Feedwater Pump Suction" (note that the PINGP Unit 1 and Unit 2 CSTs are normally cross-tied and therefore, their collective volumes will be available to either unit without any additional action):

1. Water can be transferred from the Water Treatment System
2. Water can be transferred from the Condenser
   a. Using the Condensate Pump
   b. Using the Condenser Spray Pump
3. Water can be transferred from the ADT Monitor Tank (Note that this option is only available when the ADT Monitor Tank has sufficient level to support this activity)

Methods to cross-tie and align cooling water to the AFW System:
The AFW system can be cross-connected between units or aligned to the cooling water system if and when required:

1. The AFW Pump(s) can be aligned to take suction from the Cooling Water System per procedures 1C28.1 AOP2, "Loss of Condensate Supply to Aux Feedwater Pump Suction" or 2C28.1 AOP2, "Loss of Condensate Supply to Aux Feedwater Pump Suction"
2. The AFW systems can be cross-connected per procedures 1C28.1, "Auxiliary Feedwater System Unit 1" or 2C28.1, "Auxiliary Feedwater System Unit 2"

Existing Plant Procedures Credited for Accomplishing Replenishment of the RWST and CST:
Details of the lineups and how the lineups are accomplished are described in the following procedures (procedures have been placed on the Portal):

1. C16, "Spent Fuel Cooling System"
2. C12.8, "CVCS Holdup, Monitor, and Concentrates Holdup Tanks"
3. 1C12.5, "Unit 1 Boron Concentration Control"
4. 2C12.5, "Unit 2 Boron Concentration Control"
5. 1C28.1 AOP2, "Loss of Condensate Supply to Aux Feedwater Pump Suction"
6. 2C28.1 AOP2, "Loss of Condensate Supply to Aux Feedwater Pump Suction"
7. 1C28.1, "Auxiliary Feedwater System Unit 1"
8. 2C28.1, "Auxiliary Feedwater System Unit 2"

A formal feasibility analysis of the methods to replenish the RWST and CST (actions to maintain safe and stable conditions) is scheduled to be completed during the implementation phase; however, the methods for RWST and CST replenishment, as credited by the NFPA 805 analysis, are defined by existing operations procedures, for which Operations has received training. By allowing the use of many different methods and / or sources to accomplish replenishment, including replenishing from the unaffected unit and aligning to cooling water, there is a high confidence that a method will be available for use when and if required. Additionally, PINGP will reinstate and revise LAR Attachment S, Table S-3 Item # 53 as described in the response to SSA RAI 02.

SSA RAI 07 - Clarification of VFDRs for FA 13 and 18

Response to SSA RAI 07 is being provided in separate correspondence by June 26, 2015.

SSA RAI 08

NFPA 805, Section 4.2.2, requires that "For each fire area either a deterministic or performance-based approach shall be selected in accordance with Figure 4.2.2. The performance-based approach shall be permitted to utilize deterministic methods for simplifying assumptions within the fire area."

In Attachment C of the LAR, Table B-3, the licensee indicated that the regulatory basis for Fire Area 71 is the deterministic approach as described in NFPA 805, Section 4.2.3. However, in the table titled "Required Fire Protection Systems and Features", an Electrical Raceway Fire Barrier System (ERFBS) was identified as being required for risk. This implies that the ERFBS was evaluated using the performance-based approach using the Fire Risk Evaluation approach in accordance with NFPA 805, Section 4.2.4.2. If the Fire Area 71 regulatory basis is the deterministic approach, to maintain compliance with NFPA 805, Section 4.2.3, the ERFBS should meet the requirements of NFPA 805, Section 3.11.5, and be identified as being required for separation.

Please clarify if the ERFBS is credited to protect a nuclear safety performance function, and if Fire Area 71 was evaluated using deterministic or performance-based approach.

NSPM Response (SSA RAI 08):

The ERFBS is credited to protect one train of Pressurizer Level Indication (1L-433) and meets the requirements of NFPA 805 Section 3.11.5. Fire Area 71 meets Deterministic Requirements. The Fire PRA performed detailed analysis of Fire Area 71, but detailed PRA analysis of the area does not necessarily mean that the area needs to be Performance Based (Section 4.2.4.2).

LAR Attachment C should be revised as follows:

REQUIRED FIRE PROTECTION SYSTEMS AND FEATURES									
Fire Area	Category	ID	Type	Required?					Notes
				S	L	E	R	D	
71	Feature	See Note	ERFBS	Y ~~N~~	N	N	Y	N	Cable 2CF-74 has 3M Interam wrap

SSA RAI 09

NFPA 805, Section 2.4.3.3, states that when performing Fire Risk Evaluations, "The PSA approach, methods, and data shall be acceptable to the AHJ" (which is the NRC).

NFPA 805, Section 3.11.2, Fire Barriers, states that "Fire barriers required by Chapter 4 shall include a specific fire-resistance rating. Fire barriers shall be designed and installed to meet the specific 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."

In Attachment C of the LAR, the licensee stated that radiant energy shields are credited for risk in Fire Area 1 to protect raceway 1CV-T421 and in Fire Area 32 to protect raceway 1SG-LB22.

Please provide specific details of the nuclear safety functions that credit these radiant energy shields and discuss the extent of how the radiant energy shields are credited in the fire risk evaluations. In your discussion include how the fire resistance rating claimed in the risk analysis has been established through fire testing.

NSPM Response (SSA RAI 09):

No nuclear safety function credit is taken for radiant energy shields and they are not credited in fire risk evaluations. The Fire PRA did not credit the radiant energy shield to protect cables in cable trays 1CV-T42l in FA 1 or 1SG-LB22 in FA 32 from fire damage. FRE-FA01 and FRE-FA32 will be revised to indicate that the Radiant Energy Shield is not credited for Risk Reduction in FA 1 and FA 32. LAR Attachments A, K, and C should be changed as follows:

Attachment A, Section 3.11.2 should be revised to delete the discussion regarding FPEE-11-020, for credit is not taken for radiant energy shields.

Attachment K (page K-18) should be revised to delete the statement that "The existing radiant energy shield in Unit 1 has been determined to be acceptable in accordance with the NFPA 805 performance based approach."

Attachment C should be revised to indicate that the Radiant Energy Shield is not credited for Risk Reduction in FA 1 and FA 32 as follows:

REQUIRED FIRE PROTECTION SYSTEMS AND FEATURES									
Fire Area	Category	ID	Type	Required?					Notes
				S	L	E	R	D	
~~1~~	~~Feature~~	~~1CV-T42I~~	~~RES~~	~~N~~	~~N~~	~~N~~	~~Y~~	~~N~~	~~Raceway 1CV-T42 is wrapped with a radiant energy shield~~

REQUIRED FIRE PROTECTION SYSTEMS AND FEATURES									
Fire Area	Category	ID	Type	Required?					Notes
				S	L	E	R	D	
~~32~~	~~Feature~~		~~RES~~	~~N~~	~~N~~	~~N~~	~~Y~~	~~N~~	~~Cable 1SG-LB22 has a radiant energy shield (RES), Marinite board~~

SSA RAI 10

NFPA 805, Section 2.4.2.4, requires "An engineering analysis shall be performed in accordance with the requirements of Section [2.4] 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." RG 1.205, Revision 1, endorsed NEI 04-02, Revision 2, as one acceptable approach to performing and documenting the engineering analyses required to transition to a risk-informed, performance-based fire protection program in accordance with 10 CFR 50.48(c) and NFPA 805. On a fire area basis, NEI 04-02 requires that the licensee document how the nuclear safety performance criteria are met. The guidance in NEI 04-02 recommends that this information be presented in Table B-3, Fire Area Transition. In the LAR, Section 4.2.4, Overview of the Evaluation Process, Step 5 - Disposition, the licensee states that the final disposition of VFDRs should be documented in Attachment C (NEI [04]-02 Table B-3).

[This RAI includes Subparts a and b, as shown below along with NSPM responses]

NRC Request (SSA RAI 10.a):

a. *Attachment S of the LAR, Table S-2, Modification Items #34 and #35 involve protecting cables from fire damage in Fire Areas 32 and 58 to ensure electrical power availability to support the nuclear safety performance criteria (NSPC). However, Attachment C of the LAR does not describe the need for these modifications in the fire area assessment for Fire Areas 32 and 58.*

   Please discuss how these modifications support the appropriate NSPC and explain why these modifications are not identified in LAR Attachment C for Fire Area 32 and 58.

NSPM Response (SSA RAI 10.a):

a. Modification #6 will re-route 1C-333 out of FA 32 and FA 58 so the 1RY offsite power supply will be available in FA 32 and FA 58 allowing the vital auxiliaries NSPC to be met.

Modification #34 will protect cables 15402-G, 15402-K, and 1CA-1140 that could bypass the sync check switch or relay for the #1 emergency diesel generator output breaker from risk significant fire initiators allowing the vital auxiliaries NSPC to be met.

Modification #35 will protect cable 1C-332 from fire damage in fire area 32 and 58 to ensure BUS 16 can be re-powered from the 1RY transformer allowing the vital auxiliaries NSPC to be met.

LAR Attachment C, should be revised to include modification item #s 34 and 35 as well as modification #6 for fire areas 32 and 58. The disposition of VFDR-058-1-11 in LAR Attachment C should be revised to read as follows:

- No recovery actions are credited.
- Modifications identified in Table S-2, Item #s 6, 34, and 35 will ensure BUS-15 or BUS-16 will remain powered from either the 1RY or their respective diesel generator source.
- This VFDR has been evaluated and it was determined that the risk, safety margin and defense-in-depth meet the acceptance criteria of NFPA 805 Section 4.2.4 with plant modifications credited.

The disposition of VFDR-032-1-02 in LAR Attachment C should be revised to add the following:

- Modification identified in Table S-2, Item #34 will protect cables, from fire damage in Fire Area 32, associated with the D1 emergency diesel generator's output breaker so that the 1RY source will remain available to BUS-16.
- Modification identified in Table S-2, Item #35 will protect cable 1C-332 from fire damage in Fire Area 32 so that the 1RY source will remain available to BUS-16.

LAR Attachment S, Table S-2, Item #34 will be revised to include cables 15402-G, 15402-K, and 1CA-1140 in the Proposed Modification Column as follows:

Item	Rank	Unit	Problem Statement	Proposed Modification	In FPRA	Comp Measure	Risk Informed Characterization
34	Medium	1, 2	A fire in FA 13, 18, 32 or 58 could damage cables that control the Emergency Diesel Generator output breaker and bypass the sync check switch or relay which could cause a spurious closure of the breaker which could cause a lock-out of the 4 kV safeguards Bus which powers one train of safeguards equipment. The loss of the Bus is risk significant for some fire scenarios.	Protect cables 15402-G, 15402-K, and 1CA-1140 to prevent bypassing the sync check switch or relay from risk significant fire initiators.	Yes	No	This will reduce risk by making modifications to reduce the number of fire scenarios that could cause fire damage to a 4kV safeguards bus.

NRC Request (SSA RAI 10.b)

b. In Table S-2 of Attachment S of the LAR, the licensee describes several modifications that specify "protecting cables or circuits" (e.g., modification #6, #14, etc.).

Please describe the protection schemes that may be used for "protecting" cables.

NSPM Response (SSA RAI 10.b):

b. Cables will be protected in accordance with the requirements specified in NFPA 805 4.2.3.2, 4.2.3.3, or 4.2.3.4 as applicable or, will be re-routed outside the Zone of Influence (ZOI) of the Fire Initiator(s) of concern to achieve compliance with NFPA 805 Section 4.2.4.2 Performance-Based Approach.

Response to RAIs – Fire Modeling

FM RAI 01- Fire Modeling Tools and Methods

Response to FM RAI 01will be provided in separate correspondence by June 26, 2015.

FM RAI 02 – Fire Modeling Damage Criteria

Response to FM RAI 02 will be provided in separate correspondence by June 26, 2015.

FM RAI 03 – Fire Modeling V&V

Response to FM RAI 03 will be provided in separate correspondence by June 26, 2015.

FM RAI 04

NFPA 805, Section 2.7.3.3, states that acceptable engineering methods and numerical models shall only be used for applications to the extent these methods have been subject to verification and validation. These engineering methods shall only be applied within the scope, limitations, and assumptions prescribed for that method.

In Section 4.7.3 of the LAR, the licensee states that engineering methods and numerical models used in support of compliance with 10 CFR 50.48(c) are used and were used appropriately as required by Section 2.7.3.3 of NFPA 805.

Regarding the limitations of use:

[This RAI includes Subparts a, b and c, as shown below along with NSPM responses]

NRC Request (FM RAI 04.a):

a. *The NRC staff notes that algebraic models cannot be used outside the range of conditions covered by the experiments on which the model is based. NUREG-1805 includes a section on assumptions and limitations that provides guidance to the user in terms of proper and improper application for each FDT.*

   Please explain how it was ensured that algebraic models were not used outside their limits of applicability as described in NUREG-1805.

NSPM Response (FM RAI 04.a):

a. The application of the algebraic fire models in support of the PINGP Fire PRA was subjected to the Fire Modeling verification and validation process documented in NUREG-1824 Volume 1 and NUREG-1934. This verification and validation process indicates that the fire modeling results generated by the algebraic models were:

- Used within their limits of applicability as described in NUREG-1824 or
- In cases where the model is outside the applicability limits, sensitivity analyses have been developed to justify the use of the fire modeling results.

Consistent with the verification and validation process described in NUREG-1824 and NUREG-1934, the applicable dimensionless parameters have been determined for the algebraic fire models and compared with the applicability limits. This process in part ensures that the fire modeling analysis has been conducted with approved models, and the models have either been used within their validated applicability range or their results have been determined to be conservative based on appropriate sensitivity analyses.

As indicated in the text of the RAI, the algebraic models described in NUREG-1805 are qualified by a set of assumptions and limitations that should be considered when applying them. In the specific case of the PINGP Fire PRA, the four algebraic models used the most are:

- Heskestad's fire plume temperature correlation,
- Point source flame radiation model,
- Method of Foote, Pagni, and Alvares (FPA) room temperature correlation, and
- Method of McCaffrey, Quintiere, and Harkleroad (MQH) room temperature correlation.

The assumptions and limitations associated with these models as described in NUREG-1805 were explicitly considered by the fire modeling analysts when developing the analysis. Specific considerations include the following:

- Heskestad's Plume Temperature Correlation: The assumptions and limitations are described in section 9.5 of NUREG-1805. The PINGP analysis is consistent with these assumptions because:
  - It is assumed that energy is released at one point in space (i.e., the point where the fire is located),
  - The plume temperature is used to determine fire hazards in the early stages of the fire (the generation of the initial zone of influence and/or the determination of the critical heat release rate for generating damage to the first target) before the hot gas layer may have an effect, and
  - It is only used for diffusion flame applications for ignition sources such as electrical cabinets as opposed to jet or pre-mixed flames.

- Point Source Flame Radiation Model: The assumptions and limitations are described in Section 5.5 of NUREG-1805. The assumption and limitations indicate that the model offers conservative estimates of the fire hazard (i.e., thermal radiation). These are general statements that are addressed specifically in the PINGP analysis through the validation process based on the scenario configuration and input parameters. An additional assumption indicates that the base of the fire is circular. Consistent with this analysis, the base of the fire in the PINGP analysis is assumed to be circular with an equivalent area of the ignition source as applicable.

- MQH and FPA Room Temperature Correlations: The assumptions and limitations are listed in Section 2.10 of NUREG-1805.

- The assumption associated with the application in conventional size compartments and its corresponding shapes are addressed through the validation process in the evaluation of the compartment aspect ratio.
- The correlation is also used for steady state and transient heat release rate profiles consistent with the listed assumptions.
- The resulting hot gas layer temperatures are treated as "global" indications of room temperatures. The results are not interpreted as localized to a specific region of the compartment.
- The correlations are used for determining if the hot gas layer temperatures reach target damage temperatures, which is a temperature range within the model capabilities and the range of validation in NUREG-1824.
- Wall and corner fire location factors are utilized when applicable when determining the hot gas layer temperatures.
- The FPA and MQH models do not require a single heat loss fraction value. Therefore, this parameter is not used.

NRC Request FM RAI 04.b):

b. *It is stated on page J-8 in Table J-1 of Attachment J of the LAR that the V&V of Alpert's ceiling jet temperature correlation is documented in NUREG-1824. It is also stated that the V&V demonstrates that the ceiling jet correlation is implemented correctly and in all cases provides conservative bounding estimates.*

   Please provide technical justification for the second statement, because the fact that a model receives V&V does not prevent its use outside the model's limitations.

NSPM Response (FM RAI 04.b):

b. The ceiling jet correlation is not used in the PINGP Fire PRA. Consequently, no verification and validation is necessary for this correlation and it does not need to be listed in Attachment J to the LAR. Additionally, the values for the horizontal component of the zone of influence (ZOI) are at least bounded by the point source radiation model (which is a validated model in NUREG-1824) per guidance in Chapter 8 and Appendix F of NUREG/CR-6850.

NRC Request (FM RAI 04.c):

c. *It is stated on page J-9 in Table J-1 of Attachment J of the LAR that "[FDS] is used within the limits of its range of applicability as documented in FPRA-PI-MCR." It is further stated that "For relevant scenarios where the input parameters are outside of the limits, control room abandonment conditions are still predicted." The two statements appear to be contradictory.*

   Please provide technical justification for the application of FDS with input parameters outside the allowable range, and for using the corresponding calculated abandonment times in the Fire PRA.

NSPM Response (FM RAI 04c):

c. The statement on page J-9 in Table J-1 of Attachment J of the LAR that "[FDS] is used within the limits of its range of applicability as documented in FPRA-PI-MCR" is incorrect. This response will provide the justification for the model inputs.

   To demonstrate that the FDS model is used within the limits of its range of applicability, the non-dimensional parameters associated with the fire strength, the compartment geometry, the ventilation, and the flame heights were compared against the applicable ranges provided in Table 2-5 of NUREG-1824, Volume 1. The non-dimensional parameters were calculated for the complete range of fire size bins, from the smallest fires (bin 1) to the largest fires (bin 15). The results, listed in FPRA-PI-MCR, Appendix I, are described below:

   - The fire Froude Number falls within the NUREG-1824 Validation & Verification (V&V) range for most of the electrical cabinet fires within the corresponding probability distribution for peak heat release rates and only fire size bins 14 and 15 for the transient fires.
     - Electrical Cabinet Fires: The fire size bins that fall outside of the validation range are bins 1 and 12-15. The higher heat release rate bins predict abandonment during the growth period of the ignition source and the heat release rate at the abandonment time produces a Froude Number within the validation range. The low heat release bin falls below the valid range because the modeling parameters select a fixed fuel area that artificially decreases the Froude Number. In this configuration, the fire plume will entrain more air, resulting in a more rapid smoke layer descent, and a corresponding faster reduction in visibility. Low Fire Froude Number cases are conservative due to the rapid reduction in visibility when compared to an equivalent case that falls within the validation space.
     - Transient Fires: Only fire size bins 14 and 15 are within the validation range. The low heat release bins fall below the valid range because the modeling parameter selected uses a fixed fuel area that artificially decreases the Froude Number. In this configuration, the fire plume will entrain more air, resulting in a more rapid smoke layer descent, and a corresponding shorter abandonment time. Low Fire Froude Number cases are conservative due to the rapid reduction in visibility when compared to an equivalent case that falls within the validation space.

     Also, the selected soot yield of the fire is conservatively chosen to produce more soot than reported for similar cable materials in the SFPE Handbook and NUREG/CR-7010. This conservative parameter selection is chosen to conservatively bias the model prediction of abandonment time sufficiently to counteract the out of range Froude Number.

   - The flame height ratio falls within the NUREG-1824 V&V range for most of the scenarios.
     - Electrical Cabinet Fires: Bin 1 falls below the valid range, while bins 10 through 15 are above the valid range.
     - Transient Fires: Bins 1 through 6 fall below the valid range, and no cases are above the valid range.

The flame height validation parameter is used primarily to evaluate the exposure of targets within the fire plume [NUREG-1824]. Control room abandonment evaluations assume inherently that occupants will be able to perform their duties outside of the fire plume zone of influence. Therefore, the flame height ratio is only important insofar as it may impact the hot gas layer temperature and visibility. Fires in which the flame impinges upon the ceiling will produce higher hot gas layer temperatures, and therefore force abandonment sooner than an equivalent case that falls within the validation range. Furthermore, fires in which the room is substantially taller than the flame height will allow greater entrainment into the plume, resulting in more rapid descent of the smoke layer. Also, the selected soot yield of the fire is conservatively chosen to produce more soot than reported for similar cable materials in the SFPE Handbook and NUREG/CR-7010. This conservative parameter selection is chosen to conservatively bias the model prediction of abandonment time sufficiently to counteract the out of range flame height ratio.

- The equivalence ratio falls within the NUREG-1824 V&V range for most of the scenarios.
  - Electrical Cabinet Fires: Bin 1 falls below the valid range and no cases are above the valid range.
  - Transient Fires: Bins 1 through 4 fall below the valid range and no cases are above the valid range.

  This parameter does not negatively impact the analysis because the parameter is simply showing that there is sufficient oxygen available for the fire to burn within the MCR volume.

- The enclosure ratio falls within the NUREG-1824 V&V range for most of the scenarios.
  - The enclosure ratios are within the NUREG-1824 V&V range for the electrical cabinets outside of the horseshoe. The enclosure ratio for the electrical cabinets inside of the horseshoe exceeds the NUREG-1824 V&V range. The electrical cabinet dimensions that are outside of the valid range are due to the extreme elevation of the fuel source, relative to the ceiling height. As discussed for the Flame Height Ratio parameter above, the effect of an elevated fuel source is considered to produce consistent or conservative results when compared to an equivalent case within the valid range.
  - The enclosure ratios are within the NUREG-1824 V&V range for all transient fires.

In addition to the conservative fire modeling parameter selections described above, the probability of abandonment calculation identifies the most conservative time to abandonment for all fires in the FDS simulations. The shortest time to abandonment is used for all fire scenarios. This is a bounding conservative treatment of the time to abandonment and it will bound the risk associated with any of the above stated V&V parameters that were identified as out of range.

FM RAI 05

NFPA 805, Section 2.7.3.4 states that personnel who use and apply engineering analysis and numerical methods shall be competent in that field and experienced in the application of these methods as they relate to nuclear power plants, nuclear power plant fire protection, and power plant operations.

In Section 4.5.1.2 of the LAR, the licensee states that fire modeling was performed as part of the Fire PRA development (NFPA 805, Section 4.2.4.2). The NRC staff notes that this requires that qualified fire modeling and PRA personnel work together. Furthermore, in Section 4.7.3 of the LAR, the licensee states that "For personnel performing fire modeling or Fire PRA development and evaluation, NSPM 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."

Regarding qualifications of users of engineering analyses and numerical models (i.e., fire modeling techniques):

[This RAI includes Subparts a, b and c, as shown below along with NSPM responses]

NRC Request (FM RAI 05.a):

a. *Please describe the requirements to qualify personnel for performing fire modeling calculations in the NFPA 805 transition.*

NSPM Response (FM RAI 05.a):

a. Fire Modeling performed to support the development of the NFPA 805 LAR was completed by qualified contractors. The vendor provided credentials for the fire modeling analysts. Credentials were reviewed to ensure analysts were appropriately qualified per the requirements of NFPA 805, Section 2.7.3.4 which included education in fire protection engineering and fire modeling, and extensive experience performing fire modeling studies. NSPM reviewed the vendor's credentials of the analysts performing the fire modeling tasks and ensured that each task was performed by analysts with appropriate training in the fire modeling area being performed. During the transition process post NFPA 805 LAR submittal, NSPM will continue to require credentials to ensure analysts are knowledgeable in fire modeling techniques, including interpreting and maintaining fire modeling software.

NRC Request (FM RAI 05.b):

b. *Please describe the process for ensuring that fire modeling personnel have the appropriate qualifications, not only before the transition but also during and following the transition.*

NSPM Response (FM RAI 05.b):

b. Fire Modeling performed for supporting the development of the NFPA 805 LAR was completed by qualified contractors. The vendor provided the fire modelers' credentials. Credentials included education in fire protection engineering and fire modeling, and extensive experience performing fire modeling studies. NSPM reviewed the vendor's credentials of the analysts performing the fire modeling tasks and ensured that each task was performed by analysts with appropriate training in the fire modeling area being performed. During the transition process post NFPA 805 LAR submittal, NSPM will continue to require credentials to ensure analysts are knowledgeable in fire modeling techniques, including interpreting and maintaining fire modeling software.

   Once the NFPA 805 transition process is completed, fire modeling calculations will be performed by a Fire Protection or PRA Engineer who meets the qualification requirements of Section 2.7.3.4 of NFPA 805. This will be ensured through qualification requirements and training that will be developed as described in Table S-3, Implementation Item #26. An analyst will be required to complete this qualification before modeling in support of the Fire PRA or a qualified person will need to review and sign off the prepared material before its use within the Fire PRA. Qualifications are tracked through NSPM's training program and are procedurally required to be checked prior to completing the task that requires fire modeling. The NSPM fire modeling qualification requires an analyst to be qualified in PRA before that analyst can model fires for their input into the Fire PRA.

NRC Request (FM RAI 05.c):

c. *When fire modeling is performed in support of the Fire PRA, please describe how proper communication between the fire modeling and Fire PRA personnel is ensured.*

NSPM Response (FM RAI 05.c):

c. During the development phase of the Fire PRA, Fire Protection Engineers (FPE) who conducted the fire modeling and the PRA engineers maintained frequent communications. Specifically, the fire modeling personnel populated the databases or spreadsheets in which all the relevant fire modeling inputs are maintained. The fire scenario frequencies generated by these tools are electronically transmitted to the PRA engineers who perform the risk quantification. Both the FPEs and the PRA engineers participated in the cutset review meetings during the development of the Fire PRA.

   The NSPM qualification for Fire Modeling precludes the use of Fire Modeling in the Fire PRA unless it is reviewed by a qualified Fire PRA Engineer. This ensures communication between Fire Modeling personnel and Fire PRA personnel.

FM RAI 06

NFPA 805, Section 2.7.3.5 states that an uncertainty analysis shall be performed to provide reasonable assurance that the performance criteria have been met.

In Section 4.7.3 of the LAR, the licensee states that "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."

Regarding the uncertainty analysis for fire modeling:

[This RAI includes Subparts a and b, as shown below along with NSPM responses]

NRC Request (FM RAI 06.a):

a. *Please describe how the uncertainty associated with the fire model input parameters was accounted for in the fire modeling analyses.*

NSPM Response (FM RAI 06.a):

a. The uncertainty associated with the fire model input parameters was accounted for by using conservative input parameters and varying input parameters in sensitivity cases, as described below:

    - The input parameter that has the most impact on results is the heat release rate. The uncertainty is accounted for by using bounding heat release rates. Heat release rates are selected to be the screening values (98th percentiles) of the distributions recommended in NUREG/CR-6850. For the case of cable fires as secondary combustibles, the guidance in NUREG/CR-7010 associated with modeling cable fires (i.e., the heat release rate per unit area recommended for the FLASH-CAT model) was utilized.
    - The analysis conservatively assigns the lowest radiant heat flux damage threshold and damage temperature threshold, 6 kW/m^2 and 205 °C, suggested in Appendix H of NUREG/CR-6850. The use of these damage criteria bounds the fire impacts expected for raceways containing a mixture of cables with varying insulation types.
    - Sensitivity cases are performed when normalized parameters are outside of the validation range. Sensitivity evaluations were run for room aspect ratio, vent sizes, and fire height versus ceiling height.

NRC Request (FM RAI 06.b):

b. *Please describe how the "model" and "completeness" uncertainty was accounted for in the fire modeling analyses.*

NSPM Response (FM RAI 06.b):

b. The fire modeling verification and validation analysis was completed as part of the development of the Fire PRA. This analysis covers all the fire modeling, including CFAST and hand calculations, used in the PINGP Fire PRA. The verification and

validation analysis determines whether models are used within their validated range. If the models are found to be used outside of the range, then the input parameters are varied in a conservative direction (i.e., more challenging fire conditions) and the revised model results are used as input to the Fire PRA; i.e., the PINGP fire modeling analysis results are applied considering the uncertainty associated with the model validation range. It should be noted that for fire models used outside the range of applicability, sensitivity cases are run, as suggested in NUREG-1934.

The uncertainty sources and treatments associated with fire scenario development and detailed fire modeling for single compartment fire scenarios in the PINGP Fire PRA include uncertainties related to the selection of transient zones, fire location, fire growth and propagation, activation and function of the detection and suppression system, the selection of damage criterion, conduit routing, selection of fire models, and the inputs to the chosen fire models. A summary of the uncertainty sources and treatments associated with the fire scenario development and detailed fire modeling for single compartment fire scenarios in the PINGP Fire PRA is provided next:

1. Uncertainty Associated with the Scenario Development Process
   a. Selection of Transient Zones: Transient zones are postulated so that most of the open floor area is captured in the analysis. These are areas where a transient fire could be postulated. There is however uncertainty in the selected sizes and boundaries of the transient zones. As a conservative practice, and when practically possible, the transient zones have been selected large enough to capture target damage beyond a typical zone of influence of an ignition source. In addition, a target overlap between transient zones has been incorporated in the analysis to account for targets near the boundaries. That is, the targets located nearby the boundaries between transient zones have been mapped to all transient zones adjacent to that boundary. When applicable, fire propagation between transient zones is included in the scenario progression.
   b. Fire Location (All fire zones except the Main Control Room): The location of the fire is unavoidably a source of uncertainty as all fires in the Fire PRA are assumed to occur at a specific point within the fire zone. This fire location impacts the heating process of nearby targets. This source of uncertainty is addressed in the Fire PRA in a consistent approach. That is, the guidance for assigning fire location has been consistently applied throughout the analysis. The guidance is based on a conservative practice of assigning fixed and transient fire locations consistently for all scenarios. For fixed sources, the fires have been postulated at the elevation of the ignition source. In the specific case of electrical panels, the fires are postulated 1' below the top of the cabinet as clarified in NUREG/CR-6850 Supplement 1 (Chapter 12). In the case of general transients or transient fires due to hotwork, the base of the fire has been postulated 2' from the floor. Since the transient fires are due to items brought temporarily into the fire zone, there is uncertainty associated with the fire elevation. The 2' elevation is a necessary practical assumption to account for combustibles that may not be located at floor level and representative of typical equipment carts, for example, that are brought into fire zones. Oil spill fires are postulated at floor level.
   c. Fire Growth and Propagation: The uncertainties associated with heat release rates, fire growth profiles, fire propagation and determination of time to target damage are documented in a number of industry documents and are not practical to list in this report. For example, NUREG-1805, NUREG-1824, and NUREG/CR-6850 list and in some cases quantify these uncertainties. For the

PINGP Fire PRA, generic industry guidance on fire growth rates has been utilized. For fire propagation, the Zone of Influence used to map targets to each ignition source conservatively bounds the damage produced by a fire reaching the 98th percentile HRR for that source. As such, damage to cable trays and conduits as the result of a fire propagating through secondary combustibles is accounted for.

2. Uncertainties Associated with Detection and Suppression
   a. Activation: The activation times for detection and suppression systems is a source of uncertainty in the Fire PRA primarily because of the complexity of the configurations encountered in the scenarios postulated in the different zones in the power plant. Examples of these complex configurations include devices located in heavily obstructed ceilings, sprinklers within cable trays, etc. As a conservative practice, detection and suppression systems are modeled such that applicable targets are damaged prior to activation.
   b. Suppression: The ability to control or completely suppress the fire as a function of time using the different suppression means available in a given scenario is a source of uncertainty. In the Fire PRA, detection and suppression is treated with an event tree approach where both outcomes, successful and failed suppression activities, are modeled. In addition, suppression times are conservatively assessed based on selected input parameters to the fire models and the use of validated models.
3. Damage Criterion

   The damage criterion is a source of uncertainty that is considered in the analysis. The generic guidance for damage criteria suggests point estimates for damage thresholds, which are used in this study. These generic point values are based on experimental observations yielding a range of values associated with cable damage. Such uncertainty is captured in a point value that for the most part is expected to be bounding (see Appendix H of NUREG/CR-6850 and Appendix A of NUREG-1805). For the PINGP Fire PRA, damage thresholds associated with Thermoplastic cables are assumed throughout the analysis.
4. Conduit Routing

   Conduit routing is a source of uncertainty in the Fire PRA. The conduit locations are partially available on drawings and are also partially labeled in the field. Consequently, routing of conduits has been evaluated on an individual basis based on conduit location and risk significance. Conduits which could not be located in the plant were failed for every applicable scenario.
5. Fire Modeling Selection

   For the PINGP Fire PRA, CFAST and hand calculations have been utilized. These models have been subjected to verification and validation studies for selected scenarios as described in NUREG-1824, where model uncertainty ranges have been documented.
6. Fire Modeling Inputs

   The uncertainty intervals associated with the input parameters include:
   a. Heat Release Rates

i. Fixed Ignition Sources: The gamma distributions of the heat release rate values for most fixed ignition source types are given in Appendix E of NUREG/CR-6850.

ii. Oil Fires: In the case of large oil fires, a two-step approach was used: Step 1) Large oil fires were assumed to cause a hot gas layer without a Severity Factor (SF) = 1.0. Step 2). If this approach was too conservative, then the amount of oil contained within the pump was determined and the approach documented. The uncertainty associated with the heat release rate intensity and duration of oil fires is mostly governed by the postulated size of the spill and the amount of oil. The FPRA assumes bounding heat release rates for oil spills to account for the uncertainty associated with the amount of oil in the ignition source (i.e., pumps). At the same time, the frequency of oil fires is calculated using two probabilistic parameters: 1) The probability that the ignition source fire is associated with oil (split fraction from Table 6-1 of NUREG/CR-6850 and 2) The Severity factor for small and large oil fires described in Appendix E of NUREG/CR-6850. These inputs are treated as uncertain parameters in the risk equation.

b. Cable fire modeling parameters:

i. Tray width is assumed to be 18" for all heat release rate contribution calculations.

ii. Tray heat release rate per unit width is taken from Table R-1 of NUREG/CR-6850 in comparison to the value provided in Section 9.2.2 of NUREG/CR-7010.

iii. The heat of combustion per unit weight of cable insulation was selected to be 16 MJ/kg for cables in general as recommended in Section 9.2.2 of NUREG/CR-7010.

iv. The number of cables per tray was assumed such that the calculated heat release rate would bound the typical arrangement found at PINGP.

c. Other Fire Modeling Inputs: The other fire modeling inputs include compartment geometry and ventilation characteristics. The compartment geometry and ventilation characteristics are obtained from plant drawings.

In summary, the fire modeling analysis in the Fire PRA has been conducted consistent with the industry guidance and practices including fire modeling verification and validation. Consistent with the guidance, uncertainties associated with the fire modeling parameters are reflected in the risk quantification as follows:

- Severity factor, which is calculated using a critical heat release rate value as described earlier in this response. In addition to the conservative determination of critical heat release rate values, the uncertainty associated with the severity factor is explicitly modeled in the uncertainty task of the Fire PRA.
- Non suppression probabilities, which are calculated using the time to damage resulting from fire modeling analysis and the conditional core damage probability/conditional large early release probability. In addition to the conservative determination of non-suppression probability values, the uncertainty associated with the severity factor is explicitly modeled in the uncertainty task of the Fire PRA.

- Conditional core damage probability/conditional large early release probability, which is calculated based on the targets associated with each fire scenario. As discussed earlier in this response, mapping of targets to the different fire scenarios follows a conservative process to ensure that the resulting probabilities are bounding.

Completeness associated with fire models is addressed in the PINGP Fire PRA within the overall quantification process, as the PRA is an integrated analysis. Fire Modeling provides inputs to a broad comprehensive Fire PRA which includes modeling of electrical systems, operator actions, and the plant systems and components needed to shutdown the plant. One of the first steps in the fire modeling process is to identify the fire scenarios that will be analyzed. In some situations, the scenario analysis invokes fire modeling capabilities that are not currently available, generating the completeness uncertainty situation described in the question. When the fire modeling does not provide a full answer or an answer with sufficient resolution, the scenario definition and target mapping within the Fire PRA conservatively compensates for the lack of information. The Fire PRA allows the analyst to conservatively compensate for the lack of fire modeling capabilities outside the fire modeling analysis so that the scenario is properly modeled in the Fire PRA. Some examples are listed below:

- The determination of time to automatic suppression. Available detection models are not fully applicable to many of the postulated scenarios; therefore, as part of the scenario definition, targets are failed intentionally before the automatic suppression is credited.

- Oil spill fires are difficult to analyze; therefore, full fire zones or transient zones have been assumed failed due to oil fires.

- Both zones of multi-compartment combinations are failed conservatively when fire modeling propagation calculations from one compartment to another are not conducted.

- Full main control board panels are failed due to the lack of analytical fire modeling methods, with appropriate verification and validation studies, to predict flame propagation within a panel.

The four examples above illustrate how the completeness uncertainty associated with fire modeling calculations is addressed "outside of the fire modeling" by conservatively failing targets in the fire scenarios so that the risk contribution is bounding.

RAI Responses – Radioactive Release

Radioactive Release RAI 01

NFPA 805, Section 1.5.2 states that "Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, limits."

Attachment E of the LAR does not identify use of outside yard areas to store radioactive materials.

Please discuss if there are any outside yard areas where radioactive materials are stored (e.g. in sealand type containers.) Since outside yard areas are open to the atmosphere, if radioactive material is stored outside, provide an analysis for a fire occurring in outside yard areas that demonstrates that the gaseous and liquid effluent releases will result in doses that are less than the 10 CFR 20 annual dose limits to a member of the public.

NSPM Response (RR RAI 01):

Although not identified in Attachment E of the LAR, radioactive materials are stored in an outside yard area and NSPM has performed an analysis that demonstrates that the gaseous and liquid effluent releases due to a fire and firefighting activities will result in doses that are less than the 10 CFR 20 annual dose limits to a member of the public.

There is a fenced off area west of the switchyard that contains thirteen (13) 20 foot Sealand containers and eight (8) other Low Level Radioactive Waste (LLRW) containers. There is a total activity of approximately 20 milli-curies total in all of the containers. (This is a typical inventory and activity level of material stored in this area.)

In addition, the spare Reactor Coolant Pump (RCP) motor that was previously located on the Turbine Operating Deck in Fire Area 8 has been relocated to the newly constructed Distribution Center Warehouse in the Owner Controlled Area (OCA). The spare RCP motor activity is 5.7E-1 milli-curies.

Technical Basis Document # 12-002, Dose Due to a DAW Trailer Fire (Reference R-1), is a calculation used to verify that offsite dose and site boundary liquid radionuclide concentrations from effluents due to a trailer fire at the plant site are below Technical Specification limits. This DAW trailer fire analysis also bounds releases due to a fire in the Distribution Center Warehouse. The methodology and results of the calculation are summarized below.

<u>Methodology</u>

The calculation is based on a fire scenario involving a 20 foot Sealand van of dry active waste (DAW). The total activity in the Sealand van of DAW is approximately 200 milli-curies, based on the highest activity DAW shipment in the prior five (5) years. This bounds the total activity of 20 milli-curies in the fenced off area west of the switchyard and the total activity of 5.7E-1 milli-curies for the spare RCP motor in the Distribution Center Warehouse.

Water used in firefighting activities and hose stream application will transport the total activity into the plant discharge canal and then into the Mississippi River. A member of the public is assumed to drink two liters at a point downstream of the plant discharge. The analysis assumes

that activity from the fire is diluted by the discharge canal flow rate and the Mississippi River low flow rate; no credit is taken for dilution from water used in firefighting or hose stream activities. Even though there is no water intake for 300 miles downriver, dilution by the Mississippi flow at the discharge point is a valid assumption and drinking the water just after this dilution occurs is conservative.

Gaseous effluent from this fire includes the total trailer radioactivity that becomes airborne over six hours. This airborne cloud is diluted using the X/Q for the Design Basis Accident exclusion area boundary and a member of the public is exposed to the cloud for six hours using the standard man breathing rate.

For both the liquid and gaseous releases, the dose conversion factors were the most limiting from US EPA Federal Guidance Report No. 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion.

Results

The total activity analyzed for the fire scenario involving a Sealand van of DAW is approximately 200 milli-curies. The total activity is 20 milli-curies in the fenced off area west of the switchyard inclusive of all Sealand and other LLRW containers. As such, the DAW trailer fire addressed in the Technical Basis Document bounds any single or combination of Sealand containers and boxes in the area. The total analyzed activity in the Sealand van of DAW also bounds the total activity of 5.7E-1 milli-curies for the spare RCP motor in the Distribution Center Warehouse.

The PINGP Offsite Dose Calculation Manual (ODCM, Reference R-2) adopted the limits of 10 CFR 50 Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents. As such, the ODCM and 10 CFR 50 Appendix I limits are the same. The ODCM and 10 CFR 50 Appendix I limits are more restrictive than the 10 CFR 20 limits.

The liquid release dose of 0.0065 mrem whole body does not exceed the ODCM and 10 CFR 50 Appendix I limit of 3 mrem/quarter (two unit site).

The gaseous release dose of 0.45 mrem whole body and 5.7 mrem lung dose (maximum organ) does not exceed the ODCM and 10 CFR 50 Appendix I limit of 15 mrem/quarter (two unit site).

Based on the results of the Technical Basis Document calculation, the DAW trailer fire scenario does not exceed ODCM, 10 CFR 50 Appendix I, and 10 CFR 20 offsite dose limits.

References:

R-1 PINGP Technical Basis Document 12-002, Dose Due to a DAW Trailer Fire, Revision 0, November 23, 2012.

R-2 PINGP Offsite Dose Calculation Manual (ODCM).

Radioactive Release RAI 02

NFPA 805, Section 1.5.2 states that "Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, limits."

Attachment E of the LAR identified Fire Area 4, the Fuel Handling Area, and Fire Area 61A, the Auxiliary Building Hatch Area, as not having ventilation where the potential transfer of contaminated smoke to the exterior can occur. Attachment E indicates that revised fire strategies will incorporate mitigating actions to monitor and filter potentially contaminated smoke based on radiological conditions identified during the conduct of firefighting activities.

Please provide information on how this will be accomplished.

NSPM Response (RR RAI 02):

For fire areas without ventilation controls such as Fire Areas 4 and 61A, NSPM will implement actions to monitor and filter potentially contaminated smoke, as stated in Attachment E of the LAR. Based on radiological conditions, these actions may include opening doors to adjacent areas with filtered ventilation (see LAR Table S-3 Item 13), or use of portable HEPA ventilation units, as will be addressed in new Table S-3 Item 65, as described below. These actions are further described below.

Fire Areas With No Ventilation Controls

Fire Area 4 runs from the 695 ft elevation to the roof of the 755 ft elevation. Fire Area 4 is open to Fire Area 61A on the 755 ft elevation by the open space over the top of Fire Area 61, the Spent Fuel Pool Area, which is a part-height enclosure on the 755 ft elevation.

During the development of Attachment E for the LAR supplement, a modification was implemented that abandoned the fan and failed closed the damper for the Laundry Fan Exhaust Filter in Fire Area 61, the Auxiliary Building Anti-C Clothing Area on the 735 ft elevation. The modification removed the ability to locally operate the Laundry Room Filters and Exhaust Fans to prevent gaseous effluent from escaping Fire Area 61. Fire Area 61 is not provided with any other form of ventilation controls and, as such, is addressed in this response as a non-ventilated area along with Fire Area 4 and Fire Area 61A.

Fire Area 4 and Fire Area 61A are part of the Common Area of the Auxiliary Building (CAAB) that is not provided with ventilation controls. Fire Area 61 is located directly below Fire Area 61A and an open stairway connects the two fire areas. Fire Area 61 is also part of the CAAB that is not provided with ventilation controls. Credit is taken for the large volume of the CAAB to contain radioactive gaseous effluent from the CAAB. Mitigating actions will be based on radiological conditions as monitored by radiation protection and communicated to the fire brigade leader during fire events.

Mitigating Actions

Mitigating actions could include opening the doors to Fire Area 62, Spent Fuel Pool Area, to direct potentially contaminated smoke from a fire in a CAAB fire area to the Spent Fuel Pool normal ventilation system. The Spent Fuel Pool Normal Ventilation System filters air through roughing and HEPA filters to the spent fuel pool normal ventilation exhaust stack prior to

release. The revised fire strategies and associated training materials described in Implementation Items 6 through 13 in LAR Table S-3 will provide procedural guidance for response to such events.

Mitigating actions could also include the use of portable HEPA filters for use in any fire area having the potential for generation of radiological gaseous effluents, not just in the identified areas without ventilation controls. Portable HEPA filters will be strategically located in the Radiologically Controlled Area (RCA) and will be available for use based on radiological conditions as monitored by radiation protection personnel and communicated to the fire brigade leader during fire events. Table S-3, Item 65, will be added to provide these portable HEPA filters as follows:

> Item 65:
> Provide portable HEPA filters strategically located in the Radiologically Controlled Area (RCA) that will be available for use based on radiological conditions as monitored by radiation protection personnel and as communicated to the fire brigade leader during fire events.

Radioactive Release RAI 03

NFPA 805, Section 1.5.2 states that "Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, limits."

Attachment E of the LAR identified Fire Area 93, the Low Level Rad Waste Area, as not having ventilation where the potential transfer of contaminated smoke to the exterior can occur. Attachment E states that a combination of containerization and administrative controls will limit the amount of exposed contaminated combustible materials, and that revised fire strategies will incorporate mitigating actions to filter potentially contaminated smoke.

Please describe what administrative controls (e.g. limits on activity, etc.) and types of containers will be used in this area that will be used to meet the applicable 10 CFR 20 requirements?

Furthermore, Table S-3, Implementation Item 15, provides a container with booms, portable filtered ventilation and other appropriate equipment to contain effluent releases in the Low Level Rad Waste Area.

Please clarify if this container [will] be staged in this building to be used in case of a fire?

NSPM Response (RR RAI 03):

<u>Background</u>

Fire Area 93, the Low Level Rad Waste Area, is used for storage of plant equipment, radioactive trash, spent resin, and spent filters. The old upper reactor internals from units 1 and 2 are stored in this building. There is no processing of waste in the building. The main open area of Fire Area 93 does not contain externally contaminated containers. The old upper reactor internals are stored behind a part-height concrete shield wall along the south wall of the building and are enclosed in a carbon steel container. The main open area does not present radiological

release concerns. The truck loading area in the southeast corner of the building is used to stage potentially contaminated waste and personal protective clothing (radioactive trash) prior to loading the material into a Sealand van. The staging area is designated by ropes and is smaller in size than a 20 foot Sealand van. Administrative controls and containerization described below apply only to this location in the truck loading area.

Administrative Controls and Types of Containers

Administrative controls and containerization will be applied to limit the exposed storage of potentially contaminated waste and personal protective clothing resulting primarily during outages. A portion of the truck loading area is typically roped off and used to store plastic bags of exposed potentially contaminated waste and personal protective clothing prior to loading of the material into a Sealand van. Administrative controls identified in Table S-3 Item 16 will be implemented to minimize the amount of such exposed materials by requiring that the plastic bags be placed in metal containers prior to being loaded instead of being left exposed in the truck loading area. The size of the metal containers will be dependent on the amount of expected materials, taking into account the need for accessing the containers for loading of the materials.

Technical Basis Document # 12-002 Rev. 0, Dose Due to a DAW Trailer Fire, dated 11/23/2012 is used to verify that offsite dose and site boundary liquid radionuclide concentrations from effluents due to a trailer fire at the plant site are below Technical Specification limits. The calculation is based on a fire scenario involving a 20 foot Sealand van of dry active waste (DAW) based on the highest activity DAW shipment in the prior five (5) years. Based on the results of the Technical Basis Document calculation, the DAW trailer fire scenario does not exceed ODCM offsite dose limits. Refer to the response to Radiological Release RAI 01 for details associated with the Technical Basis Document calculation.

The DAW trailer fire scenario in the Technical Basis Document bounds the maximum size of metal containers that will be used to contain outage-related plastic bags of potentially contaminated waste and personal protective clothing. The DAW trailer fire scenario also bounds the maximum activity of DAW in the metal containers that will be used to contain outage-related plastic bags of potentially contaminated waste and personal protective clothing.

The container described in Table S-3, Implementation Item 15, with booms, portable filtered ventilation and other appropriate equipment to contain effluent releases in the Low Level Rad Waste Area, will be stored in a location adjacent to, but not within, the Low Level Rad Waste Area. The specific location will be determined as part of implementation, and will be in the normal travel path to the Low Level Rad Waste Area.

RAI Responses – Probabilistic Risk Assessment (PRA)

PRA RAI 01 – Fire Event Facts and Observations

Section 2.4.3.3 of NFPA 805 states that the probabilistic safety assessment (PSA) (PSA is also referred to as PRA) approach, methods, and data shall be acceptable to the authority having jurisdiction (AHJ), which is the NRC. The RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a fire PRA (FPRA) and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the NRC staff for adopting a fire protection program consistent with NFPA 805. The RG 1.200 describes a peer review process utilizing an associated ASME/ANS standard (currently ASME/ANS-RA-Sa-2009) as one acceptable approach for determining the technical adequacy of the PRA once acceptable consensus approaches or models have been established for evaluations that could influence the regulatory decision. The primary result of a peer review are the facts and observations (F&Os) recorded by the peer review and the subsequent resolution of these F&Os.

Please clarify the following dispositions to fire F&Os and Supporting Requirement (SR) assessments identified in Attachment V of the LAR that have the potential to impact the FPRA results and do not appear to be fully resolved:

[This RAI includes Subparts a through h, as shown below along with NSPM responses]

NRC Request (PRA RAI 01.a):

a. *ES-C1-01: This F&O cites incomplete treatment of instrumentation needed to support fire response operator actions; however, Appendix D of FPRA-PI-ES, Revision 1, appears to only address instrumentation required for credited internal events actions.*

   Please describe how fire-induced instrument failure is addressed by the FPRA human reliability analysis (HRA) for both internal events and fire response operator actions. Include a description of how instrumentation that is relied on for credited operator actions was identified and verified as available to a level of detail commensurate with the risk importance and quantification of human error probabilities (HEPs).

NSPM Response (PRA RAI 01.a):

a. Fire-induced instrument failures have been addressed by the Fire PRA human reliability analysis. The quantification process fails the instrument cables that are mapped to the different fire scenarios within the plant. Consequently, the instruments are failed in the model, which influences the quantification of CCDP and CLERP values. Key elements of the quantification process are as follows:

   - If the instrument indications necessary to provide the cue for a given HFE are failed, the fault tree logic causes the associated human action to be unsuccessful.
   - In some situations where only partial instrumentation indications are available, but still sufficient to provide a cue for needed operator action, a specific HFE was developed that accounts for the adverse effect of degraded instrumentation in the human error probability (HEP) (as opposed to the lower HEP associated with full instrumentation).

- For some HFEs, the relied upon instrumentation is sufficiently redundant and diverse to ensure that a cue will be available for operator action, thus only the HFE is modeled in the fault tree.
- For some HFEs, the action is directed procedurally such that instrumentation-supplied cues are not required for success of the action. In this situation, no instrumentation is modeled in the FPRA, only the HFE is.

The table in Appendix D to the Equipment Selection notebook, FPRA-PI-ES, was updated in Revision 2 to include all instrumentation required for Human Failure Events credited in the Fire PRA.

NRC Request (PRA RAI 01.b):

b. *CS-A10-01: The disposition to this F&O states that cables are routed by fire area in the cable database, and according to FPRA-PI-SCA, Revision 1, this database lacks unique conduit identifiers.*

   As a result, please clarify the FPRA's treatment of conduits with unknown routing and include justification of the process used to map such conduits to fire compartments.

NSPM Response (PRA RAI 01.b):

b. The PINGP Fire PRA uses an export from SAFE GENESIS (the Fire PRA cable database) for cable routing to map targets. The SAFE GENESIS model relates cable to equipment, cable to raceway, and raceway to location for analysis purposes. The export from the cable routing database does not include unique conduit identifiers for each fire area. For example, "400-CND" identifies only that a 4 inch conduit is routed through a fire compartment. To facilitate conduit mapping to fire scenarios, the generic cable routing information from SAFE GENESIS is manipulated to map specific cable/conduit combinations to specific fire areas. The cable routing sequence for each cable is studied to evaluate the location of the cable through different raceways (cable trays and conduits). The conduit location in the routing is established by comparing the location of the known cable trays in the route sequence to the known fire area locations of the cable itself and determining the conduit to fire area relationship based on that routing. The conduit generic identification is concatenated with the cable name.

   For example, from the "cable to raceway" export from SAFE GENESIS, cable 111C-4 has five sequence points on its route:

Table 1: SAFE GENESIS Cable Route for 111C-4

CABLE	RACEWAY	SEQ ID
111C-4	400-CND	1
111C-4	1SG-LA29	2
111C-4	1SR-LA2	3
111C-4	1SM-LA4	4
111C-4	2SM-LA27	5

The first route point in the cable sequence is conduit 400-CND (First SEQ-ID in Table Q-1 above). The last four sequence route points are routed through cable trays. From the "cable to fire area" export from SAFE GENESIS, cable 111C-4 is located in 5 fire areas:

Table 2: 111C-4 to Fire Area

Cable ID	Fire Area
111C-4	29
111C-4	32
111C-4	37
111C-4	80
111C-4	81

From the "raceway to fire area" export from SAFE GENESIS, the four cable trays have known locations in fire areas 32, 80, and 81. Therefore, conduit 400-CND-111C-4 (Concatenation of Conduit ID and Cable ID) is mapped to the "remaining" fire areas, 29 and 37. Further, the conduit is also conservatively mapped to fire area 32 such that the path from the conduit to the Cable Tray in 1SG-LA29 (Route Point #2) is captured; it is assumed that the cable could run through conduit in that fire area before it reaches cable tray 1SG-LA29. Similarly, if a conduit falls in the middle of a route sequence, the conduit is mapped to the fire area of the cable tray in sequence before and after the conduit.

The approach described above provides a high level of confidence that conduits containing Fire PRA target cables are mapped to the appropriate fire compartments. For cases where additional refinement was needed for detailed fire modeling, additional walkdowns were performed to identify conduits and cable trays within the zone of influence.

NRC Request (PRA RAI 01.c):

c. *PRM-A1-02: The disposition to this F&O indicates that due to the presence of piping with soldered joints, the instrument air system is only credited for a limited number of fire scenarios within the Relay Room (Fire Area 18).*

Please justify the criteria (e.g., damage threshold, system response, etc.) used to determine those fire scenarios that do not lead to failure of the instrument air system.

NSPM Response (PRA RAI 01.c):

c. The instrument air system is not currently credited within the Fire PRA, and all scenarios result in the failure of the instrument air system.

NRC Request (PRA RAI 01.d):

d. <u>FSS-B2-01</u>: The licensee's analysis (Section 6.0 of FPRA-PI-MCR) indicates that there are "a large number of cable raceways, particularly cable trays," located under the raised floor within the main control room (MCR); however, it appears that these raceways are excluded from the MCR scenario development, both as ignition sources (i.e., self-ignited cable fires) and potential targets of other ignition sources (i.e., transient fires, transient fires due to welding and cutting, and cable fires due to welding and cutting).

Please justify this exclusion.

NSPM Response (PRA RAI 01.d):

d. There are five types of ignition sources that could be postulated as fire scenario initiators under the raised floor of the Main Control Room. These ignition sources are: 1) self-ignited cable fires, 2) junction boxes, 3) transient fires, 4) transient fires due to hotwork, and 5) cable fires due to hotwork.

The identification of cable type and linear feet was done on a cable-by-cable basis, with the total linear feet of each cable type also identified. The thermoset cable length is removed since it cannot self-ignite. Subsequently, a series of filters were applied to the thermoplastic and cables with unknown cable jacket or insulation material in order to assess the potential for self-induced cable fires at PINGP. Cables were filtered for identifying cables in conduit, low voltage (under 125VDC) cables, and cables that are in the database as placeholder duplicates for future engineering changes. The results of the applied filters indicate that the entire quantity of thermoplastic cable and cable with unknown jacket or insulation material are associated with cables in conduits, low voltage cables, or cables in the database as placeholder duplicates for future engineering changes.

The above described methodology identified the following:

Total Linear Feet:	3,825,854 ft
Thermoset Linear Feet:	86%
Thermoplastic Linear Feet:	13% (*See Note 1)
Unknown Linear Feet:	1% (*See Note 1)

*Note 1: Cables with thermoplastic jacket or insulation material and the cables with unknown insulation material are associated cables in conduits, low voltage cables, or cables outside of Fire PRA compartments.

NUREG-1805 section 7.3 states "It is common practice to consider only self-induced cable fires to occur in power cable trays since they carry enough electrical energy for

ignition. Control and instrumentation cables typically do not carry enough electrical energy for self-ignition."

Self-Ignited Cable Fires: Based on these results, it is concluded that that self-ignited cable fires do not need to be included in the PINGP Fire PRA model, including the area under the raised floor, by documenting that the thermoplastic cabling at PINGP is either in conduit or low voltage (low energy) cabling. All of the cables beneath the flooring in the control room were found to not be subject to self-ignition.

Cable Fires due to Welding/Cutting & Junction Boxes: The risk contribution of cable fires due to hotwork and junction box fires in the main control room was assessed following the guidance recommended in Fire PRA FAQ 13-0005 and Fire PRA FAQ 13-0006 respectively. The analysis suggests that the risk contribution is relatively low (i.e., less than 1E-8) for each of the ignition sources.

Transient Fires & Transient Fires due to Welding/Cutting: Transient fires (including transient fires due to hotwork) that could occur in locations near cabinets or the main control board are postulated in the Main Control Room as follows. For each panel scenario identified, a transient fire ignition frequency and a transient fire due to hotwork contribution are added. That is, the total scenario ignition frequency for a main control board panel or electrical cabinet scenario includes a contribution based on multiplying the geometric floor area factor times the total transient frequency in the control room. The postulated transient fires are considered to have the same impacts as the fixed-ignition-source-scenarios originating in the panel or cabinet. These transient fire scenarios incorporate all of the cable failures associated with the adjacent panels. For example, using the methodology from NUREG/CR-6850, transient scenarios involving a 317 kW fire postulated in the open floor area around each main control board panel or electrical cabinet includes damage to cabinets within 4 feet of the edge of the panel. Thus, all cables that are associated with cabinets in this geometric area will be included in a transient scenario. Exposed cabling, not associated with the cabinets in this geometric area that are located under the MCR floor, are excluded from these transient scenarios based on the following justification: the low likelihood of hotwork being conducted beneath the floor while the plant is in operation and the low likelihood of fire propagation due to quick suppression activities in the control room.

Finally, the fire modeling results for the Main Control Room abandonment analysis suggests that electrical cabinets alone are enough to generate abandonment conditions. Since the Fire PRA cables are mapped to the applicable panels as targets, the quantified CCDPs and CLERPs include the impact of cables that may be routed through the under floor. Furthermore, given the relative quick suppression activities in the control room (as suggested by a manual suppression curve with a constant of 0.33), the analysis assumes that transient and cabinet fires will, on average, be controlled before propagating to the underfloor.

NRC Request (PRA RAI 01.e):

e. <u>FSS-B2-01</u>: Evaluation of MCB Fires

Response to subpart (e) of this RAI will be submitted in separate correspondence by June 26, 2015.

NRC Request (PRA RAI 01.f):

f. <u>FSS-C5-01</u>: The disposition to this F&O states that self-ignited cable fires are screened from consideration for all locations on the basis that all cables are either qualified or routed through conduit, as concluded in Engineering Change (EC) 20695. However, the licensee's analysis indicates that not all cable trays identified as targets are listed in EC 20695 (Section 3.1 of FPRA-PI-RRA); there are significant amounts of thermoplastic cabling located within cable trays in Fire Area 18, a risk significant area based on risk results in Attachment W of the LAR (Section 5.1.3 of FPRA-PI-RRA); and that a differing conclusion of EC 20695 may exist (Section F.3 of FPRA-PI-SCA).

Given these discrepancies, please provide justification for the exclusion of self-ignited cable fires from the FPRA. If such fires are excluded on the basis of cable voltage, provide a technical basis for doing so. Alternatively, provide updated risk results as part of the aggregate change-in-risk analysis requested in PRA RAI 03, treating such fires consistent with accepted guidance (e.g., FAQ 13-0005).

NSPM Response (PRA RAI 01.f):

f. Self-ignited cable fires are screened for all locations in the plant based on the evaluation documented in the Engineering Change Package 20695 (EC 20695). The Engineering Change Package 20695 concludes the following:

The identification of cable type and linear feet was done on a cable-by-cable basis, with the total linear feet of each cable type also identified. The thermoset cable length is removed since it cannot self-ignite. Subsequently, a series of filters were applied to the thermoplastic and cables with unknown cable jacket or insulation material in order to assess the potential for self-induced cable fires at PINGP. Cables were filtered for identifying cables in conduit, low voltage (under 125VDC) cables, and cables that in the database as placeholder duplicates for future engineering changes. The results of the applied filters indicate that the entire quantity of thermoplastic cable and cable with unknown jacket or insulation material are associated with cables in conduits, low voltage cables, or cables in the database as placeholder duplicates for future engineering changes.

The above described methodology identified the following:

Total Linear Feet:	3,825,854 ft
Thermoset Linear Feet:	86%
Thermoplastic Linear Feet:	13% (*See Note 1)
Unknown Linear Feet:	1% (*See Note 1)

*Note 1: Cables with thermoplastic jacket or insulation material and the cables with unknown insulation material are associated cables in conduits, low voltage cables, or cables outside of Fire PRA compartments.

NUREG-1805 section 7.3 states "It is common practice to consider only self-induced cable fires to occur in power cable trays since they carry enough electrical energy for ignition. Control and instrumentation cables typically do not carry enough electrical energy for self-ignition."

Based on these results, it is concluded that that self-induced cable fires do not need to be included in the PINGP Fire PRA model by documenting that the thermoplastic cabling at PINGP is either in conduit or low voltage (low energy) cabling.

NRC Request (PRA RAI 01.g):

g. *FSS-D7-02: Total failure probability of credited detection and suppression systems*

Response to subpart (g) of this RAI will be submitted in separate correspondence by June 26, 2015.

NRC Request (PRA RAI 01.h):

h. *IGN-A1-01: Fire ignition frequencies*

Response to subpart (h) of this RAI will be submitted in separate correspondence by July 24, 2015, pending resolution of outstanding RAIs.

PRA RAI 02 – Internal Event F&Os

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. The RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the staff for adopting a fire protection program consistent with NFPA 805. The RG 1.200 describes a peer review process utilizing an associated ASME/ANS standard (currently ASME/ANS-RA-Sa-2009) as one acceptable approach for determining the technical adequacy of the PRA once acceptable consensus approaches or models have been established. The primary results of a peer review are the F&Os recorded by the peer review and the subsequent resolution of these F&Os.

Please provide clarification to the following dispositions to Internal Events F&Os and SR assessments identified in Attachment U of the LAR that have the potential to impact the FPRA results and do not appear to be fully resolved:

[This RAI includes Subparts a, b, and c, as shown below along with NSPM responses]

NRC Request (PRA RAI 02.a):

a. *QU-C2: Joint human error probabilities (HEPs)*

Response to subpart (a) of this RAI will be submitted in separate correspondence by June 26, 2015.

NRC Request (PRA RAI 02.b):

b. *SY-A8: The disposition to this F&O states that in evaluating the extent of the inconsistency highlighted by the peer review, similar modeling inconsistencies related to*

instrumentation and control components were identified yet, "the current model is considered conservative and adequate for the risk-informed NFPA-805 application".

Please summarize how this conclusion was reached.

NSPM Response (PRA RAI 02.b):

b. The referenced finding (SY-A8) found that a more consistent treatment of the component boundaries was needed. Although this Finding remains open, the current modeling is considered conservative. This conservatism is due to inclusion of individual basic events within the Fire PRA model that are already accounted for within the overall component boundary (and are therefore included in the associated failure rate) of a larger component. This results in a slight increase in the failure rates associated with these components, which is deemed conservative. This double counting originates from the Internal Events model upon which the Fire PRA is based. This conservative modeling is reflected in both compliant and variant cases of the model.

NRC Request (PRA RAI 02.c):

c. SY-B14: The disposition to this F&O does not address the issue associated with loss of pump net positive suction head (NPSH) identified by the peer review.

Please clarify whether and how the PRA assesses the impact of not crediting containment fan cooler units or containment spray on NPSH for recirculation.

NSPM Response (PRA RAI 02.c):

c. The PRA model considers potential NPSH issues related to RHR pump operability during events in which the containment sump is used as the supply source to the Reactor Coolant System (RCS). Based on pump NPSH requirements and NPSH testing and analysis, it has been concluded that:

- Pump operability would not be impacted by debris if containment systems operate as designed (sump water is passed through filter/strainer prior to pump suction).
- Pump operability would also not be adversely impacted by the rising temperatures in the containment sump and does not rely on containment over pressure for NPSH requirements.

Based on the above assumptions and supporting thermodynamic calculations, it was concluded that there is no need to model Containment Fan Coil Units and/or Containment Spray operation/failure in the accident sequence evaluations.

PRA RAI 03 – Integrated Analysis

Response to PRA RAI 03 will be submitted in separate correspondence by July 24, 2015, pending resolution of outstanding RAIs.

PRA RAI 04 – Transient Fire Placement at Pinch Points

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. The RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the staff for adopting a fire protection program consistent with NFPA 805. Methods that have not been determined to be acceptable by the NRC staff, or acceptable methods that appear to have been applied differently than described, require additional justification to allow the NRC staff to complete its review of the proposed method.

The NRC staff could not identify in the LAR a description of how "pinch points" for transient fires were treated in the FPRA. Per NUREG/CR-6850 Section 11.5.1.6, transient fires should at a minimum be placed in locations within the plant PAUs where CCDPs are highest for that PAU, i.e., at "pinch points." Pinch points include locations of redundant trains or the vicinity of other potentially risk-relevant equipment. Cable congestion is typical for areas like the Cable Spreading Room (CSR), and so placement of transient fire at pinch points in those locations is important. Hot work should be assumed to occur in locations where hot work is possible, even if improbable, keeping in mind the same philosophy.

[This RAI includes Subparts a and b, as shown below along with NSPM responses]

NRC Request (PRA RAI 04.a):

a. *Please clarify how "pinch points" were identified and modeled for general transient fires and transient fires due to hot work.*

NSPM Response (PRA RAI 04.a):

a. The ignition frequencies of transient fires, transient fires due to hotwork, and cable fires due to hotwork are not apportioned by counting ignition sources, as is the case for fixed sources such as electrical cabinets, pumps, etc. Instead, the frequency of individual fire scenarios postulated in a given compartment is apportioned using the following factors when applicable: (1) the floor area ratio (also referred as the geometrical weighting factor); (2) the maintenance, storage, and occupancy influence factors as defined in Chapter 6 of NUREG/CR-6850; and (3) cable load weighting factors as recommended in Chapter 6 of NUREG/CR-6850.

   As a practical and convenient approach, the contribution from these ignition sources is rigorously assigned to the same transient fire locations (referred to as transient zones in the PINGP Fire PRA) so that no location in the plant is left without a contribution from these ignition sources. These transient zones are defined so that the entire open floor area of the compartment is accounted for. No portion of a floor area has been excluded. Thus, the contribution from these three ignition sources has been included in every open floor area within every compartment in the scope of the Fire PRA. These fires are postulated so that they propagate through the cable trays in the transient zone and damage conduits that are also located in the transient zone. This approach ensures that no pinch points in the compartments are missed.

Trays and conduits in the transition of transient zones have been mapped to the adjacent transient zones to ensure targets are accounted for in adjacent fire scenarios.

NRC Request (PRA RAI 04.b):

b. Please describe how general transient fires and transient fires due to hot work are distributed within the PAUs at Prairie Island. In particular, identify the criteria used to determine where such ignition sources are placed within the PAUs.

NSPM Response (PRA RAI 04.b):

b. Transient fires and transient hot work fires are included in all compartments quantified in the fire PRA. For compartments where no detailed fire modeling is performed, the total transient fire frequency (e.g., transient fire frequency plus the transient hot work fire frequency) is included in the total fire compartment frequency and all fire PRA targets in the compartment are damaged. For fire compartments where detailed fire scenarios have been defined, transient fires are postulated in smaller areas designated as transient zones that together cover the total floor area in each fire compartment (i.e., there are no open floor areas where transient fire scenarios are excluded). It should be noted that there is no physical overlap between the transient zone floor areas however the targets are extended beyond the specific zones to incorporate spread of the fire.

PRA RAI 05 – Cable Fires Caused by Welding and Cutting

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. The RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the NRC staff for adopting a fire protection program consistent with NFPA 805. In a letter dated July 12, 2006, to NEI (ADAMS Accession No. ML061660105), the NRC established the ongoing FAQ process where official agency positions regarding acceptable methods can be documented until they can be included in revisions to RG 1.205 or NEI 04-02.

Appendix H of the LAR does not indicate that FAQ 13-0005, "Cable Fires Special Cases: Self-Ignited and Caused by Welding and Cutting," dated June 26, 2013, was used in preparation of the FPRA.

Please explain whether the treatment of cable fires caused by welding and cutting is consistent with FAQ 13-0005, and if not, provide justification. If justification cannot be provided, then provide treatment of such fires consistent with NRC guidance in the integrated analysis provided in response to PRA RAI 03.

NSPM Response (PRA RAI 05):

The Prairie Island Fire PRA follows the guidance of Fire PRA FAQ 13-0005 for assessing cable fires caused by welding and cutting. No exceptions from the FAQ were used. Fire PRA FAQ 13-0005 documents a method for quantifying the risk contribution of cable fires due to hot work. The FAQ suggests that cable fires due to hot work will be limited to one cable tray. Based on

the assumption that fires will be limited to one cable tray, the first two screening steps documented in Fire PRA FAQ 13-0005 suggest to calculate the CCDP for each of the cable trays in the fire compartment. The maximum CCDP per fire compartment is selected and multiplied by the cable fire due to hot work frequency assigned to the fire compartment. That multiplication results in a CDF associated with cable fires due to hot work for each fire compartment. The first two steps in the FAQ described above were performed for the fire compartment receiving detailed fire modeling analysis. For the remaining fire compartments (i.e., "those fire compartments treated as full compartment burn"), the full frequency of cable fires due to welding and cutting corresponding to the specific fire compartment is assigned.

PRA RAI 06 – Junction Boxes

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. The RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the NRC staff for adopting a fire protection program consistent with NFPA 805. In letter to NEI dated July 12, 2006 (ADAMS Accession No. ML061660105), the NRC established the ongoing FAQ process where official agency positions regarding acceptable methods can be documented until they can be included in revisions to RG 1.205 or NEI 04-02.

Appendix H of the LAR does not indicate that FAQ 13-0006, "Modeling Junction Box Scenarios in a FPRA," dated May 6, 2013, was used in preparation of the FPRA.

Please explain whether the treatment of junction box fires is consistent with FAQ 13-0006, and if not, provide justification. If justification cannot be provided, then provide treatment of junction box fires consistent with NRC guidance in the integrated analysis provided in response to PRA RAI 03.

NSPM Response (PRA RAI 06):

Fire PRA FAQ 13-0006 was used for the PINGP Fire PRA. No exceptions from the FAQ were used. Fire PRA FAQ 13-0006 guidance documents a method for quantifying the risk contribution of junction boxes. The FAQ suggests that junction box fires will be limited to one junction box. The screening process documented in the FAQ is similar to that described for cable fires due to hot work. Option 2 of the FAQ was used for apportioning junction box frequency. Since the ignition frequency of cable fires due to hot work is higher than the junction box frequency, and the count of raceways is assumed to be similar to the count of junction boxes the risk contributions for cable fires due to hot work provide a bounding estimate of what the contribution for junction boxes can be. Therefore, these are relatively low contributions compared to the plant CDF.

PRA RAI 07 – Sensitive Electronics

Response to PRA RAI 06 will be provided in separate correspondence by June 26, 2015.

PRA RAI 08 – Conditional Probabilities of Spurious Operations

Response to PRA RAI 08 will be provided in separate correspondence by June 26, 2015.

PRA RAI 09 – Counting and Treatment of Bin 15 Electrical Cabinets

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. The RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the NRC staff for adopting a fire protection program consistent with NFPA 805. Methods that have not been determined to be acceptable by the NRC staff, or acceptable methods that appear to have been applied differently than described, require additional justification to allow the NRC staff to complete its review of the proposed method.

The NRC staff could not identify in the LAR or the licensee's analysis how the licensee counted and treated Bin 15 Electrical Cabinets. In light of this, address the following:

[This RAI includes Subparts a through d, as shown below along with NSPM responses]

NRC Request (PRA RAI 09.a):

a. *Per Section 6.5.6 of NUREG/CR-6850, fires originating from within "well-sealed electrical cabinets that have robustly secured doors (and/or access panels) and that house only circuits below 440V" do not meet the definition of potentially challenging fires and, therefore, should be excluded from the counting process for Bin 15. By counting these cabinets as ignition sources within Bin 15, the frequencies applied to other cabinets may be inappropriately reduced.*

   Please clarify that this guidance is being applied. If not, then address the impact as part of the integrated analysis performed in response to PRA RAI 03.

NSPM Response (PRA RAI 09.a):

a. Cabinets were counted in Bin 15 following the guidance of NUREG/CR-6850. Additional supporting guidance from Supplement 1 of NUREG/CR-6850 was used on characterizing and counting this bin, including dividing Bin 15 into two bins, Bin 15.1 – Non-High Energy Arcing Fault (HEAF) and Bin 15.2 – HEAF. Consistent with this guidance, well-sealed electrical cabinets that have robustly secured doors (and/or access panels) and that house only circuits below 440V were not counted as electrical cabinets (i.e., Bin 15).

NRC Request (PRA RAI 09.b):

b. *Please clarify if the criteria used to evaluate whether electrical cabinets below 440V are "well sealed" are consistent with guidance in Chapter 8 of Supplement 1 of NUREG/CR-*

6850. If not, then address the impact as part of the integrated analysis performed in response to PRA RAI 03.

NSPM Response (PRA RAI 09.b):

b. The counting guidance provided in Chapter 8 of Supplement 1 of NUREG/CR-6850 is followed when counting "well sealed" cabinets below 440V.

NRC Request (PRA RAI 09.c):

c. All cabinets having circuits of 440V or greater should be counted for purposes of Bin 15 frequency apportionment based on the guidance in Section 6.5.6 of NUREG/CR-6850.

Please clarify that this guidance is being applied. If not, then address the impact as part of the integrated analysis performed in response to PRA RAI 03.

NSPM Response (PRA RAI 09.c):

c. Identified cabinets having circuits above 440V were counted in Bin 15 following the guidance of NUREG/CR-6850. Additional supporting guidance from Supplement 1 of NUREG/CR-6850 was used on characterizing and counting this bin, including dividing Bin 15 into two bins, Bin 15.1 – Non-HEAF and Bin 15.2 – HEAF.

NRC Request (PRA RAI 09.d):

d. For those cabinets that house circuits of 440V or greater, propagation of fire outside the ignition source should be evaluated based on guidance in Chapter 6 of NUREG/CR-6850, which states that "an arcing fault could compromise panel integrity (an arcing fault could burn through the panel sides, but this should not be confused with the high energy arcing fault type fires)."

Please describe how fire propagation outside of well-sealed cabinets greater than 440 V is evaluated. If propagation is not evaluated, then address the impact as part of the integrated analysis performed in response to PRA RAI 03.

NSPM Response (PRA RAI 09.d):

d. Well-Sealed cabinets were not considered for cabinets greater than 440V in the Ignition Frequency binning process; as such fires in these cabinets are evaluated as if the cabinet was vented (i.e., fire propagation was modeled outside of the cabinets). Fire propagation outside of electrical cabinets greater than 440V is evaluated as outlined in Chapter 6 of NUREG/CR-6850.

PRA RAI 10 – High Energy Arcing Faults (HEAF)

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. The RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-

02, Revision 2, as providing methods acceptable to the NRC staff for adopting a fire protection program consistent with NFPA 805. Methods that have not been determined to be acceptable by the NRC staff, or acceptable methods that appear to have been applied differently than described, require additional justification to allow the NRC staff to complete its review of the proposed method.

The NRC staff could not identify in the LAR or licensee's analysis a description of how HEAFs were modeled. Per Appendix P of NUREG/CR-6850, HEAF events and other types of fires have different non-suppression probability (NSP) curves. In addition, the NRC staff's interpretation of the NUREG/CR-6850 guidance is that the growth of a fire subsequent to a HEAF event, unlike other types of fires, instantaneously starts at a non-zero HRR because of the intensity of the initial heat release from the HEAF.

Please confirm that HEAF events have been modelled using the acceptable HEAF evaluation methods. If alternative methods have been used, provide a justification of the FPRA's treatment of HEAF events and the ensuing fire that includes a discussion of conservatisms and non-conservatisms relative to the accepted methods and assesses the associated impacts on the fire total and delta risk results. Alternatively, replace the current approach with an acceptable approach in the integrated analysis performed in response to PRA RAI 03. Note that the response should address the treatment of all HEAF scenarios, including in the hot gas layer and multi-compartment analyses.

NSPM Response (PRA RAI 10):

The PINGP Fire PRA models high energy arcing faults using acceptable evaluation methods. The Fire PRA HEAF analysis follows the guidance described in Appendix M of NUREG/CR-6850. This approach suggests that high energy arcing faults are events with two distinct phases: an initial energetic phase (the arcing explosion), and an ensuing fire phase. Based on this approach, the peak heat release rate value is assigned to the switchgear or load center where the high energy arcing fault is postulated. It is assumed that the peak heat release rate is achieved at the time of the explosion (i.e., no fire growth phase is credited). Fire propagation through secondary combustibles above the switchgear or load center is modeled following the guidance described in Appendix M of NUREG/CR-6850.

The high energy arcing fault fire scenarios in the Fire PRA model are quantified using the suppression curve listed in Table 14-2 of Supplement 1 to NUREG/CR-6850 for high energy arcing fault scenarios. This suppression curve for high energy arcing faults is used for determining the non-suppression probabilities associated with the different damage states associated with each fire scenario. The Non-Suppression Probability (NSP) credit taken for HEAF scenarios is 0.011.

PRA RAI 11 – Time to Delayed Detection

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. The RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and

describes a general framework to determine the acceptability of risk-informed changes. The NRC staff's review of the information in the LAR has identified additional information that is required to fully characterize the risk estimates.

The licensee's analysis (Section 3.0 of FPRA-PI-SCA) appears to indicate that regardless of location, the FPRA assumes a time to delayed detection of 10 minutes instead of the 15 minutes used in Appendix P of NUREG/CR-6850 for fire scenarios should automatic detection be unavailable or a fire watch not be present.

Please use the generic 15 minutes or provide justification for using 10 minutes.

NSPM Response (PRA RAI 11):

The PINGP Fire PRA no longer uses the delayed detection time of 10 minutes. The time to delayed detection has been updated to 15 minutes.

An assumption has been included within the FPRA-PI-SCA notebook to state the usage of a 15 minute delayed detection time. The Integrated analysis associated with the response to PRA RAI 03 will reflect the use of a 15 minute delayed detection time.

PRA RAI 12 – MCR Abandonment

Response to PRA RAI 12 will be provided in separate correspondence by June 26, 2015.

PRA RAI 13 – Calculation of ΔCDF, ΔLERF and Additional Risk of Recovery Actions

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. The RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. The NRC staff's review of the information in the LAR has identified additional information that is required to fully characterize the risk estimates.

Section W.2 of the LAR provides some description of how the change-in-risk and the additional risk of recovery actions associated with variances from deterministic requirements (VFDRs) is determined but not enough detail to make the approach completely understood. As a result, please provide the following:

[This RAI includes Subparts a, b, and c, as shown below along with NSPM responses]

NRC Request (PRA RAI 13.a):

a. A summary of how the change in risk was determined for fire areas that credit MCR abandonment due to loss of habitability and due to loss of control (e.g., Relay Room). Include a discussion of how the CCDPs/CLERPs were determined for both the variant plant and the compliant plant models for these areas.

NSPM Response (PRA RAI 13.a):

a. The change in risk (i.e., delta risk) associated with VFDRs was calculated as the difference in risk between the variant plant (i.e., the post-transition plant including plant modifications) and the compliant plant (i.e., the post-transition plant in which the VFDRs are assumed to be deterministically resolved).

The change in risk in Fire Areas 13 and 18 was calculated as a delta core damage frequency (delta CDF) and delta large early release frequency (delta LERF) for each Unit in the same manner as for other fire areas and was evaluated as follows.

A determination of whether the control room would be abandoned or not was made for each fire scenario within the Fire PRA. A fire scenario that was found to lead to abandonment in the variant plant was also considered to lead to abandonment in the compliant plant. MCR abandonment was credited only for some scenarios occurring in Fire Areas 13 (Control Room) and 18 (Relay and Cable Spreading Room).

For each fire scenario, the conditional core damage probability (CCDP) and conditional large early release probability (CLERP) of the variant plant were calculated. To calculate the CCDP (CLERP) associated with a given VFDR, the compliant plant was modeled by altering the model logic to mimic a plant where the VFDR was deterministically resolved, i.e., it no longer existed. In particular:

- Components credited to establish the relied-upon success path and identified as impacted in the VFDR were assumed to be unaffected by fire impacts.
- For recovery actions credited to help resolve the VFDR, the execution portion of the human error probability (HEP) representing the recovery action failure was set to zero. This approach was selected to represent a compliant plant where the operator action is assumed to take place in the control room (when it is not abandoned) or at a hypothetical primary control station (when the control room is abandoned), and therefore is not a recovery action. Setting the execution portion of the HEP to zero is conservative because operator actions, even when taking place under optimal conditions in the control room, may fail to be executed properly. Setting the execution portion to zero is otherwise an adequate proxy to represent the fact that, in the compliant plant, operator actions would be easier to execute than in the variant plant, where these actions are performed away from the control room. The potential for failing to take a required action (cognitive failure) exists both in the variant and compliant plant; therefore, this portion of the HEP is kept in both the variant and compliant plant models. For some VFDRs, the cognitive failure portion of the HEP was also set to zero in the compliant plant, in effect conservatively setting the entire HEP of the recovery to zero in the compliant plant.

- VFDRs for which a credited component or recovery action was not modeled in the Fire PRA are addressed separately in the response to Item c of this RAI.

The changes described above lead to a CCDP (CLERP) in the variant plant higher than in the compliant plant. The change in risk for a given VFDR is calculated as the sum, over all the fire scenarios impacted by the VFDR, of the difference in CDF (LERF) of each fire scenario, calculated as the fire scenario frequency multiplied by the difference in variant plant CCDP (CLERP) minus compliant plant CCDP (CLERP).

NRC Request (PRA RAI 13.b):

b. A description of how the reported additional risk of recovery actions was calculated, including any special calculations performed for the MCR and other abandonment areas (if applicable). Note that it is unclear why the discussion provided in Section W.2.2 of the LAR states that "the additional risk of recovery actions is calculated separately by comparing the fire area CDF and LERF of the variant and compliant plant" when the additional risk of recovery actions is not equivalent to the delta risk for all fire areas.

NSPM Response (PRA RAI 13.b):

b. For a given fire area, the additional risk of recovery actions was calculated as the difference in CDF (LERF) in the variant plant (post-transition plant where the credited recovery actions take their nominal HEPs), and the CDF (LERF) of the post-transition plant where the credited recovery actions have the execution portion of their HEP set to 0 (or, for some VFDRs, the entire HEP of the associated credited recovery action conservatively set to 0). This method is applied to all fire areas, regardless of whether a fire area credits, or not, control room abandonment.

This calculation method was selected based on the following considerations:

- Because the fire area CDFs (LERFs) calculated with that method differ only by their HEP values, the resulting differences in risk are due to credited recovery actions only. This differs from the general calculation of changes in risk (delta risks) described in the response to Item a of this RAI. The overall change in risk of a fire area could be used as an adequate, bounding proxy for the additional risks of its recovery actions, but would suffer from the drawback that some of its underlying cutsets pertain to equipment lost to the fire in the variant plant, but otherwise not related to the recovery actions credited in the fire area.
- Setting the execution portion of the HEP to 0 is conservative because no recovery action is perfectly reliable. That is, in reality, the execution portion of the HEP is always greater than 0. Consequently, the differences in CDF and LERF calculated with this method are conservative approximations of the additional risk of recovery actions. There is added conservatism when the entire HEP is set to 0.

In Section W.2.2 of the LAR, the sentence stating: "the additional risk of recovery actions is calculated separately by comparing the fire area CDF and LERF of the variant and compliant plant" was intended to outline the fact that the additional risk of recovery actions was not approximated by the change in risk (delta CDF and delta LERF due to VFDRs in each fire area, calculated as described in response to Item a of this RAI).

While such an approximation would be adequate and bounding, it was judged that a separate calculation focusing on the changes in CDF and LERF due only to changes in HEPs of credited recovery actions (representing the differences in HEPs between the variant and compliant plant) was a more accurate representation of the additional risk of recovery actions.

NRC Request (PRA RAI 13.c):

c. *A summary of the types of VFDRs that were identified but not modeled in the FPRA. Include any qualitative rationale for excluding these from the change-in-risk calculations.*

NSPM Response (PRA RAI 13.c):

c. There are essentially two types of VFDRs that were identified but not modeled in the FPRA, "not modeled" here meaning that no change to the compliant plant logic was needed to model the VFDR. They are as follows:

- VFDRs resolved by plant modification. These VFDRs correspond to the cases where a proposed plant modification provides a success path for the nuclear safety performance criteria previously identified as challenged in the VFDR, thereby indicating that the VFDR will no longer exist when the modification is implemented. In these cases, the delta risk is zero.
- VFDRs with an insignificant delta risk based on a qualitative evaluation. Some components identified in a VFDR as potentially failed due to fire effects may not be modeled in the Fire PRA. This was done in the situations where it was shown that the change in risk associated with the component is insignificant. An example would be the situation where there is sufficient diversity and redundancy of instrumentation to ensure that a monitoring parameter, identified as lost in a VFDR, would be in fact available. In effect, the delta risk is zero.

PRA RAI 14 – Attachment W Inconsistencies

Inconsistencies were noted within Attachment W for particular fire areas. In light of this,

[This RAI includes Subparts a, b, and c, as shown below along with NSPM responses]

NRC Request (PRA RAI 14.a.i):

a. *Provide clarification on the following inconsistencies, and discuss their significance to the risk results reported in Tables W-6 and W-7.*

   (i) *In Table W-6, Unit 1 Fire Area 84 is indicated as having VFDRs (i.e., there is a "Yes'" under the "VFDR" column); however, there is an "N/A" in the column for ΔCDF/ΔLERF.*

NSPM Response (PRA RAI 14.a.i):

(i) This is a typo; there is no VFDR for Fire Area 84. Documentation for Attachment W has been updated to reflect that there should be a "No" in the VFDR column of Table W-6 for Fire Area 84. A revision to LAR Attachment W will be provided with the response to PRA RAI 03.

NRC Request (PRA RAI 14.a.ii):

(ii) In Table W-7, Unit 2 Fire Areas 1 and 20 are indicated as having no VFDRs (i.e., there is a "No" under the "VFDR" column); however, there is an "ε," or epsilon in the column for ΔCDF/ΔLERF.

NSPM Response (PRA RAI 14.a.ii):

(ii) This is a typo; VFDRs exist for Fire Areas 1 and 20. Documentation for Attachment W has been updated to reflect that there should be a "Yes" in the VFDR column of Table W-7 for Fire Areas 1 and 20. A revision to LAR Attachment W will be provided with the response to PRA RAI 03.

NRC Request (PRA RAI 14.b):

b. Describe what is meant by the use of "ε," or epsilon, in columns for Fire Area ΔCDF/ΔLERF and additional risk of RAs. Address if epsilon is defined by a specific cut-off value(s).

NSPM Response (PRA RAI 14.b):

b. Epsilon or "ε" is a common mathematical term used to represent a small positive infinitesimal quantity, whose limit is usually taken as $\varepsilon \rightarrow 0$. There was no "cutoff" value used for epsilon, other than the FRANX truncation limits (1E-12 for CDF and 1E-13 for LERF) applied during the baseline and compliant case quantifications. In the specific case of this analysis, the variable epsilon was used if either there was no impact to PRA-modeled equipment due to the VFDR, or if the impact was too small to survive truncation.

A note was added to Attachment W explaining epsilon, as follows:

> "ε" represents a small positive infinitesimal quantity whose impact is too small to affect the analysis.

An updated version of Attachment W will be provided with the response to PRA RAI 03.

NRC Request (PRA RAI 14.c):

c. Describe what is meant by the use of "N/A" in columns for Fire Area CDF/LERF, ΔCDF/ΔLERF and additional risk of RAs.

NSPM Response (PRA RAI 14.c):

c. If no Fire Risk Evaluation was required for the specific Fire Area an "N/A" (i.e., Not Applicable) was assigned for the columns CDF/LERF, ΔCDF/ΔLERF, and Additional Risk of RAs.

For some Fire Areas there was an FRE (i.e., the Fire Area had VFDRs that were evaluated using performance based methods) but had no recovery actions credited. In those cases there was either a delta-risk or epsilon provided for the Fire Area delta-risk columns, but "N/A" was used for the "Additional risk of recovery actions" columns.

A note was added to Attachment W explaining "N/A," as follows:

> "N/A" indicates that no Fire Risk Evaluation was required or no Recovery Actions were credited.

An updated version of Attachment W will be provided with the response to PRA RAI 03.

PRA RAI 15 – Implementation Item Impact on Risk Estimates

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. The RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. The NRC staff's review of the information in the LAR has identified additional information that is required to fully characterize the risk estimates.

Implementation Item 20 in Table S-3 of the LAR commits to updating the FPRA and verifying the risk results after Table S-2 plant modifications have been incorporated. However, Table S-3 includes a number of procedural modifications that may affect the as-built and as-operated plant risk models.

[This RAI includes Subparts a, b, and c, as shown below along with NSPM responses]

NRC Request (PRA RAI 15.a):

a. Please update Implementation Item 20 to include incorporation of all risk relevant modifications in Tables S-1, S-2, and S-3 into the FPRA before the final risk result verification.

NSPM Response (PRA RAI 15.a):

a. An updated Implementation Item 20 that commits to including all risk-relevant items in Tables S-1, S-2, and S-3 into the FPRA before the final risk result verification is provided below. The Item is also updated for clarity with respect to the actions that will be taken should the quantified risk measures for either unit exceed the RG 1.205 acceptance guidelines (see response to PRA RAI 15(b) below):

Current description provided in Table S-3 of the April 30, 2014 LAR Supplement:

Item 20:
Update the Fire PRA Model, as necessary, after all modifications identified in Table S-2 are complete and as-built. If the revised Fire PRA indicates an increase in risk metrics such that the RG 1.205 acceptance guidelines are not met, the configuration control process described in LAR Section 4.7.2 will be implemented.

New, revised Table S-3 description:

Item 20:
Update the Fire PRA Model, as necessary, after all modifications, <u>procedure changes, and other risk-relevant items</u> identified in Table<u>s S-1,</u> S-2<u>, and S-3</u> are complete and as-built. If the revised Fire PRA indicates an increase in risk metrics such that the RG 1.205 acceptance guidelines are not met<u>, changes will be made such that the Fire PRA results will fall within the acceptance guidelines. These changes may include additional analysis, procedure enhancements, plant modifications, or other changes determined to be necessary to reduce the overall risk metrics to within the acceptance guidelines.</u>

NRC Request (PRA RAI 15.b):

b. *Implementation Item 20 states, "If the revised Fire PRA indicates an increase in risk metrics such that the RG 1.205 acceptance guidelines are not met, the configuration control process described in LAR Section 4.7.2 will be implemented."*

   Please clarify this statement.

NSPM Response (PRA RAI 15.b):

b. The intent of Table S-3 Implementation Item 20 is to state that after implementation of all plant modifications (and after implementation of all other risk-relevant S-1, S-2 and S-3 table items; see response to PRA RAI 15.a above), the Fire PRA will be revised to incorporate these changes, and the internal fires risk metrics will be recomputed. If these metrics are found to exceed the RG 1.205 acceptance guidelines, then any necessary changes will be made to reduce the overall risk metrics to within the acceptance guidelines. Table S-3 Implementation Item 20 is revised as shown in the response to PRA RAI 15.a above to better describe the analysis that will be performed.

NRC Request (PRA RAI 15.c):

c. *Tables S-1 and S-2 include the new RCP seals for which no acceptable PRA model exists yet and the time until an acceptable model exists is difficult to determine.*

   Please clarify how transition to NFPA-805 could be achieved with the current implementation items if an acceptable RCP seal model is delayed for an extended time.

NSPM Response (PRA RAI 15.c):

c. All of the Reactor Coolant Pump (RCP) seals in both Prairie Island units have now been replaced with the Flowserve Three-Stage N-9000 Seal package with the Abeyance Seal.

   The new RCP seals for the Prairie Island Nuclear Generating plant are modelled using general PRA methods consistent with the consensus model endorsed by the NRC: WCAP-16175-P-A, Model for Failure of RCP Seals Given Loss of Seal Cooling in CE NSSS Plants. The model developed for use in the Prairie Island internal events PRA models and for the Fire PRA uses the failure model of WCAP-16175 as the basis and then incorporates the effect of failure modes and the Abeyance Seal on resulting seal failure flow rates. Additionally, the SER for WCAP-16175 acknowledges that no damage to the seals is expected if the pumps are tripped within 20 minutes of the loss of seal cooling. However, WCAP-16175 evaluated pumps with a smaller volume of cold water around the thermal barrier heat exchanger. Therefore, the PRA model for Prairie Island has provided for additional time available to purge the cold water from Westinghouse pumps, consistent with the analyses in WCAP-16175. The Flowserve report, PRA Model for Flowserve 3 Stage N-Seals with Abeyance Seal, Revision 0, dated 12/20/13, provides the bases for the development of the PRA model used in the Prairie Island fire PRA. Transition to NFPA-805 can be achieved with the current implementation items by revising Table S-3 to include the following new Implementation Item #66:

   > Item 66:
   > Upon NRC approval of the Flowserve topical report for the Reactor Coolant Pump (RCP) seals and related PRA model, the Prairie Island PRA model shall be reviewed using the final version of the topical report as well as any exceptions/clarifications included in the NRC approval to determine if the internal events and Fire PRA require a revision. The Prairie Island internal events and Fire PRA will be updated, if applicable, with the latest RCP seal information. If the updates result in a risk increase greater than the self-approval limits (1E-07/yr for CDF and 1E-08/yr for LERF), NSPM will take action to reduce the risk results to within the self-approval limits. Compensatory measures established prior to the RCP seal replacement shall remain in place until the calculated risk increase is within the self-approval limits.

PRA RAI 16 – Use of Incipient Detection

Response to PRA RAI 16 will be provided in separate correspondence by June 26, 2015.

PRA RAI 17 – RCP Seal PRA Modeling

Response to PRA RAI 17 will be provided in separate correspondence by June 26, 2015.

PRA RAI 18 – Deviations from Acceptable Methods

Response to PRA RAI 18 will be provided in separate correspondence by June 26, 2015.

PRA RAI 19 – Defense-in-Depth and Safety Margin

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. The RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. The NRC staff's review of the information in the LAR has identified additional information that is required to fully characterize the risk estimates.

Section 4.5.2.2 of the LAR provides a high-level description of how the impact of transition to NFPA 805 impacts defense-in-depth (DID) and safety margin was reviewed, including using the criteria from Section 5.3.5 of NEI 04-02 and from RG 1.205. However, no explanation is provided of how specifically the criteria in these documents were utilized and/or applied in these assessments.

[This RAI includes Subparts a and b, as shown below along with NSPM responses]

NRC Request (PRA RAI 19.a):

a. *Please provide further explanation of the method(s) or criteria used to determine when a substantial imbalance between DID echelons existed in the Fire Risk Evaluations (FREs), and identify the types of plant improvements made in response to this assessment.*

NSPM Response (PRA RAI 19.a):

a. Section 1.2 of NFPA 805 defines defense-in-depth (DID) as:
   1. Preventing fires from starting
   2. Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage
   3. Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

   In general, DID is considered satisfied if the proposed licensing change does not result in a substantial imbalance among these elements (or echelons). The review of DID is qualitative and addresses each of the elements with respect to the proposed licensing change. It involves a review of plant documents such as the fire protection program, pre-fire plans, and administrative procedures.

   In the context of the NFPA 805 transition, the DID evaluation accounts for the fact that the fundamental elements of the fire protection program and the design requirements for fire protection systems and features have been addressed in a manner consistent with the requirements of Chapter 3 of NFPA 805. Accordingly, the DID evaluation focuses on potential enhancements that may be required to maintain the balance of DID echelons.

Systems and features associated with the first echelon of Defense in Depth (DID) deal with control of combustibles and control of hot work. Generally, the Fire PRA assumes limited credit for Administrative Controls that may limit the size, placement, and frequency of transients. Unless transient placement is precluded by design, the Fire PRA assumes transients can be placed near targets and hot work transients anywhere in the 3-dimensional space near pinch points. With this approach, the Fire PRA helps uncover potential plant vulnerabilities regarding transients or hot work locations and provides insights on the adequacy of Echelon 1. Combustible Control and Hot Work Control were indirectly credited in the Fire PRA through the ignition frequencies for transient combustibles and hot work. Because these administrative controls reduce risk of fires in all nuclear plants, generic ignition frequencies used in the Fire PRA credit these controls to be in place. The fire risk evaluations (FREs) found that the existing combustible and hot work controls were adequate and no plant improvement was required for this echelon.

The second echelon of DID involves the detection and suppression of fires. To evaluate the adequacy of this echelon, insights from the Fire PRA can be used as a reference point. For example, fire detection and associated fire pre-plan procedures in a given fire area may be credited in order to meet DID considerations for this area, but otherwise may not have been credited in the Fire PRA. This could be based on the consideration that, for example, the relatively significant amount of diesel fuel oil present in the area could challenge firefighting activities, thereby making it beneficial to credit detection for DID, to ensure that the fire is promptly controlled and extinguished. The fire risk evaluations (FREs) found that the existing fire protection features and pre-fire plans were adequate and that no plant improvement was required for this echelon.

The third echelon of DID deals with the adequacy of fire barriers, fire rated cables and systems free of fire damage including procedural controls. This echelon also deals with operator actions such as recovery actions aimed at mitigating the adverse effects of the fire. As for the other echelons, the Fire PRA can be used as a reference point to identify potential improvements for this echelon. The FREs identified improvements for this echelon. For example, procedure F5 Appendix D, *Impact of Fire Outside Control/Relay Room*, was updated to include several additional fire areas in the list of fire areas where a fire could potentially require manual operation of cooling water strainers.

NRC Request (PRA RAI 19.b):

b. *Please provide further discussion of the approach in applying the criteria for assessing safety margin in the FREs as described in NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," Revision 2, dated April 2008 (ADAMS Accession No. ML081130188).*

NSPM Response (PRA RAI 19.b):

b. In accordance with Section 5.3.5.3 of NEI 04-02, the adequacy of Safety Margin is assessed by the consideration of categories of analyses utilized by the FRE. Safety margins are considered to be maintained if:

   - Codes and standards or their alternatives accepted for use by the NRC are met, and

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

The following summarizes the bases for ensuring the maintenance of safety margins:

- The risk-informed, performance based processes utilized are based upon NFPA 805, endorsed by the NRC in 10 CFR 50.48(c).
- The fire risk evaluation process is in accordance with NEI 04-02, which is endorsed by the NRC in Regulatory Guide 1.205.
- The Fire PRA is developed in accordance with NUREG/CR-6850, which was developed jointly between the NRC and EPRI.
- The internal events PRA and Fire PRA have received a formal industry peer review based on the NEI guidelines, in order to ensure the Fire PRA meets the appropriate quality standards of ASME/ANS RA-Sa-2009. Peer review of the Fire PRA model has been conducted by diverse groups of PRA practitioners from other PWR plants and industry. These reviews generally cover all aspects of the Fire PRA model and the administrative processes used to maintain and update the model. The peer review has generated specific recommendations for model changes, as well as guidance for improvements to processes and methodologies used in the Fire PRA model, and enhancements to the documentation of the model and the administrative procedures used for model updates. The Fire PRA model and administrative requirements are assessed and revised or clarified to address the issues identified through peer reviews.
- Fire protection systems and features determined to be required by NFPA 805 Chapter 4 have been confirmed to meet the requirements of NFPA 805 Chapter 3 and their associated referenced codes and listings, or provided with acceptable alternatives using processes accepted for use by the NRC.
- Fire modeling performed in support of the transition has been performed within the Fire PRA utilizing codes and standards developed by industry and NRC staff which have been verified and validated in authoritative publications, such as NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." In general, the fire modeling performed in support of the fire risk evaluations has been performed using conservative methods and input parameters that are based upon NUREG/CR-6850.

Enclosure 2

Licensee Identified Changes

This Enclosure identifies changes to LAR sections not directly related to RAI responses, and includes the following:

LAR Section	**Change**
Attachment S	Revise schedule statements on Page S-2 to clarify that two full refueling cycles are needed for completion of modifications
Attachment S	Delete S-3 item 30

Licensee Identified Issue #1: Modification Schedule

The implementation schedule in Attachment S, page S-2, should be revised for consistency with Section 5.5, Transition Implementation Schedule. NSPM is requesting two full operating cycles, per unit, to complete the modifications listed in Table S-2, Plant Modifications Committed. This avoids the need to work on both protection trains in the same outage, consistent with PINGP outage scheduling practices.

Section 5.5 identifies that Table S-2 modifications will be completed before the end of the second full operating cycle for each unit after approval of the LAR. Also, Table S-2, page S-2 includes a statement that NSPM is requesting "two full refueling cycles" beyond SE issuance to fully implement modifications. However, page S-2 also includes a statement that S-2 modifications will be complete by the "completion of the second refueling outage" per unit after issuance of the NFPA 805 license amendment; this would not allow the two full operating cycles (which are the same as refueling cycles) that are described in Section 5.5 and were intended by NSPM. The modification schedule statement on page S-2 of the LAR should be changed to read as follows:

NSPM will complete implementation of the modifications described in Table S-2 as follows:

- By the completion of the first refueling outage per unit after issuance of the NFPA 805 license amendment: Items 8, 9, and 16
- By the completion of the refueling outage after the second full refueling cycle per unit after issuance of the NFPA 805 license amendment: all remaining items

Attachment S will be revised and submitted with the response to PRA RAI 03, and this update will include the above change.

Licensee Identified Issue #2: Delete S-3 Item Regarding Compressed Air Bottles

Table S-3, Item 30 is an Implementation Item to "Revise/initiate procedures and/or procure additional compressed air bottles to achieve 30 hours to ensure we are 'safe and stable' at 24 hours." This item was included in the initial 2012 LAR submittal (Reference 1 to the cover letter for this enclosure). However, the revised analyses for the 2014 LAR Supplement (Reference 2 to the cover letter for this enclosure) no longer require compressed air and this item was inadvertently not deleted from Attachment S. Attachment S will be revised and submitted with the response to PRA RAI 03, and the update will include the deletion of Item 30 from Table S-3.

Enclosure 3

LAR Attachment L (revised)

The following pages include a revision to

LAR Attachment L, NFPA 805 Chapter 3 Requirements for Approval,

and includes information in response to FPE RAI 05.

7 pages follow

L. NFPA 805 Chapter 3 Requirements for Approval (10 CFR 50.48(c)(2)(vii))

6 Pages Attached

Note that Revision 1 is a complete revision and no revision indicators are included.

Approval Request 1

NFPA 805 Section 3.5.16

NFPA 805 Section 3.5.16 states:

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

Basis for Request:

Contrary to the requirements of NFPA 805 Section 3.5.16, the fire protection water supply system at PINGP may periodically be utilized to supply water for non-fire protection purposes. NRC approval is being requested since the fire water system can be aligned for screenwash system use and spent fuel pool makeup, and as such it does not meet the requirement or allowed exceptions.

Acceptance Criteria Evaluation:

The Mississippi river provides fire protection water. The system consists of two horizontal centrifugal fire pumps each rated at 2,000 gpm at 125 psig. One pump is motor driven (MDFP) and the other pump is diesel driven (DDFP). The 10" fire header is maintained between 108 and 113 psig by a jockey pump. The motor driven fire pump will automatically start at 95 psig. If the header pressure drops to 90 psig, the diesel-driven fire pump will start. The motor and diesel-driven fire pumps are designed to pump 2,000 gpm at a discharge pressure of 125 psi. The screenwash pump can be aligned to the fire protection header to supply 2,000 gpm at a discharge pressure of 125 psi.

Each of the requested non-fire protection uses is discussed below.

<u>Fire protection water for screenwash function:</u>

The motor-driven fire pump can be aligned to provide a backup water supply to the Screenhouse screenwash system in the event the screenwash pump is unavailable.

To use the fire pump to supplement the screenwash header flow, the pump must be started manually either locally or remotely. Once the pump is operating, if no auto start signal exists from the fire protection header (i.e., if pressure in the fire water header has not been reduced by other uses), the discharge valve to the screenwash header (Control Valve CV-31131) opens automatically via Solenoid Valve SV-33049 and supplies water to the screenwash header when required. If a demand is placed on the fire header from a suppression system actuation, the fire protection header low pressure signal will cause SV-33049 to close CV-31131 thereby realigning all water flow to the fire protection header. The diesel driven fire pump (DDFP) will also be available to supply water to the fire protection water piping.

There is also a bypass line around control valve CV-31131. If manual valve FP-27-1 in the bypass line around CV-31131 is opened, it will be manually closed in the event of a fire to

realign water flow to the fire protection header. If the manual bypass valve (FP-27-1) to divert fire water to the screenwash system is opened, it is tracked as a fire impairment to the MDFP, and therefore operators are aware and can shut the manual bypass valve to restore fire protection water flow and pressure to the fire protection header if needed.

There are two potential impacts to the use of the fire protection water system for screenwash purposes – one is a fire within the Screenhouse Fire Area 41, and one for all other fire areas, as delineated below:

For a fire in any fire area except Fire Area 41

The plant P&ID drawing indicates that SV-33049 will only open CV-31131 if the 121 MDFP is not running and the pressure demand on the fire header is not below 90 psi. If there is not a demand on the fire header and the pump is manually started, the control valve will open and allow fire protection water into the Screenwash system to clean the screens. If a demand is placed on the fire header from a suppression system actuation, the pressure drop will cause SV -33049 to close CV-31131 thereby realigning the water flow to the fire protection header.

The control cables for SV-33049/CV-31131 run from the MDFP room FA 41B (elev. 670' screenhouse) to FA 41, (elev. 695' screenhouse).

A fire event in any fire area other than FA 41 will cause a pressure drop on the fire header thereby closing CV-31131 via SV-33049 and re-aligning the MDFP water supply from the traveling screens to the fire header.

For a fire in Fire Area 41

The control cables for SV-33049/CV-31131 run from the MDFP room FA 41B (elev. 670' screenhouse) to FA 41, (elev. 695' screenhouse). A postulated fire in FA 41 may cause cable damage, thereby creating the potential for a hot short of the circuit resulting in CV-31131 failing in the open position. This would divert some fire protection water from the MDFP away from the fire header. If this scenario occurs and there is a demand on the fire header, the DDFP will start, providing the fire header with 2,000 gpm at 125 psi. FA 41 suppression system, PA-9, has a demand of 1,094.1 gpm at 89.1 psi. This demand is within the design capacity of the DDFP. Check valve FP-28-2 will prevent water in the fire protection header from entering the Screenwash diversion piping network.

The use of the MDFP for Screenwash cleaning will not impact the ability of the fire protection header to deliver the system demand for fire suppression activities in any plant fire area.

<u>Fire protection water to maintain spent fuel pool inventory:</u>

Fire protection water is available through hose stations to supply water to the spent fuel pool. As described below, there are six other preferred sources of water that will be used first to maintain inventory in the spent fuel pool (SFP). However, in the event that all other water sources are unavailable, fire protection water may be needed to maintain spent fuel pool inventory.

One of the justifications for this non-fire protection use of fire protection water is that it is unlikely that fire protection water would be needed to simultaneously extinguish a fire using the highest system demand and supply makeup water to the spent fuel pool. NUREG/CR-6850, EPRI/NRC-RES, Fire PRA Methodology for Nuclear Power Facilities, states that multiple initiating events from the same root cause may use a qualitative approach to address those interactions resulting between fire and a seismic event. Consistent with this guidance, a

qualitative discussion is provided to address events requiring the simultaneous need for firewater for fire suppression purposes and for spent fuel pool inventory makeup.

Actions to makeup spent fuel pool inventory are well described in an existing abnormal plant procedure, C16 AOP01, Loss of SFP Inventory. There are four preferred borated sources for make-up water and two non-borated sources that will be depleted before fire protection water is utilized for spent fuel pool inventory replenishment.

Use of the other six water sources for SFP inventory makeup provides time to respond to the fire and/or get additional sources of equipment and resources. The existing mutual aid agreement with the City of Red Wing Fire Department and their participation in annual fire drills will ensure that they can provide a timely response and can also provide the ability to independently draw water from the river and augment the fire protection water supply and pressure.

In the event that fire suppression water is needed after firewater flow to the SFP is established, control room operators will assess the relative significance of these two needs and balance water flows as considered appropriate. Factors in this consideration will include the location, size, and potential significance of the fire to ensuring safe shutdown of the plant, and the amount of water remaining above fuel stored in the spent fuel pool and the rate of change in SFP water level.

Nuclear Safety and Radiological Release Performance Criteria:

An evaluation per 10 CFR 50.48, Fire Protection, Section 2.C(vii)(A) is provided below for each of the requested non-fire protection uses.

Fire protection water for screenwash function:

(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release

The ability to use fire protection water for the screenwash function does not impact the requirements of nuclear safety performance criteria as defined in NFPA 805 Section 1.5.1. This use will not directly result in compromising automatic fire suppression functions, manual fire suppression functions, or post-fire safe shutdown capability since water will automatically or manually be re-directed to the fire protection header upon fire protection header demand. The diesel driven fire pump will also auto start if needed as a result of a pressure drop in the fire header to provide for fire protection header demands. Therefore there is no impact to performance goals or objectives as described in LAR Attachment 2 (B-2 Table).

The ability to use fire protection water for the screenwash function has no impact on the radiological release performance criteria since water will be automatically or manually re-directed to the fire protection header upon fire protection header demand. The radiological release review addresses the release of firefighting water potentially containing radioactive materials and is not affected by the use of fire protection water to supplement the screenwash function. The ability to use fire protection water for the screenwash function does not change the radiological release evaluation and does not add additional radiological materials to the area or challenge system boundaries.

Fire protection water to maintain spent fuel pool inventory:

(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;

In the unlikely event requiring fire protection water for suppression system use and use as a non-fire protection related source for spent fuel pool inventory makeup, there are procedures in place and trained operators that will maintain the suppression system demand for firefighting. Therefore, nuclear safety performance criteria will not be impacted.

The ability to use fire protection water only after the other six water sources have been exhausted supports the radiological release performance criteria. Allowing the SFP water level to decrease could result in spent fuel being uncovered reducing spent fuel decay heat removal, and creating an extremely hazardous radiation environment. Therefore, in this worst case scenario the use of fire protection water for non-firefighting uses is justified to reduce the potential for radiological release.

Safety Margin and Defense-in-Depth:

An evaluation per 10 CFR 50.48, Fire Protection, Section 2.C(vii)(B) and Section 2.C(vii)(C) is provided below for each of the requested non-fire protection uses.

Fire protection water for screenwash function:

(B) Maintains safety margins

The ability to use fire protection water for the screenwash function does not change the safety margin since it has no impact on safety analysis acceptance criteria or on functional capabilities of equipment relied upon for safe and stable plant conditions. Due to the automatic and manual actions to re-direct fire protection water flow, and the ability of the DDFP to supply additional water to the fire header, the use of fire protection water for cleaning intake screens will not impact the ability of the fire protection header to deliver the system demand for fire suppression requirements.

(C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

The ability to use fire protection water for the screenwash function does not compromise the defense-in-depth elements, as follows:

- Fire prevention functions are maintained because controls such as combustible controls and hot work controls are not affected.
- Fire detection, automatic fire suppression, manual fire suppression, and mitigation functions are maintained because water will automatically or manually be re-directed to the fire protection header, and water from the DDFP will be supplied if needed, upon actuation of a fire suppression system in the event of a fire.
- Post-fire safe shutdown capability is maintained because fire barriers are maintained so that a fire will not spread and prevent operation of the equipment required to establish and maintain safe and stable plant conditions.

Fire protection water to maintain spent fuel pool inventory:

(B) Maintains safety margins:

In the unlikely event of a fire and fuel pool loss of inventory, there will be no initial impact to the safety margin of the fire protection water supply since there will be no immediate diversion of water to the spent fuel pool. There are multiple preferred borated and non-borated water sources that will be procedurally utilized prior to using fire suppression water for spent fuel pool inventory control. If the preferred sources of water are depleted (or unavailable) and fire protection water must be used to maintain spent fuel pool inventory, procedures in place and trained operators will maintain the suppression system demand until fire extinguishment.

(C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

The ability to use fire protection water for the SFP makeup function does not compromise the defense-in-depth elements, as follows:

- Fire prevention functions are maintained because controls such as combustible controls and hot work controls are not affected.
- Fire detection, automatic fire suppression, manual fire suppression, and mitigation functions are maintained because fire protection water will only be used for SFP makeup after all other sources of water are depleted or are otherwise unavailable, at which time fires will likely have been extinguished; also, by this time other sources of fire protection water, e.g., support from the Red Wing Fire Department, should also be available if needed in the event of a fire.
- Post-fire safe shutdown capability is maintained because fire barriers are maintained so that a fire will not spread and prevent operation of the equipment required to establish and maintain safe and stable plant conditions.

Operators would balance the need to supply firefighting water and maintain SFP level. The additional time provided by the six other water supplies would allow additional equipment and resources to arrive and support the fire protection water supply demand. There will be no impact to fire prevention, fire detection, automatic fire suppression functions, manual fire suppression functions, mitigation or post-fire safe shutdown capability.

Conclusion

The use of fire water to perform the screenwash function and to maintain spent fuel pool inventory does not meet the requirement or allowed exceptions of NFPA 805 Section 3.5.16. The evaluation determined that the performance-based approach utilized to evaluate a variance from the requirements of NFPA 805 Chapter 3:

(A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;

(B) Maintains safety margins; and

(C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

Therefore, the risk associated with the use of fire protection water to provide the screenwash function and maintain spent fuel pool inventory is low and PINGP can maintain safe and stable

conditions. NRC approval is being requested to permit the fire water system to be aligned for the screenwash function and to maintain spent fuel pool inventory.

Enclosure 4

LAR Attachment T (revised)

The following pages include a revision to

LAR Attachment T, Clarification of Prior NRC Approvals,

and includes information in response to SSA RAI 02.

T. Clarification of Prior NRC Approvals

2 Pages Attached

Introduction

The elements of the pre-transition fire protection program licensing basis for which specific NRC previous approval is uncertain are included in this attachment. Also included is sufficient detail to demonstrate how those elements of the pre-transition fire protection program licensing basis meet the requirements in 10 CFR 50.48(c) (RG 1.205, Revision 1, Regulatory Position 2.2.1).

Prior Approval Clarification Request 1 of 1: Operator Action to Isolate Power to PORV Control Circuits

Pre-transition Fire Protection Program Licensing Basis:

The Prairie Island Nuclear Generating Plant (PINGP) pre-transition licensing basis relative to the preclusion of spurious operation of pressurizer power operated relief valve (PORV) flow paths, for fires involving control room evacuation, included a previously approved exemption from the requirements of Section III.G.1 of Appendix R to 10 CFR 50. Specifically, the exemption allowed operators to close the Unit 1 and 2 PORV block valves prior to evacuating the control room, and then taking the follow-on action to remove control power fuses from the PORV control circuits for both units at their respective branch circuit panels.

The exemption was required because the removal of fuses involved the use of a fuse-pulling tool, which was considered to be a "repair action." This repair action was interpreted as a non-compliance to Section III.G.1 of Appendix R to 10 CFR 50 which requires, in part, that fire protection features shall be provided for structures, systems, and components important to safe shutdown so that one train of systems necessary to achieve and maintain hot shutdown conditions be free of fire damage.

This exemption was approved in a letter dated February 21, 1995. In 1999, PINGP performed a plant modification (99DC03), which included modification of the PORV control power supplies such that disconnect switches could be used in lieu of pulling control power fuses. The feasibility of utilizing the disconnect switches (no tool required) has been validated and has proven to be a beneficial change with respect to this activity.

Background/Basis:

NSP Exemption Request Letter, dated May 2, 1994

NSP requested an exemption from the requirements of Section III.G.2 of Appendix R to 10 CFR 50, to allow the manual removal of fuses from the PORV control circuits in the event of a fire, in lieu of modifying plant hardware. The reference to III.G.2 was later revised to III.G.1 during a follow-up phone call between NSP and NRR.

Issuance of Exemption Letter, dated February 21, 1995

The NRC issued an exemption from certain requirements of Appendix R to 10 CFR Part 50 to allow NSP to remove fuses from the PORV control circuits as a means of ensuring the reactor coolant system inventory in the event of a control room fire.

PINGP Plant Modification (99DC03)

Summary:
This modification relocated EQ circuit power supplies from harsh environments to mild environments. This modification repowered the Unit 1 and 2 PORV control circuits, from new distribution panels PNL 171, PNL 181, PNL 271, and PNL 281 respectively, which were, in turn, powered by upstream feeder distribution panels PNL 11, PNL 12, PNL 21, and PNL 22 respectively. An added benefit of this modification is that it allowed the PORV control circuits to be de-energized via disconnect switches in the feeder distribution panels, thus eliminating the need to pull control power fuses for fire events requiring control room evacuation.

Request

As part of this LAR submittal and transition to NFPA 805, it is requested that the NRC accept the following clarification of a prior NRC approval, with respect to the exemption granted to NSP on February 21, 1995:

> This operator action (recovery action) to preclude PORV opening remains a required action for PINGP under NFPA 805. The use of recovery actions is not allowed under the deterministic requirements of NFPA 805 Section 4.2.3.1. Clarification is requested to allow the previous exemption for operator actions to be extended to the NFPA 805 program.
>
> In addition, clarification is requested to extend the previous allowance to pull fuses to instead allow the operation of disconnect switches. The manual operation to open disconnect switches, demonstrated by PINGP to be feasible and reliable, is simpler than pulling fuses and therefore, for the purposes of this request, is requested to be deemed equivalent in intent and function.
>
> Clarification is also requested that the term "control room fire", as referred to in the exemption letter, applies to fires occurring in Fire Area 013 (Control Room) and Fire Area 018 (Relay Room). Under the pre-transition (Appendix R) program, both Fire Area 013 and Fire Area 018 were analyzed as one analysis area.