UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 24, 2013

Mr. Lawrence J. Weber
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c) (TAC NOS. ME6629 AND ME6630)

Dear Mr. Weber:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 322 to Renewed Facility Operating License No. DPR-58 and Amendment No. 305 to Renewed Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, respectively. The amendments consist of changes to the licenses in response to your application dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, May 1, 2013, June 21, 2013, and September 16, 2013.

The proposed amendments transition the fire protection program to a risk-informed, performance-based program based on National Fire Protection Association Standard 805 (NFPA 805), "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition, in accordance with Title 10 of the *Code of Federal Regulations*, Paragraph 50.48(c)(3). NFPA 805 allows the use of performance-based methods, such as fire modeling and fire risk evaluations, to demonstrate compliance with the nuclear safety performance criteria.

The amendments revise the fire protection license condition in each unit's license. As a result of placing the new license condition in each unit's license, the NRC is issuing additional license pages for each unit due to repagination of subsequent license pages. The only changes to the licenses are the changes to the fire protection license condition.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 322 to DPR-58
2. Amendment No. 305 to DPR-74
3. Safety Evaluation

cc w/encls: Distribution via ListServ

**ENCLOSURE 1**

AMENDMENT NO. 322

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-315

UNITED STATES
**NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 322
License No. DPR-58

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, May 1, 2013, June 21, 2013, and September 16, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended as indicated in the attachment to this license amendment, and paragraph 2.C(4) of Renewed Facility Operating License No. DPR-58 Condition 2.C(4) is amended to read as follows:

(4) Fire Protection Program

Indiana Michigan Power Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, May 1, 2013, June 21, 2013, and September 16, 2013, and as approved in the Safety Evaluation dated October 24, 2013. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed Fire PRA (FPRA) model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than $1x10^{-7}$/year (yr) for CDF and less than $1x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) Other Changes that May Be Made Without Prior NRC Approval

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program and Design Elements

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);

- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

   Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated October 24, 2013, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) <u>Transition License Conditions</u>

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2.C.(4)(c)2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2.C.(4)(b)2. above.

2. The licensee shall implement the modifications to its facility, as described in Enclosure 5, Attachment S, Table S-2, "Plant Modifications Committed," of I&M letter AEP-NRC-2013-75, dated September 16, 2013, to complete the transition to full compliance with 10 CFR 50.48(c) by October 24, 2014. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Enclosure 5, Attachment S, Table S-3, "Implementation Items," of I&M letter AEP-NRC-2013-75, dated September 16, 2013, by October 24, 2014.

4. The licensee shall complete an FPRA focused-scope peer review and resolve findings associated with the revised

FPRA LERF values, prior to self-approval of changes that result in more than a minimal increase in risk.

5. The licensee shall complete a focused-scope peer review and resolve findings of the PRA upgrade related to reduced mission times for cutsets containing a test and maintenance event combined with a running failure, prior to self-approval of changes that result in more than a minimal increase in risk.

3. This license amendment is effective as of its date of issuance and shall be implemented by October 24, 2014.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert D. Carlson, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-58

Date of Issuance: October 24, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 322

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Renewed Facility Operating License No. DPR-58 with the attached revised pages. The changed areas are identified by a marginal line.

| REMOVE | INSERT |
|---|---|
| 3 | 3 |
| -- | 4 |
| -- | 5 |
| 4 | 6 |
| 4A | 7 |
| 4B | 8 |
| 5 | 9 |
| 6 | 10 |

and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not to exceed 3304 megawatts thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 321, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Less than Four Loop Operation

The licensee shall not operate the reactor at power levels above P-7 (as defined in Table 3.3.1-1 of Specification 3.3.1 of Appendix A to this renewed operating license) with less than four reactor coolant loops in operation until (a) safety analyses for less than four loop operation have been submitted, and (b) approval for less than four loop operation at power levels above P-7 has been granted by the Commission by amendment of this license.

(4) Fire Protection Program

Indiana Michigan Power Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, May 1, 2013, June 21, 2013, and September 16, 2013, and as approved in the Safety Evaluation dated October 24, 2013. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) <u>Risk-Informed Changes that May Be Made Without Prior NRC Approval</u>

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed Fire PRA (FPRA) model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than $1x10^{-7}$/year (yr) for CDF and less than $1x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) <u>Other Changes that May Be Made Without Prior NRC Approval</u>

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program and Design Elements

   Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

   Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated October 24, 2013, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) <u>Transition License Conditions</u>

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2.C.(4)(c)2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2.C.(4)(b)2. above.

2. The licensee shall implement the modifications to its facility, as described in Enclosure 5, Attachment S, Table S-2, "Plant Modifications Committed," of I&M letter AEP-NRC-2013-75, dated September 16, 2013, to complete the transition to full compliance with 10 CFR 50.48(c) by October 24, 2014. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Enclosure 5, Attachment S, Table S-3, "Implementation Items," of I&M letter AEP-NRC-2013-75, dated September 16, 2013, by October 24, 2014.

4. The licensee shall complete an FPRA focused scope peer review and resolve findings associated with the revised FPRA LERF values, prior to self-approval of changes that result in more than a minimal increase in risk.

5. The licensee shall complete a focused scope peer review and resolve findings of the PRA upgrade related to reduced mission times for cutsets containing a test and maintenance event combined with a running failure, prior to self-approval of changes that result in more than a minimal increase in risk.

(5) Deleted by Amendment No. 279

(6) Deleted by Amendment No. 80

(7) Deleted by Amendment No. 287

(8) Deleted by Amendment No. 279

(9) Deleted by Amendment No. 279

(10) Deleted by Amendment No. 279

(11) Deleted by Amendment No. 279

(12) The 72 hour allowed outage time of Technical Specifications 3.1.2.4 and 3.5.2, Action "a," which was entered at 0130 on January 13, 2005, may be extended by an additional 24 hours to complete repair and testing of the 1 West Centrifugal Charging Pump.

(13) The 72 hour allowed outage time of Technical Specifications 3.8.1.1 Action "a" may be extended to 14 days one time for the 69 kilovolt (alternate) independent offsite power circuit when it is made inoperable to complete connection of the Supplemental Diesel Generators to the existing plant electrical system and to perform upgrades to the alternate offsite power supply circuit.

(14) Implementation of Amendment No. 287

This amendment authorizes the relocation of certain current Technical Specification requirements and operating license conditions to other licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the other documents, as described in (1) Section 5.0 of the NRC staff's Safety Evaluation and (2) Table LA of Removed Details and Table R of Relocated Specifications attached to the NRC staff's Safety Evaluation, which is enclosed with this amendment.

The schedule for the performance of new and revised Surveillance Requirements (SRs) shall be as follows:

For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval, which begins on the date of implementation of this amendment.

For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.

For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment, except as noted below for SRs that have modified acceptance criteria as a result of revised allowable values.

For SRs that have modified acceptance criteria as a result of revised allowable values, the current allowable Values and current channel calibration frequencies are required to be met until the trip setpoints are changed to reflect the new allowable values and channel calibration frequencies. The trip setpoints are required to be changed no later than the unit startup after the first planned outage of sufficient duration to change all of the trip setpoints for the unit following implementation of this amendment.

For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to implementation of this amendment, except as noted above for SRs that have modified acceptance criteria as a result of revised allowable values.

(15) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protecton and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitgation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(16) The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

(17) Ice Condenser Ice Fusion Time Requirement

The licensee is authorized to change the Updated Final Safety Analysis Report (UFSAR) to allow inspection of each ice condenser within 24 hours of experiencing a seismic event greater than or equal to an operating-basis earthquake within the 5-week period after ice basket replenishment has been completed to confirm that adverse ice fallout has not occurred which could impede the ability of the ice condenser lower inlet doors to open. This action would be taken, in lieu of requiring a 5-week waiting period following ice basket replenishment, prior to beginning ascension to power operations, as set forth in the application for amendment dated February 29, 2008, and evaluated in the safety evaluation accompanying Amendment No. 303. The licensee shall update the UFSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

(18) Upon implementation of Amendment No. 307 adopting TSTF-448, Revision 3, the determination of CRE unfiltered air inleakage as required by SR 3.7.10.4, in accordance with TS 5.5.16.c.(i), the assessment of CRE habitability as required by TS 5.5.16.c.(ii), and the measurement of CRE pressure as required by TS 5.5.16.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.10.4, in accordance with TS 5.5.16.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from June 1999, the date of the most recent successful tracer gas test, as stated in the December 4, 2003, letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, TS 5.5.16.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from June 1999, the date of the most recent successful tracer gas test, as stated in the December 4, 2003, letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, TS 5.5.16.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from the date of the most recent successful pressure measurement test, or within 182 days if not performed previously.

D. Physical Protection

The Indiana Michigan Power Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revision to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Donald C. Cook Nuclear Plant Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 1," submitted by letter dated May 10, 2006.

The Indiana and Michigan Power Company shall fully implement and maintain in effect all provisions of the Commission-approved Donald C. Cook Nuclear Plant Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Donald C. Cook Nuclear Plant CSP was approved by License Amendment No. 315 as supplemented by a change approved by License Amendment No. 319.

E. Deleted by Amendment No. 80

F. Deleted by Amendment No. 80

G. In all places of this renewed operating license, the reference to the Indiana and Michigan Electric Company is amended to read Indiana Michigan Power Company.

H. Deleted by Amendment No. 287

I. Deleted by Amendment No. 287

J. The licensee is authorized to use digital signal processing instrumentation in the reactor protection system.

K. Updated Final Safety Analysis Report

The Indiana Michigan Power Company Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to the period of extended operation. The Indiana Michigan Power Company shall complete these activities no later than October 25, 2014, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement, as revised, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed operating license. Until that update is complete, Indiana Michigan Power Company may make changes to the programs and activities described in the supplement without prior Commission

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan

approval, provided that Indiana Michigan Power Company evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

L. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. .

3. This renewed operating license is effective as of the date of issuance and shall expire at midnight, October 25, 2034.

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A – Technical Specifications
2. Appendix B – Environmental Technical Specifications

Date of Issuance: August 30, 2005

**ENCLOSURE 2**

AMENDMENT NO. 305

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-316

**UNITED STATES**
**NUCLEAR REGULATORY COMMISSION**
**WASHINGTON, D.C. 20555-0001**

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 305
License No. DPR-74

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, May 1, 2013, June 21, 2013, and September 16, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended as indicated in the attachment to this license amendment, and paragraph 2.C.(3)(o) of Renewed Facility Operating License No. DPR-74 is amended to read as follows:

(o) <u>Fire Protection Program</u>

Indiana Michigan Power Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, May 1, 2013, June 21, 2013, and September 16, 2013, and as approved in the Safety Evaluation dated October 24, 2013. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

I. <u>Risk-Informed Changes that May Be Made Without Prior NRC Approval</u>

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed Fire PRA (FPRA) model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than $1x10^{-7}$/year (yr) for CDF and less than $1x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

II. <u>Other Changes that May Be Made Without Prior NRC Approval</u>

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program and Design Elements

   Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

   The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

   - "Fire Alarm and Detection Systems" (Section 3.8);
   - "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);

- “Gaseous Fire Suppression Systems” (Section 3.10); and,
- “Passive Fire Protection Features” (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

   Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated October 24, 2013, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

III. Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2.C.(3)(o)(III)2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2.C.(3)(o)(II)2. above.

2. The licensee shall implement the modifications to its facility, as described in Enclosure 5, Attachment S, Table S-2, “Plant Modifications Committed,” of I&M letter AEP-NRC-2013-75, dated September 16, 2013, to complete the transition to full compliance with 10 CFR 50.48(c) by October 24, 2014. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Enclosure 5, Attachment S, Table S-3, “Implementation Items,” of I&M letter AEP-NRC-2013-75, dated September 16, 2013, by October 24, 2014.

4. The licensee shall complete an FPRA focused-scope peer review and resolve findings associated with the revised

FPRA LERF values, prior to self-approval of changes that result in more than a minimal increase in risk.

5. The licensee shall complete a focused-scope peer review and resolve findings of the PRA upgrade related to reduced mission times for cutsets containing a test and maintenance event combined with a running failure, prior to self-approval of changes that result in more than a minimal increase in risk.

3. This license amendment is effective as of its date of issuance and shall be implemented by October 24, 2014.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert D. Carlson, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-74

Date of Issuance: October 24, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 305

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Renewed Facility Operating License No. DPR-74 with the attached revised pages. The changed areas are identified by a marginal line.

| REMOVE | INSERT |
|---|---|
| 4 | 4 |
| -- | 5 |
| -- | 6 |
| 5 | 7 |
| 5A | 8 |
| 5B | 9 |
| 6 | 10 |
| 7 | 11 |

residual heat removal, safety injection and boron injection systems in accordance with the specifications of Section XI of the American Society of Mechanical Engineers Code. In addition, prior to completion of the first inservice testing interval, test connections which allow individual leak testing of the charging pump system discharge check valves shall be installed and the check valves shall be leak tested. The tests shall be repeated at the conclusion of each subsequent inservice inspection interval.

(d) Deleted by Amendment No. 39

(e) Deleted by Amendment No. 5

(f) Deleted by Amendment No. 2

(g) Deleted by Amendment No. 60

(h) Deleted by Amendment No. 63

(i) Deleted by Amendment No. 19

(j) <u>Power Operation with Fewer than Four Reactor Coolant Pumps in Operation</u>

Indiana Michigan Power Company shall not operate the reactor at power levels above P-7 (as defined in Table 3.3.1-1 of Specification 3.3.1 of Appendix A to this renewed operating license) with fewer than four reactor coolant loops in operation until safety analyses for fewer than four loop operation have been submitted and approval for fewer than four loop operation at power levels above P-7 has been granted by the Commission by Amendment of this license.

(k) Deleted by Amendment No. 16

(l) Deleted by Amendment No. 63

(m) Deleted by Amendment No. 19

(n) Deleted by Amendment No. 28

(o) <u>Fire Protection Program</u>

Indiana Michigan Power Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, May 1, 2013, June 21, 2013, and September 16, 2013, and as approved in the Safety Evaluation dated October 24, 2013. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

I. <u>Risk-Informed Changes that May Be Made Without Prior NRC Approval</u>

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed Fire PRA (FPRA) model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1x10-7/year (yr) for CDF and less than 1x10-8/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

II. <u>Other Changes that May Be Made Without Prior NRC Approval</u>

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program and Design Elements

   Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

   Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated October 24, 2013, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

III. <u>Transition License Conditions</u>

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2.C.(3)(o)(III)2. below, risk- informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2.C.(3)(o)(II)2. above.

2. The licensee shall implement the modifications to its facility, as described in Enclosure 5, Attachment S, Table S-2, "Plant Modifications Committed," of I&M letter AEP-NRC-2013-75, dated September 16, 2013, to complete the transition to full compliance with 10 CFR 50.48(c) by October 24, 2014. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Enclosure 5, Attachment S, Table S-3, "Implementation Items," of I&M letter AEP-NRC-2013-75, dated September 16, 2013, by October 24, 2014.

4. The licensee shall complete an FPRA focused scope peer review and resolve findings associated with the revised FPRA LERF values, prior to self-approval of changes that result in more than a minimal increase in risk.

5. The licensee shall complete a focused scope peer review and resolve findings of the PRA upgrade related to reduced mission times for cutsets containing a test and maintenance event combined with a running failure, prior to self-approval of changes that result in more than a minimal increase in risk.

(p) Deleted by Amendment No. 121

(q) Deleted by Amendment No. 2

(r) Deleted by Amendment No. 68

(s) Deleted by Amendment No. 261

(t) Deleted by Amendment No. 63

(u) Deleted by Amendment No. 261

(v) Deleted by Amendment No. 269

(w) Deleted by Amendment No 261

(x) Deleted by Amendment No. 261

(y) Deleted by Amendment No. 261

(z) The 72 hour allowed outage time of Technical Specifications 3.8.1.1 Action "b". which was entered at 0923, on December 7, 2003, may be extended one time by an additional 72 hours to complete repair and testing of the 2 AB diesel generator.

(aa) The 72 hour allowed outage time of Technical Specifications 3.8.1.1 Action "a" may be extended to 14 days one time for the 69 kilovolt (alternate) independent offsite power circuit when it is made inoperable to complete connection of the Supplemental Diesel Generators to the existing plant electrical system and to perform upgrades to the alternate offsite power supply circuit.

(bb) Implementation of Amendment No. 269

This amendment authorizes the relocation of certain current Technical Specification requirements and operating license conditions to other licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the other documents, as described in (1) Section 5.0 of the NRC staff's Safety Evaluation and (2) Table LA of Removed Details and Table R of Relocated Specifications attached to the NRC staff's Safety Evaluation, which is enclosed with this amendment.

The schedule for the performance of new and revised Surveillance Requirements (SRs) shall be as follows:

For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval, which begins on the date of implementation of this amendment.

For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.

For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment, except as noted below for SRs that have modified acceptance criteria as a result of revised allowable values.

For SRs that have modified acceptance criteria as a result of revised allowable values, the current allowable Values and current channel calibration frequencies are required to be met until the trip setpoints are changed to reflect the new allowable values and channel calibration frequencies. The trip setpoints are required to be changed no later than the unit startup after the first planned outage of sufficient duration to change all of the trip setpoints for the unit following implementation of this amendment.

For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to implementation of this amendment, except as noted above for SRs that have modified acceptance criteria as a result of revised allowable values.

(cc) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(I) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training and response personnel

(II) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment

6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(III) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(dd) The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

(ee) Ice Condenser Ice Fusion Time Requirement

The licensee is authorized to change the Updated Final Safety Analysis Report (UFSAR) to allow inspection of each ice condenser within 24 hours of experiencing a seismic event greater than or equal to an operating-basis earthquake within the 5-week period after ice basket replenishment has been completed to confirm that adverse ice fallout has not occurred which could impede the ability of the ice condenser lower inlet doors to open. This action would be taken, in lieu of requiring a 5-week waiting period following ice basket replenishment, prior to beginning ascension to power operations, as set forth in the application for amendment dated February 29, 2008, and evaluated in the safety evaluation accompanying Amendment No. 286. The licensee shall update the UFSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

(ff) Upon implementation of Amendment No. 289 adopting TSTF-448, Revision 3, the determination of CRE unfiltered air inleakage as required by SR 3.7.10.4, in accordance with TS 5.5.16.c.(i), the assessment of CRE habitability as required by TS 5.5.16.c.(ii), and the measurement of CRE pressure as required by TS 5.5.16.d, shall be considered met. Following implementation:

(I) The first performance of SR 3.7.10.4, in accordance with TS 5.5.16.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from June 1999, the date of the most recent successful tracer gas test, as stated in the December 4, 2003, letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(II) The first performance of the periodic assessment of CRE habitability, TS 5.5.16.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from June 1999, the date of the most recent successful tracer gas test, as stated in the December 4, 2003, letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(III) The first performance of the periodic measurement of CRE pressure, TS 5.5.16.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from the date of the most recent successful pressure measurement test, or within 182 days if not performed previously.

D. Physical Protection

The Indiana Michigan Power Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Donald C. Cook Nuclear Plant Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 1," submitted by letter dated May 10, 2006.

The Indiana and Michigan Power Company shall fully implement and maintain in effect all provisions of the Commission-approved Donald C. Cook Nuclear Plant Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Donald C. Cook Nuclear Plant CSP was approved by License Amendment No. 299 as supplemented by a change approved by License Amendment No. 303.

E. Deleted by Amendment No. 63

F. In all places of this renewed operating license, the reference to the Indiana and Michigan Electric Company is amended to read Indiana Michigan Power Company.

G. Deleted by Amendment No. 269

H. Deleted by Amendment No. 269

I. Deleted by Amendment No. 261

(1) Deleted by Amendment No. 261

(2) Deleted by Amendment No. 261

J. The licensee is authorized to use digital signal processing instrumentation in the reactor protection system.

K. Updated Final Safety Analysis Report

The Indiana Michigan Power Company Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to the period of extended operation. The Indiana Michigan Power Company shall complete these activities no later than December 23, 2017, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

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[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

The Updated Final Safety Analysis Report supplement, as revised, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed operating license. Until that update is complete, Indiana Michigan Power Company may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Indiana Michigan Power Company evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

L. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion.

3. This renewed operating license is effective as of the date of issuance and shall expire at midnight, December 23, 2037.

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Preoperational Tests, Startup Tests and Other Items Which Must Be Completed Prior to Proceeding to Succeeding Operational Modes.
2. Appendix A – Technical Specifications
3. Appendix B – Environmental Technical Specifications

Date of Issuance: August 30, 2005

**ENCLOSURE 3**

SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED

FIRE PROTECTION PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c)

AMENDMENT NOS. 322 AND 305 TO RENEWED FACILITY OPERATING

LICENSE NOS. DPR-58 AND DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

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SAFETY EVALUATION

**ATTACHMENTS**

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED

FIRE PROTECTION PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c)

AMENDMENT NOS. 322 AND 305 TO RENEWED FACILITY OPERATING

LICENSE NOS. DPR-58 AND DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

# 1.0 INTRODUCTION

## 1.1 Background

The U.S. Nuclear Regulatory Commission (NRC) started developing fire protection requirements in the 1970s and, in 1976, the NRC published comprehensive fire protection guidelines. Subsequently, the NRC performed fire protection reviews for the operating reactors, and documented the results in safety evaluation reports (SERs) or supplements to SERs. In 1980, to resolve issues identified in those reports, the NRC amended its regulations for fire protection in operating nuclear power plants and published its Final Rule, Fire Protection Program for Operating Nuclear Power Plants (45 FR 76602; November 19, 1980) (adding Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.48, "Fire protection" and Appendix R to 10 CFR Part 50 "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979). Paragraph 50.48(a)(1) requires each operating nuclear power plant to have a fire protection plan that satisfies General Design Criterion (GDC) 3 of Appendix A to 10 CFR 50 and states that the fire protection plan must describe the overall fire protection program; identify the positions responsible for the program and the authority delegated to those positions; outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage. Paragraph 50.48(a)(2) states that the fire protection plan must describe the specific features necessary to implement the program described in paragraph (a)(1) including administrative controls and personnel requirements; automatic and manual fire detection and suppression systems; and the means to limit fire damage to structures, systems, and components (SSCs) to ensure the capability to safely shut down the plant. Paragraph 50.48(a)(3) requires that the licensee retain the fire protection plan and each change to the plan as a record until the Commission terminates the license.

In the 1990s, the NRC worked with the National Fire Protection Association (NFPA) and industry to develop a risk-informed (RI), performance-based (PB) consensus standard for fire protection. In 2001, the NFPA Standards Council issued NFPA 805, which describes a methodology for establishing fundamental fire protection program (FPP) design requirements and elements, determining required fire protection systems and features, applying performance-based requirements, and administering fire protection for existing light-water reactors during operation, decommissioning, and permanent shutdown. It provides for the establishment of a minimum set of fire protection requirements but allows performance-based or deterministic approaches to be used to meet performance criteria.

Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1 (RG 1.205) (Reference 1), states, in part, that:

> On March 26, 1988, the staff sent to the Commission SECY-98-058, "Development of a Risk-Informed, Performance-Based Regulation for Fire Protection at Nuclear Power Plants," dated March 26, 1998 [Reference 2], in which it proposed to work with NFPA and the industry to develop a risk-informed, performance-based consensus standard for nuclear power plant fire protection. This consensus standard could be endorsed in a future rulemaking as an alternative set of fire protection requirements to the existing regulations in 10 CFR 50.48. In SECY-00-0009, "Rulemaking Plan, Reactor Fire Protection Risk-Informed, Performance-Based Rulemaking," dated January 13, 2000 [Reference 3], the NRC staff requested and received Commission approval to proceed with a rulemaking to permit reactor licensees to adopt NFPA 805 as an alternative to existing fire protection requirements. On February 9, 2001, the NFPA Standards Council approved the 2001 edition of NFPA 805, ["Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" (Reference 4)] as an American National Standard for performance-based fire protection for light-water nuclear power plants.

An adoptee of NFPA 805 must meet the performance goals, objectives, and criteria that are itemized in Chapter 1 of NFPA 805 through the implementation of performance-based or deterministic approaches. The goals include ensuring that reactivity control, inventory and pressure control, decay heat removal, vital auxiliaries, and process monitoring are achieved and maintained. The adoptee then must establishes plant fire protection requirements using the methodology in Chapter 2 of NFPA 805 such that the minimum FPP elements and design criteria contained in Chapter 3 of NFPA 805 are satisfied. Next, an adoptee identifies fire areas and fire hazards though a plant-wide analysis, and then applies either a performance-based or a deterministic approach to meet the performance criteria. As part of a performance-based approach, an adoptee will use engineering evaluations, probabilistic safety assessments, and fire modeling calculations to show that the criteria are met. Chapter 4 of NFPA establishes the methodology to determine the fire protection systems and features required to achieve the performance criteria. It also specifies that at least one success path to achieve the nuclear safety performance criteria (NSPC) shall be maintained free of fire damage by a single fire.

RG 1.205 also states, in part, that:

> Effective July 16, 2004, the Commission amended its fire protection requirements in 10 CFR 50.48 to add 10 CFR 50.48(c), which incorporates by reference the 2001 edition of NFPA 805, with certain exceptions, and allows licensees to apply for a license amendment to comply with the 2001 edition of NFPA 805 (69 FR 33536). NFPA has issued subsequent editions of NFPA 805, but the regulation does not endorse them.

Throughout this safety evaluation (SE), where the NRC staff states that the licensee's FPP element is in compliance with (or meeting the requirements of) NFPA 805, the NRC staff is referring to NFPA 805 with the exceptions, modifications, and supplementation described in 10 CFR 50.48(c)(2).

RG 1.205 also states, in part, that:

> In parallel with the Commission's efforts to issue a rule incorporating the risk-informed, performance-based fire protection provisions of NFPA 805, [the Nuclear Energy Institute (NEI)] published implementing guidance for the specific provisions of NFPA 805 and 10 CFR 50.48(c) in NEI 04-02.

RG 1.205 provides the NRC staff's position on NEI 04-02, Revision 2, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," April 2008 (NEI 04-02) (Reference 5), and offers additional information and guidance to supplement the NEI document and assist licensees in meeting the NRC's regulations in 10 CFR 50.48(c) related to adopting an RI/PB FPP.

Accordingly, by letter dated July 1, 2011 (Reference 6), Indiana Michigan Power Company (I&M or the licensee), submitted a license amendment request (LAR) to allow the licensee to maintain the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, fire protection program in accordance with 10 CFR 50.48(c).

## 1.2 Requested Licensing Action

By application sent to the U.S. NRC dated July 1, 2011 (Reference 6), as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, May 1, 2013, June 21, 2013, and September 16, 2013 (References 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, and 17, respectively) I&M submitted an application for a license amendment to transition the CNP FPP from 10 CFR 50.48(b) to 10 CFR 50.48(c), NFPA 805, "Performance-Based Standard for Fire Protection For Light Water Reactor Electric Generating Plants," 2001 Edition. The supplements provided additional information that clarified the application, but did not expand the overall scope of the application as originally noticed, and did not change the NRC staff's original proposed opportunity for a hearing on the initial application as published in the *Federal Register* (FR) on October 4, 2011 (76 FR 61396). The June 29, 2013, letter in its entirety, and portions of the letters dated July 1, 2011, April 27, 2012, August 9, 2012, October 15, 2012, November 9, 2012, and May 1, 2013, contain sensitive security-related information and, accordingly, have been withheld from public disclosure.

The licensee requested an amendment to the CNP renewed operating licenses in order to establish and maintain an RI/PB FPP in accordance with the requirements of 10 CFR 50.48(c).

Specifically, the licensee requested to transition from the existing deterministic fire protection licensing basis established in accordance with Final Safety Analysis Report and as approved in the SERs dated December 12, 1977 (Reference 18), July 31, 1979 (Reference 19), January 30, 1981 (Reference 20), February 7, 1983 (Reference 21), November 22, 1983 (Reference 22), December 23, 1983 (Reference 23), March 16, 1984 (Reference 24), August 27, 1985 (Reference 25), June 30, 1986 (Reference 26), January 28, 1987 (Reference 27), May 26, 1987 (Reference 28), June 16, 1988 (Reference 29), June 17, 1988 (Reference 30), June 7, 1989 (Reference 31), February 1, 1990[1] (Reference 32), February 9, 1990 (Reference 33), March 26, 1990 (Reference 34), April 26, 1990 (Reference 35), March 31, 1993 (Reference 36), April 8, 1993 (Reference 37), December 14, 1994 (Reference 38), January 24, 1995 (Reference 39), April 19, 1995 (Reference 40), June 8, 1995 (Reference 41), and March 11, 1996 (Reference 42), to a performance-based FPP in accordance with 10 CFR 50.48(c), that uses risk information, in part, to demonstrate compliance with the fire protection and nuclear safety goals, objectives, and performance criteria of NFPA 805. As such, the proposed FPP at CNP is referred to as RI/PB throughout this SE.

In its LAR, the licensee has provided a description of the revised FPP for which it is requesting NRC approval to implement, a description of the FPP that it will implement under 10 CFR 50.48(a) and (c), and the results of the evaluations and analyses required by NFPA 805.

This SE documents the NRC staff's evaluation of the licensee's LAR and the NRC staff's conclusion that:

(1) The licensee has identified any orders and license conditions that must be revised or superseded, and has provided the necessary revisions to the plant's TSs and bases, as required by 10 CFR 50.48(c)(3)(i).

(2) The licensee has completed its implementation of the methodology in Chapter 2, "Methodology," of NFPA 805 (including all required evaluations and analyses), and the NRC staff has approved the licensee's modified fire protection plan, which reflects the decision to comply with NFPA 805, as required by 10 CFR 50.48(a).

(3) The licensee will modify its FPP, as described in the LAR, in accordance with the implementation schedule set forth in this SE and the accompanying license condition, as required by 10 CFR 50.48(c)(3)(ii).

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1 The Renewed Facility Operating Licenses reflect a safety evaluation date of February 1, 1990; however, the associated Amendment Nos. 130 and 115 related to changes to the fire protection technical specifications were issued with a safety evaluation dated February 8, 1990.

The licensee proposed a new fire protection license condition reflecting the new RI/PB FPP licensing basis. Section 2.4.2 and Section 4.0 of this SE discuss the license condition in detail.

## 2.0 REGULATORY EVALUATION

Section 50.48, "Fire Protection," of 10 CFR provides the NRC requirements for nuclear power plant fire protection.

The NRC regulations include specific requirements for requesting approval for an RI/PB FPP based on the provisions of NFPA 805 (Reference 4). Paragraph 50.48(c)(3)(i) of 10 CFR states, in part, that:

> A licensee may maintain a fire protection program that complies with NFPA 805 as an alternative to complying with paragraph (b) of this section for plants licensed to operate before January 1, 1979, or the fire protection license conditions for plants licensed to operate after January 1, 1979. The licensee shall submit a request to comply with NFPA 805 in the form of an application for license amendment under § 50.90. The application must identify any orders and license conditions that must be revised or superseded, and contain any necessary revisions to the plant's technical specifications and the bases thereof.

In addition, 10 CFR 50.48(c)(3)(ii) states that:

> The licensee shall complete its implementation of the methodology in Chapter 2 of NFPA 805 (including all required evaluations and analyses) and, upon completion, modify the fire protection plan required by paragraph (a) of this section to reflect the licensee's decision to comply with NFPA 805, before changing its fire protection program or nuclear power plant as permitted by NFPA 805.

The intent of 10 CFR 50.48(c)(3)(ii) is given in the statement of considerations for the Final Rule, Voluntary Fire Protection Requirements for Light Water Reactors; Adoption of NFPA 805 as a Risk-Informed, Performance-Based Alternative (69 FR 33536; June 16, 2004), which states, in part, that:

> This paragraph requires licensees to complete all of the Chapter 2 methodology (including evaluations and analyses) and to modify their fire protection plan before making changes to the fire protection program or to the plant configuration. This process ensures that the transition to an NFPA 805 configuration is conducted in a complete, controlled, integrated, and organized manner. This requirement also precludes licensees from implementing NFPA 805 on a partial or selective basis (*e.g.*, in some fire areas and not others, or truncating the methodology within a given fire area).

As stated in 10 CFR 50.48(c)(3)(i), the Director of the Office of Nuclear Reactor Regulation (NRR), or a designee of the Director, may approve the application if the Director or designee determines that the licensee has identified orders, license conditions, and the technical specifications that must be revised or superseded, and that any necessary revisions are adequate.

The regulations also allow for flexibility that was not included in the NFPA 805 standard. Licensees who choose to adopt 10 CFR 50.48(c), but wish to use the PB methods permitted elsewhere in the standard to meet the fire protection requirements of NFPA 805 Chapter 3, "Fundamental Fire Protection Program and Design Elements," may do so by submitting an LAR in accordance with 10 CFR 50.48(c)(2)(vii).

> The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the performance-based approach;
>
> (A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
>
> (B) Maintains safety margins; and
>
> (C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

Alternatively, licensees may choose to use RI or PB alternatives to comply with NFPA 805 by submitting an LAR in accordance with 10 CFR 50.48(c)(4).

> The Director of the Office of Nuclear Reactor Regulation, or designee of the Director, may approve the application if the Director or designee determines that the proposed alternatives:
>
> (i) Satisfy the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
>
> (ii) Maintain safety margins; and
>
> (iii) Maintain fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

In addition to the conditions outlined by the rule that require licensees to submit an LAR for NRC review and approval in order to adopt an RI/PB FPP, a licensee may also submit additional elements of its FPP for which it wishes to receive specific NRC review and approval, as set forth in Regulatory Position C.2.2.1 of RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1 (Reference 1). Inclusion of these elements in the NFPA 805 LAR is meant to alleviate uncertainty in portions of the current FPP licensing bases as a result of the lack of specific NRC approval of these elements. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission. Accordingly, any submittal addressing these additional FPP elements needs to include sufficient detail to allow the NRC staff to assess whether the licensee's treatment of these elements meets the 10 CFR 50.48(c) requirements.

The purpose of the FPP established by NFPA 805 is to provide assurance, through a defense-in-depth (DID) philosophy, that the NRC's fire protection objectives are satisfied. NFPA 805 Section 1.2, "Defense-in-Depth," states the following:

> Protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations is paramount to this standard. The fire protection standard shall be based on the concept of defense-in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:
>
> (1) Preventing fires from starting
>
> (2) Rapidly detecting 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 addition, in accordance with GDC 3, "Fire protection," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, fire protection systems must be designed such that their failure or inadvertent operation does not significantly impair the ability of the SSCs important to safety to perform their intended safety functions.

## 2.1 <u>Applicable Regulations</u>

The following regulations address fire protection:

- GDC 3, "Fire protection," to 10 CFR Part 50, Appendix A:

  Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

- GDC 5, "Sharing of structures, systems, and components," to 10 CFR Part 50, Appendix A, relates to shared fire protection systems, and potential fire impacts on shared SSCs important to safety.

- 10 CFR 50.48(a)(1), requires that each holder of an operating license have a fire protection plan that satisfies General Design Criterion 3 of Appendix A to 10 CFR Part 50.

- 10 CFR 50.48(c), incorporates NFPA 805 (2001 Edition) by reference, with certain exceptions, modifications, and supplementation.. This regulation establishes the requirements for using an RI/PB FPP in conformance with NFPA 805 as an alternative to the requirements associated with 10 CFR 50.48(b) and Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50, or the specific plant fire protection license condition.

- 10 CFR Part 20, "Standards for Protection Against Radiation," establishes the radiation protection limits used as NFPA 805 radioactive release performance criteria, as specified in NFPA 805, Section 1.5.2, "Radioactive Release Performance Criteria."

## 2.2 Applicable Staff Guidance

The NRC staff review also relied on the following additional codes, RGs, and standards:

- RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, issued December 2009 (Reference 1), provides guidance for use in complying with the requirements that the NRC has promulgated for RI/FP FPPs that comply with 10 CFR 50.48 and the referenced 2001 Edition of the NFPA standard. It endorses portions of NEI 04-02, Revision 2, where it has been found to provide methods acceptable to the NRC for implementing NFPA 805 and complying with 10 CFR 50.48(c). The regulatory positions in Section C of RG 1.205 include clarification of the guidance provided in NEI 04-02, as well as NRC exceptions to the guidance. RG 1.205 sets forth regulatory positions, emphasizes certain issues, clarifies the requirements of 10 CFR 50.48(c) and NFPA 805, clarifies the guidance in NEI 04-02, and modifies the NEI 04-02 guidance where required. Should a conflict occur between NEI 04-02 and this RG, the regulatory positions in RG 1.205 govern.

- The 2001 edition of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," which specifies the minimum fire protection requirements for existing light-water nuclear power plants during all phases of plant operations, including shutdown, degraded conditions, and decommissioning. NFPA 805 was developed to provide a comprehensive risk-informed, performance-based standard for fire protection. The NFPA 805 Technical Committee on Nuclear Facilities is composed of nuclear plant licensees, the NRC, insurers, equipment manufacturers, and subject matter experts. The standard was developed in accordance with NFPA processes, and consisted of a number of technical meetings and reviews of draft documents by committee and industry representatives. The scope of NFPA 805 includes goals related to nuclear safety, radioactive release, life safety, and plant damage/business interruption. The standard addresses fire protection requirements for nuclear plants during all plant operating modes and conditions, including shutdown and decommissioning, which had not been explicitly addressed by previous requirements and guidelines. NFPA 805 became effective on February 9, 2001.

- NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," which provides guidance for implementing the requirements of 10 CFR 50.49(c), and represents methods for implementing, in whole or in part, a risk-informed, performance-based FPP. This implementing guidance for NFPA 805 has two primary purposes: (1) provide direction and clarification for adopting NFPA 805 as an acceptable approach to fire protection, consistent with 10 CFR 50.48 (c); and (2) provide additional supplemental technical guidance and methods for using NFPA 805 and its appendices to demonstrate compliance with fire protection requirements. Although there is a significant amount of detail in NFPA 805 and its appendices, clarification and additional guidance for select issues help ensure consistency and effective utilization of the standard. The NEI 04-02 guidance focuses attention on the risk-informed, performance-based fire protection goals, objectives, and performance criteria contained in NFPA 805 and the risk-informed, performance-based tools considered acceptable for demonstrating compliance. Revision 2 of NEI 04-02 incorporates guidance from RG 1.205 and approved Frequently Asked Questions (FAQs).

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, issued May 2011 (Reference 43), which provides the NRC staff's recommendations for using risk information in support of licensee-initiated licensing basis (LB) changes to a nuclear power plant that require such review and approval. The guidance provided does not preclude other approaches for requesting LB changes. Rather, RG 1.174 is intended to improve consistency in regulatory decisions in areas in which the results of risk analyses are used to help justify regulatory action. As such, the RG provides general guidance concerning one approach that the NRC has determined to be acceptable for analyzing issues associated with proposed changes to a plant's LB and for assessing the impact of such proposed changes on the risk associated with plant design and operation.

- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, issued March 2009 (Reference 44), which provides guidance to licensees for use in determining the technical adequacy of the base probabilistic risk assessment (PRA) used in a risk-informed regulatory activity, and endorses standards and industry peer review guidance. The RG provides guidance in four areas:

  (1) a definition of a technically acceptable PRA

  (2) the NRC's position on PRA consensus standards and industry PRA peer review program documents

  (3) demonstration that the baseline PRA (in total or specific pieces) used in regulatory applications is of sufficient technical adequacy

  (4) documentation to support a regulatory submittal

  It does not provide guidance on how the base PRA is revised for a specific application or how the PRA results are used in application-specific decision-making processes.

- RG 1.189, "Fire Protection for Operating Nuclear Power Plants," Revision 2, issued October 2009 (Reference 45), which provides guidance to licensees on the proper content and quality of engineering equivalency evaluations used to support the FPP. The NRC staff developed the RG to provide a comprehensive fire protection guidance document and to identify the scope and depth of fire protection that the staff would consider acceptable for nuclear power plants.

- NUREG-0800, Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection Program," Revision 0, issued December 2009 (Reference 46), which provides the NRC staff with guidance for evaluating LARs that seek to implement an RI/PB FPP in accordance with 10 CFR 50.48(c).

- NUREG-0800, Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3, issued September 2012 (Reference 47), which provides the NRC staff with guidance for evaluating the technical adequacy of a licensee's PRA results when used to request RI changes to the licensing basis.

- NUREG-0800, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," Revision 0, issued June 2007 (Reference 48), which provides the NRC staff with guidance for evaluating the risk information used by a licensee to support permanent RI changes to the licensing basis.

- NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," Volumes 1 and 2, and Supplement 1 (References 49, 50, and 51), which presents a compendium of methods, data, and tools to perform a fire probabilistic risk assessment and develop associated insights. In order to address the need for improved methods, the NRC Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI) embarked upon a program to develop state-of-art Fire PRA methodology. Both RES and EPRI have provided specialists in fire risk analysis, fire modeling, electrical engineering, human reliability analysis, and systems engineering for methods development. A formal technical issue resolution process was developed to direct the deliberative process between RES and EPRI. The process ensures that divergent technical views are fully considered, yet encourages consensus at many points during the deliberation. Significantly, the process provides that each party maintain its own point of view if consensus is not reached. Consensus was reached on all technical issues documented in NUREG/CR-6850. The methodology documented in this report reflects the current state-of-the-art in Fire PRA. These methods are expected to form a basis for risk-informed analyses related to the plant fire protection program. Volume 1, the Executive Summary, provides general background and overview information including both programmatic and technical, and project insights and conclusions. Volume 2 provides the detailed discussion of the recommended approach, methods, data and tools for conduct of a Fire PRA.

The NRC staff notes that, based on new experimental information, the reduction in hot short probabilities for circuits provided with control power transformers (CPT) identified in NUREG/CR 6850 cannot be repeated in experiments and therefore may be too high and should be reduced.

- NUREG-1805, "Fire Dynamics Tools (FDTs): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program" (Reference 52), which provides quantitative methods, known as "Fire Dynamics Tools" (FDTs), to assist regional fire protection inspectors in performing fire hazard analysis. The FDTs are intended to assist fire protection inspectors in performing risk-informed evaluations of credible fires that may cause critical damage to essential safe-shutdown equipment, as required by the new reactor oversight process defined in the NRC's inspection manual.

- NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," Volumes 1 through 7 (Reference 53), which provide technical documentation regarding the predictive capabilities of a specific set of fire models for the analysis of fire hazards in nuclear power plant scenarios. This report is the result of a collaborative program with the EPRI and the National Institute of Standards and Technology (NIST). The selected models are:

  (1) FDTs developed by NRC (Volume 3)
  (2) FIVE-Rev1 developed by EPRI (Volume 4)
  (3) The zone model CFAST developed by NIST (Volume 5)
  (4) The zone model MAGIC developed by Electricite de France (EdF) (Volume 6)

(5) The computational fluid dynamics model Fire Dynamics Simulator (FDS) developed by NIST (Volume 7).

In addition to the fire model volumes, Volume 1 is the comprehensive main report and Volume 2 is a description of the experiments and associated experimental uncertainty used in developing this report.

- NUREG/CR-7010, "Cable Heat Release, Ignition, and Spread In Tray Installations during Fire (CHRISTIFIRE), Volume 1: Horizontal Trays" (Reference 54), which describes Phase 1 of the CHRISTIFIRE testing program conducted by NIST. The overall goal of this multi-year program is to quantify the burning characteristics of grouped electrical cables installed in cable trays. This first phase of the program focuses on horizontal tray configurations. CHRISTIFIRE addresses the burning behavior of a cable in a fire beyond the point of electrical failure. The data obtained from this project can be used for the development of fire models to calculate the heat release rate (HRR) and flame spread of a cable fire.

- NUREG-1855, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (Reference 55), which provides guidance on how to treat uncertainties associated with PRA in risk-informed decision-making. The objectives of this guidance include fostering an understanding of the uncertainties associated with PRA and their impact on the results of PRA and providing a pragmatic approach to addressing these uncertainties in the context of the decision-making. To meet the objective of the NUREG, it is necessary to understand the role that PRA results play in the context of the decision process. To define this context, NUREG-1855 provides an overview of the risk-informed decision-making process itself.

- NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines – Final Report" (Reference 56), which presents the state of the art in fire human reliability analysis (HRA) practice. This report was developed jointly between RES and EPRI to develop the methodology and supporting guidelines for estimating human error probabilities for human failure events following the fire-induced initiating events of a fire PRA. The report builds on existing HRA methods, and is intended primarily for practitioners conducting a fire HRA to support a fire PRA.

- NUREG-1934, "Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG)" (Reference 57), which describes the implications of the verification and validation results from NUREG-1824 for fire model users. The features and limitations of the fire models documented in NUREG-1824 are discussed relative to their use to support nuclear power plant fire hazard analyses. The report also provides information to assist fire model users in applying this technology in the nuclear power plant environment.

- NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," Revision 1 (Reference 58), which provides a simplified approach for using PRA to estimate the frequency of containment failure and bypass events that result in radioactive releases to the environment with the potential for causing early fatalities. The approach uses LERF as a measure of the risk of early fatality, and provides guidance for estimating LERF under low power and shutdown conditions.

- NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture" (Reference 59), which presents the basis, results, and related risk implications of an analysis performed by an NRC working group to assess the containment bypass potential attributable to steam generator tube rupture induced by severe accident conditions. The main result of the analysis was an estimate of the probabilities of pressure and temperature-induced failure of steam generator tubes and containment bypass frequency for the severe accident conditions considered.

- Generic Letter (GL) 2006-03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations" (Reference 60), which requested that licensees evaluate their facilities to confirm compliance with the existing applicable regulatory requirements in light of the information provided in this GL and, if appropriate, take additional actions.

- NEI 00-01, "Guidance for Post Fire Safe Shutdown Circuit Analysis," Revision 2 (Reference 61), which provides a deterministic methodology for performing post-fire safe shutdown analysis. In addition, NEI 00-01 includes information on risk-informed methods (when allowed within a plant's license basis) that may be used in conjunction with the deterministic methods for resolving circuit failure issues related to Multiple Spurious Operations (MSOs). The risk-informed method is intended for application by licensees to determine the risk significance of identified circuit failure issues related to MSOs.

- American Society of Mechanical Engineers/American Nuclear Society ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 62), which provides guidance PRAs used to support risk-informed decisions for commercial light-water reactor nuclear power plants and prescribes a method for applying these requirements for specific applications. The standard gives guidance for a Level 1 PRA of internal and external hazards for all plant operating modes. In addition, the standard provides guidance for a limited Level 2 PRA sufficient to evaluate large early release frequency (LERF). The only hazards explicitly excluded from the scope are accidents resulting from purposeful human-induced security threats (e.g., sabotage). The standard applies to PRAs used to support applications of risk-informed decision-making related to design, licensing, procurement, construction, operation, and maintenance.

## 2.3 NFPA 805 Frequently Asked Questions

In the LAR, the licensee proposed to use a number of documents commonly known as NFPA 805 Frequently Asked Questions (FAQs). The following table provides the set of FAQs the licensee used that the NRC staff referenced in the preparation of this SE, as well as the SE section(s) to which each FAQ was referenced.

Table 2.3-1: NFPA 805 Frequently Asked Questions

| FAQ # | FAQ Title and Summary | SE Section |
|---|---|---|
| 07-0030 | "Establishing Recovery Actions"<br>• This FAQ provides an acceptable process for determining the recovery actions for NFPA 805 Chapter 4 compliance. The process includes:<br>▪ Differentiation between recovery actions and activities in the main control room or at primary control station(s).<br>▪ Determination of which recovery actions are required by the NFPA 805 fire protection program.<br>▪ Evaluate the additional risk presented by the use of recovery actions.<br>▪ Evaluate the feasibility of the identified recovery actions.<br>▪ Evaluate the reliability of the identified recovery actions. | 3.2.5 |
| 07-0038 | "Lessons Learned on Multiple Spurious Operations (MSOs)"<br>• This FAQ reflects an acceptable process for the treatment of MSOs during transition to NFPA 805:<br>▪ Step 1 – Identify potential MSO combinations of concern.<br>▪ Step 2 – Expert panel assesses plant specific vulnerabilities and reviews MSOs of concern.<br>▪ Step 3 – Update the fire PRA and Nuclear Safety Capability Assessment to include MSOs of concern.<br>▪ Step 4 – Evaluate for NFPA 805 compliance.<br>▪ Step 5 – Document the results. | 3.2.1, 3.2.4 |

| FAQ # | FAQ Title and Summary | SE Section |
|---|---|---|
| 07-0039 | "Incorporation of Pilot Plant Lessons Learned – Table B-2"<br>• This FAQ provides additional detail for the comparison of the licensee's safe shutdown strategy to the endorsed industry guidance, NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Revision 1 (Reference 63). In short, the process has the licensees:<br>▪ Assemble industry and plant-specific documentation;<br>▪ Determine which sections of the guidance are applicable;<br>▪ Compare the existing safe shutdown methodology to the applicable guidance; and<br>▪ Document any discrepancies. | 3.2.1 |
| 07-0040 | "Non-Power Operations (NPO) Clarifications"<br>• This FAQ clarifies an acceptable NFPA 805 NPO program. The process includes:<br>▪ Selecting NPO equipment and cabling.<br>▪ Evaluation of NPO Higher Risk Evolutions (HRE).<br>▪ Analyzing NPO key safety functions (KSF).<br>▪ Identifying plant areas to protect or "pinch points" during NPO HREs and actions to be taken if KSFs are lost. | 3.5.3, 3.5.3.1 thru 3.5.3.4 |
| 08-0054 | "Compliance with Chapter 4 of NFPA 805"<br>• This FAQ provides an acceptable process to demonstrate Chapter 4 compliance for transition:<br>▪ Step 1 – Assemble documentation<br>▪ Step 2 – Document Fulfillment of Nuclear Safety Performance Criteria<br>▪ Step 3 – Variance From Deterministic Requirements (VFDR) Identification, Characterization, and Resolution Considerations<br>▪ Step 4 – Performance-Based Evaluations<br>▪ Step 5 – Final VFDR Evaluation<br>▪ Step 6 – Document Required Fire Protection Systems and Features | 3.1.1, 3.5.1.7 |
| 10-0059 | "Monitoring Program"<br>• This FAQ provides clarification regarding the implementation of an NFPA 805 monitoring program for transition. It includes:<br>▪ Monitoring program analysis units;<br>▪ Screening of low safety significant structures, systems, and components;<br>▪ Action level thresholds; and<br>▪ The use of existing monitoring programs. | 3.7 |

| FAQ # | FAQ Title and Summary | SE Section |
|---|---|---|
| 12-0062 | "Updated Final Safety Analysis Report (UFSAR) Content"<br>• This FAQ provides the necessary level of detail for the transition of the fire protection sections within the UFSAR. | 2.4.4 |
| 12-0064 | "Hot Work/Transient Fire Frequency Influence Factors"<br>• This FAQ clarifies and updates the treatment of hot work and transient fire frequency influence factors. The updated treatment involves the use of sensitivity studies when the updated influence factors are used. | 3.4.2.2, 3.4.7 |

## 2.4 Orders, License Conditions, and Technical Specifications

Paragraph 50.48(c)(3)(i) of 10 CFR states, in part, that the LAR "... must identify any orders and license conditions that must be revised or superseded, and contain any necessary revisions to the plant's technical specifications and the bases thereof."

### 2.4.1 Orders

The NRC staff reviewed Section 5.2.3, "Orders and Exemptions," and Attachment O, "Orders and Exemptions," of CNP's LAR dated July 1, 2011 (Reference 6), with regard to NRC-issued Orders pertinent to CNP that are being revised or superseded by the NFPA 805 transition process. The LAR stated that the licensee conducted a review of I&M docketed correspondence to determine if there were any orders or exemptions that needed to be superseded or revised. The LAR also stated that the licensee conducted a review to ensure that compliance with the physical protection requirements, security orders, and adherence to those commitments applicable to CNP are maintained. The licensee discussed the affected orders and exemptions in Attachment O of the LAR. The licensee requested that 13 exemptions be rescinded, and determined that no orders need to be superseded or revised to implement an FPP at CNP that complies with 10 CFR 50.48(c).

This review, conducted by I&M, included an assessment of docketed correspondence files and electronic searches, including the NRC's Agencywide Documents Access and Management System (ADAMS). The review was performed to ensure that compliance with the physical protection requirements, security orders, and adherence to commitments applicable to CNP are maintained. The NRC staff accepts the licensee's determination that 13 exemptions need to be rescinded and that no orders need to be superseded or revised to implement NFPA 805 at CNP. See Section 2.5 of this SE for the staff's detailed evaluation of the exemptions being rescinded.

In addition, the licensee performed a specific review of the license amendment that incorporated the mitigation strategies required by 10 CFR 50.54(hh)(2) to ensure that any changes being made in order to comply with 10 CFR 50.48(c) do not invalidate existing commitments applicable to CNP. The licensee's review of this regulation and the related license amendment demonstrated that changes to the FPP during transition to NFPA 805 will not affect the mitigation measures required by 10 CFR 50.54(hh)(2). The NRC staff accepts the licensee's determination in regard to 10 CFR 50.54(hh)(2).

2.4.2 License Conditions

The NRC staff reviewed LAR Section 5.2.1, "License Condition Changes," and Attachment M, "License Condition Changes," regarding changes the licensee seeks to make to the CNP fire protection license condition in order to adopt NFPA 805, as required by 10 CFR 50.48(c)(3).

The NRC staff reviewed the revised license condition, which replaces the current CNP fire protection license conditions, for consistency with the format and content guidance in Regulatory Position C.3.1 of RG 1.205, Revision 1, and with the proposed plant modifications identified in the LAR.

The license conditions provide a structure and detailed criteria to allow self-approval for RI/PB as well as other types of changes to the FPP. The structure and detailed criteria result in a process that meets the requirements in Sections 2.4, Engineering Analyses, 2.4.3, Fire Risk Evaluations and 2.4.4, Plant Change Evaluation of NFPA 805. These sections establish the requirements for the content and quality of the engineering evaluations to be used for approval of changes.

The license conditions also define the limitations imposed on the licensee during the transition phase of plant operations when the physical plant configuration does not fully match the configuration represented in the fire risk analysis. The limitations on self-approval are required because NFPA 805 requires that the risk analyses be based on the as-built, as-operated and maintained plant, and reflect the operating experience at the plant. Until the proposed implementation items and plant modifications are completed, the risk analysis is not based on the as-built, as-operated and maintained plant.

Overall, the revised license conditions provide structure and detailed criteria to all self-approval for FPP changes that meet the requirements of NFPA 805 with regard to Engineering Analysis, Fire Risk Evaluations, and Plant Change Evaluations. The staff's evaluation of the Self-Approval Process for Fire Protection Program Changes (Post-Transition) is contained in Section 2.6 of this SE. The license conditions also reference the plant-specific modifications, and associated implementation schedules that must be accomplished at CNP to complete transition to NFPA 805 and achieve full compliance with 10 CFR 50.48(c). In addition, the license conditions includes a requirement that appropriate compensatory measures will remain in place until implementation of the specified plant modifications is completed. These modifications and implementation schedules are identical to those identified elsewhere in the LAR and supplements, as discussed by the NRC staff in Sections 2.8.1 and 2.8.2, and reviewed in Section 3.0, of this SE.

Because (1) the licensee's revised license conditions are consistent with the content and format of the sample license condition in RG 1.205, Revision 1, considering that the plant-specific modifications referenced in the license conditions are identical to those reviewed in this SE, and (2) this SE and the associated license conditions supersede all previous FPP SERs, the NRC staff concludes that the revised license conditions for CNP, Units 1 and 2, are acceptable. Section 4.0 of this SE provides the revised CNP FPP license conditions.

2.4.3 Technical Specifications

The NRC staff reviewed LAR Section 5.2.2, "Technical Specifications," and Attachment N, "Technical Specification Changes," with regard to proposed changes to the CNP TSs that are being revised or superseded during the NFPA 805 transition process. According to the LAR, the licensee conducted a review of the CNP TSs to determine which, if any, TS sections will be impacted by the transition to an RI/PB FPP based on 10 CFR 50.48(c) and identified that no changes are needed. The NRC staff found that the licensee had previously requested, and obtained NRC approval for, removal of fire protection requirements from the CNP TSs in Amendment 208 (Unit 1) and 192 (Unit 2) (Reference 42). As a result of the licensee's removal of fire protection requirements from the CNP TSs, the NRC staff concludes that no additional changes to the CNP TSs were required to support the NFPA 805 transition process.

2.4.4 Updated Final Safety Analysis Report

The NRC staff reviewed LAR Attachment S, "Table S-3 implementation items," with regard to changes CNP is proposing to make to the Updated Final Safety Analysis Report (UFSAR). Attachment S, Table S-3, Item S-3.13, states that the UFSAR will be updated following the guidance provided in NRC FAQ 12-0062, "UFSAR Content" (Reference 64), which provides guidance for the content and level of detail for changes to the fire protection-related sections of the post-transition UFSAR.

Since the licensee stated that the update to the UFSAR after approval of the LAR, will be in accordance with 10 CFR 50.71(e) and also that the content will be consistent with the guidance contained in NEI 04-02, the NRC staff concludes that the licensee's method to update the UFSAR following the guidance in FAQ 12-0062 is acceptable.

## 2.5 Rescission of Exemptions

The NRC staff reviewed LAR Section 5.2.3, "Orders and Exemptions," Attachment O, "Orders and Exemptions," and Attachment K, "Existing Licensing Action Transition," with regard to previously-approved exemptions to Appendix R to 10 CFR Part 50, which the transition to an FPP licensing basis in conformance with NFPA 805 will supersede. These exemptions will no longer be required since upon approval of the RI/PB FPP in accordance with NFPA 805, Appendix R will not be the licensing basis for CNP.

The licensee previously requested and received NRC approval for 13 exemptions from 10 CFR Part 50 Appendix R. These exemptions were discussed in detail in Attachment K of the LAR. The NRC staff individually addressed the applicability and continuing validity of these exemptions as incorporated into the NFPA 805 FPP as part of the staff's review of the appropriate section or fire area involved.

Disposition of Appendix R exemptions may follow two different paths during transition to NFPA 805:

- The exemption was found to be unnecessary since the underlying condition has been evaluated using RI/PB methods (fire modeling and/or fire risk evaluation) and found to be acceptable and no further actions are necessary by the licensee.
- The exemption was found to be appropriate as a qualitative engineering evaluation that meets the deterministic requirements of NFPA 805 and is carried forward as part of the engineering analyses supporting NFPA 805 transition.

The following exemptions are rescinded as requested by the LAR and the underlying condition has been evaluated using RI/PB methods and found to be acceptable with no further actions (numbering scheme provided by the licensee):

- Exemption 7.2 - Appendix R Exemption, Auxiliary Building Lack of Automatic Suppression (Criteria III.G.2.c)
- Exemption 7.3 - Appendix R Exemption, Transformer Room Unit 1 Lack of Fixed Fire Suppression (Criteria III.G.3)
- Exemption 7.4 - Appendix R Exemption, Transformer Room Unit 2 Lack of Fixed Fire Suppression (Criteria III.G.3)
- Exemption 7.5 - Appendix R Exemption, Unit 1 ESW [Emergency Service Water] Pumps and MCCs [Motor Control Centers] Lack of Fixed Fire Suppression (Criteria III.G.3)
- Exemption 7.6 - Appendix R Exemption, Unit 2 ESW Pumps and MCCs Lack of Fixed Fire Suppression (Criteria III.G.3)
- Exemption 7.8 - Appendix R Exemption, Unit 1 East Main Steam Valve Enclosure and Contractor Access Control Building Lack of Fixed Fire Suppression (Criteria III.G.3)
- Exemption 7.9 - Appendix R Exemption, Unit 2 East Main Steam Valve Enclosure Lack of Fixed Fire Suppression (Criteria III.G.3)
- Exemption 7.10 - Appendix R Exemption, Auxiliary Building South Lack of 1-Hour Fire Barrier (Criteria III.G.2.c)
- Exemption 7.11 - Appendix R Exemption, Unit 1 Main Control Room and Unit 2 Hot Shutdown Panel Enclosure Lack of Fixed Fire Suppression (Criteria III.G.3)
- Exemption 7.12 - Appendix R Exemption, Unit 2 Main Control Room and Unit 1 Hot Shutdown Panel Enclosure Lack of Fixed Fire Suppression (Criteria III.G.3)

- Exemption 7.16 - Appendix R Exemption, Lack of 8-Hour DC [Direct Current] Power for Emergency Lighting (Criteria III.J)

The following exemptions are rescinded but the engineering evaluation of the underlying condition will be used as a qualitative engineering evaluation for transition to NFPA 805:

- Exemption 7.7 - Appendix R Exemption, Screenhouse Auxiliary MCC Room Lack of Automatic Suppression (Criteria III.G.2.c)

- Exemption 7.15 - Appendix R Exemption, RCP [Reactor Coolant Pump] Lube Oil Collection System (Criteria III.O)

## 2.6 Self-Approval Process for Fire Protection Program Changes (Post-Transition)

Upon completion of the implementation of the RI/PB FPP and issuance of the license condition discussed in Section 2.4.2 of this SE, changes to the approved FPP must be evaluated by the licensee to ensure that they are acceptable. NFPA 805, Section 2.2.9, "Plant Change Evaluation," states the following:

> In the event of a change to a previously approved fire protection program element, a risk-informed plant change evaluation shall be performed and the results used as described in 2.4.4 to ensure that the public risk associated with fire-induced nuclear fuel damage accidents is low and that adequate defense-in-depth and safety margins are maintained.

NFPA 805, Section 2.4.4, "Plant Change Evaluation," states:

> A plant change evaluation shall be performed to ensure that a change to a previously approved fire protection program element is acceptable. The evaluation process shall consist of an integrated assessment of the acceptability of risk, defense-in-depth, and safety margins.

### 2.6.1 Post-Implementation Plant Change Evaluation Process

The NRC staff reviewed LAR Section 4.7.2, "Compliance with Configuration Control Requirements in Section 2.7.2 and 2.2.9 of NFPA 805," for compliance with the NFPA 805 plant change evaluation process requirements to address potential changes to the NFPA 805 RI/PB FPP after implementation is completed. The licensee developed a change process that is based on the guidance provided in NEI 04-02, Revision 2 (Reference 5), Section 5.3, "Plant Change Process," as well as Appendices B, I, and J, as modified by RG 1.205, Revision 1, Regulatory Positions 2.2.4, 3.1, 3.2, and 4.3.

LAR Section 4.7.2 states that the plant change process consists of four subtasks:

- defining the change
- preliminary risk screening
- risk evaluation
- acceptability determination

In the LAR, the licensee stated that the plant change evaluation process begins by defining the change or altered condition to be examined and the baseline configuration. The baseline is defined by the design basis and licensing basis. The licensee also stated that the baseline is defined as that plant condition or configuration that is consistent with the design basis and licensing basis. Conversely, the changed or altered condition or configuration that is not consistent with the design basis and licensing basis is defined as the proposed alternative.

The licensee stated that, once the definition of the change is established, a screening will then be performed to identify and resolve minor changes to the FPP and that the screening will be consistent with fire protection regulatory review processes currently in place. The licensee further stated that the screening process will be modeled after NEI 02-03, "Guidance for Performing a Regulatory Review of Proposed Changes to the Approved Fire Protection Program," Revision 0, June 2003 (Reference 65), and that the process will address most administrative changes (e.g., changes to the combustible control program, organizational changes, etc.).

The licensee stated that the screening will be followed by engineering evaluations that may include fire modeling and risk assessment techniques and that the results of these evaluations will then be compared to the acceptance criteria. The licensee further stated that changes that satisfy the acceptance criteria of NFPA 805, Section 2.4.4 and the fire protection license condition can be implemented within the framework provided by NFPA 805, and that changes that do not satisfy the acceptance criteria cannot be implemented within this framework. The licensee further stated that the acceptance criteria will require that the resultant change in core damage frequency (CDF) and large early release frequency (LERF) be consistent with the fire protection license condition, and that the acceptance criteria will also include consideration of defense-in-depth and safety margin, which would typically be qualitative in nature.

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

The licensee stated that the plant change evaluations will be assessed for acceptability using the $\Delta$CDF (change in core damage frequency) and $\Delta$LERF (change in large early release frequency) criteria from the license condition and that the proposed changes will also be assessed to ensure it is consistent with the defense-in-depth philosophy and that sufficient safety margins are maintained.

The licensee stated that its FPP configuration is defined by the program documentation. To the greatest extent possible, the existing configuration control processes for modifications, calculations and analyses, and FPP license basis reviews will be used to maintain configuration control of the FPP documents. The licensee further stated that the configuration control procedures that govern the various CNP documents and databases that currently exist will be revised to reflect the new NFPA 805 licensing bases requirements.

The licensee stated that several NFPA 805 document types such as: Nuclear Safety Capability Assessment (NSCA) supporting information, Non-Power Mode Review, Fire Modeling Reports, Fire Safety Assessments, risk evaluations, etc., generally require new control procedures and processes to be developed since they are new documents and databases created as a result of the transition to NFPA 805. In addition, the new procedures will be modeled after the existing processes for similar types of documents and databases. The licensee further stated that system level design basis documents will be revised to reflect the NFPA 805 role that the systems and components will play and that new procedures will be developed and existing documentation revised as part of license amendment implementation.

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

- *Deterministic Approach:* Comply with NFPA 805, Chapter 3, and 4.2.3 requirements.

- *Performance-Based Approach:* Use the NFPA 805 change process developed in accordance with NEI 04-02, RG 1.205, and the CNP NFPA 805 fire protection license condition to assess the acceptability of the proposed change. This process will be used to determine if prior NRC approval of the proposed change is required.

The licensee stated that this process follows the requirements in NFPA 805 and the guidance outlined in RG 1.174 (Reference 43), which requires the use of qualified individuals, procedures that require calculations and evaluations be subject to independent review and verification, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when errors are discovered.

Based on the information provided by the licensee, the NRC staff concluded that the licensee's plant change evaluation process is considered acceptable because it meets the guidance in NEI 04-02, Revision 2, as well as RG 1.205, Revision 1, and addresses required attributes for using fire risk evaluations (FREs) in accordance with NFPA 805. Section 2.4.4 requires that plant change evaluations consist of an integrated assessment of risk, defense-in-depth and safety margins. Section 2.4.3.1 requires that the probabilistic safety assessment (PSA) use CDF and LERF as measures for risk, Section 2.4.3.3 requires that the risk assessment approach, methods, and data shall be acceptable to the Authority Having Jurisdiction (AHJ), which is the NRC. Section 2.4.3.3 also requires that the PSA be appropriate for the nature and scope of the change being evaluated, be based on the as-built and as-operated and maintained plant, and reflect the operating experience at the plant.

The licensee's plant change evaluation process included the required delta risk calculations, uses risk assessment methods acceptable to the NRC, uses appropriate risk acceptance criteria in determining acceptability, involves the use of a fire probabilistic risk assessment (FPRA) of acceptable quality, and includes an integrated assessment of risk, DID, and safety margins as discussed above.

2.6.2 Requirements for the Self-Approval Process Regarding Plant Changes

Risk assessments performed to evaluate plant change evaluations must use methods that are acceptable to the NRC staff. Acceptable methods to assess the risk of the proposed plant change may include methods that have been used in developing the peer-reviewed FPRA model, methods that have been approved by the NRC via a plant-specific license amendment or through NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

Based on the information provided by the licensee in the LAR, the process established to evaluate post-transition plant changes meets the guidance in NEI 04-02, Revision 2, as well as RG 1.205, Revision 1. The NRC staff concludes that the proposed plant change evaluation process at CNP, which includes defining the change, a preliminary risk screening, a risk evaluation, and an acceptability determination, as described in Section 2.6.1, is acceptable because it addresses the required delta risk calculations, uses risk assessment methods acceptable to the NRC, uses appropriate risk acceptance criteria in determining acceptability, involves the use of an FPRA of acceptable quality, and includes an integrated assessment of risk, DID, and safety margins.

However, before achieving full compliance with 10 CFR 50.48(c) by implementing the plant modifications listed in Section 2.7.1 of this SE (i.e., during full implementation of the transition to NFPA 805), risk-informed changes to the licensee's FPP may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact using its screening process discussed above. This is because the risk analysis is not consistent with the as-built, as-operated and maintained plant, since the modifications have not been completed. In addition, the licensee is required to ensure that fire protection DID and safety margins are maintained during the transition process. The "Transition License Conditions" in the proposed NFPA 805 license condition include the appropriate acceptance criteria and other attributes to form an acceptable method for meeting Regulatory Position C.3.1 of RG 1.205, Revision 1, with respect to the requirements for FPP changes during transition, and therefore demonstrate compliance with 10 CFR 50.48(c).

The proposed NFPA 805 license condition also includes a provision for self-approval of changes to the FPP that may be made on a qualitative, rather than risk-informed, basis. Specifically, the license condition states that prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental FPP elements and design requirements for which an engineering evaluation demonstrates that the alternative to the NFPA 805, Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3 element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement (i.e., has not impacted its contribution toward meeting the nuclear safety and radioactive release performance criteria), using a relevant technical requirement or standard.

Use of this approach does not fall under NFPA 805, Section 1.7, "Equivalency," because the condition can be shown to meet the NFPA 805 Chapter 3 requirement. Section 1.7 of NFPA 805 is a standard format used throughout NFPA standards. It is intended to allow owners/operators to use the latest state of the art fire protection features, systems, and equipment, provided the alternatives are of equal or superior quality, strength, fire resistance, durability, and safety. However, the intent is to require approval from the authority having jurisdiction because not all of these state of the art features are in current use or have relevant operating experience. This is a different situation than the use of functional equivalency since functional equivalency demonstrates that the condition meets the NFPA 805 code requirement. Alternatively, the licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3 elements are acceptable because the changes are "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement (with respect to the ability to meet the nuclear safety and radioactive release performance criteria), using a relevant technical requirement or standard. NFPA 805 Section 2.4 states that engineering analysis is an acceptable means of evaluating a fire protection program against performance criteria. Engineering analyses shall be permitted to be qualitative or quantitative. Use of qualitative engineering analyses by a qualified fire protection engineer to determine that a change has not affected the functionality of the component, system, procedure or physical arrangement is allowed by NFPA 805 Section 2.4.

The four specific sections of NFPA 805, Chapter 3 for which prior NRC review and approval are not required to implement alternatives that an engineering evaluation has demonstrated are adequate for the hazard are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

The engineering evaluations described above (i.e., functionally equivalent and adequate for the hazard) are engineering analyses governed by the NFPA 805 guidelines. In particular, this means that the evaluations must meet the requirements of NFPA 805, Section 2.4, "Engineering

Analyses," and NFPA 805, Section 2.7, "Program Documentation, Configuration Control, and Quality." Specifically, the effectiveness of the fire protection features under review must be evaluated and found acceptable in relation to their ability to detect, control, suppress, and extinguish a fire and provide passive protection to achieve the performance criteria and not exceed the damage threshold for the plant being analyzed. The associated evaluations must also meet the documentation content (as outlined by NFPA 805, Section 2.7.1, "Content") and quality requirements (as outlined by NFPA 805, Section 2.7.3, "Quality") of the standard in order to be considered adequate. Note that the NRC staff's review of the licensee's compliance with NFPA 805, Sections 2.7.1 and 2.7.3 is provided in Section 3.8 of this SE.

According to the LAR, the licensee intends to use an FPRA to evaluate the risk of proposed future plant changes. Section 3.4.2, "Quality of the Fire Probabilistic Risk Assessment," of this SE discusses the technical adequacy of the FPRA, including the licensee's process to ensure that the FPRA remains current. The NRC staff determined that the quality of the licensee's FPRA and associated administrative controls and processes for maintaining the quality of the PRA model is sufficient to support self-approval of future RI changes to the FPP under the proposed license conditions, the NRC staff concludes that the licensee's process for self-approving future FPP changes is acceptable.

The NRC staff also concludes that the FRE methods used at CNP to model the cause and effect relationship of associated changes as a means of assessing the risk of plant changes during transition to NFPA 805 may continue to be used after implementation of the RI/PB FPP, based on the licensee's administrative controls to ensure that the models remain current and to assure continued quality (see SE Section 3.4.2, "Quality of the Fire Probabilistic Risk Assessment"). Accordingly, these cause and effect relationship models may be used after transition to NFPA 805 as a part of the FREs conducted to determine the change in risk associated with proposed plant changes.

## 2.7 Implementation

Regulatory Position C.3.1 of RG 1.205, Revision 1 (Reference 1), states that a license condition included in an NFPA 805 LAR should include: (1) a list of modifications being made to bring the plant into compliance with 10 CFR 50.48(c); (2) a schedule detailing when these modifications will be completed; and (3) a statement that the licensee shall maintain appropriate compensatory measures in place until implementation of the modifications are completed.

### 2.7.1 Modifications

The NRC staff reviewed LAR Attachment S, "Plant Modifications and Items to be Completed during Implementation," which describes the CNP plant modifications necessary to implement the NFPA 805 licensing basis, as proposed. These modifications are identified in the LAR as necessary to bring CNP into compliance with either the deterministic or performance-based requirements of NFPA 805. As described below, Attachment S provides a description of each of the proposed plant modifications, presents the problem statement explaining why the modification is needed, and identifies the compensatory actions required to be in place pending completion/implementation of the modification.

The NRC staff's review confirmed that the modifications identified in LAR Tables S-1 and S-2 are the same as those identified in LAR Table B-3, "Fire Area Transition," on a fire area basis, as the modifications being credited in the proposed NFPA 805 licensing basis. The staff also confirmed that LAR Tables S-1, S-2, and S-3, modifications, implementation items, and associated implementation schedule are the same as those referenced in the proposed NFPA 805 license condition.

LAR Attachment S, Table S-1, provides a listing of the already completed modifications performed at CNP as part of the NFPA 805 transition.

LAR Table S-2 provides a detailed listing of the plant modifications that must be completed in order for CNP to be in full compliance with NFPA 805, implement many of the attributes upon which this SE is based, and thereby meet the requirements of 10 CFR 50.48(c). These modifications will be implemented in accordance with the schedule provided in the proposed NFPA 805 license condition, which states that all modifications will be in place within 12 months from the issuance of the license amendments.

2.7.2 Schedule

LAR Section 5.4, as revised by supplemental letter dated May 1, 2013, provides the overall schedule for completing the NFPA 805 transition at CNP. The licensee stated that it will complete the implementation of the new program, including procedure changes, process updates, and training to affected plant personnel to implement the NFPA 805 FPP within 12 months after issuance of the NFPA 805 SE.

Revised LAR Section 5.4 also states that I&M will initiate the implementation of plant modifications following submittal of the LAR and anticipates completion of installation in the plant within 12 months from the issuance of the NFPA 805 SE.

## 2.8 Summary of Implementation Items

Implementation items are items that the licensee has not fully completed or implemented as of the issuance date of the license amendments, but which will be completed during implementation of the license amendment to transition to NFPA 805 (e.g., procedure changes that are still in process, or NFPA 805 programs that have not been fully implemented). These items do not impact the bases for the safety conclusions made by the NRC staff in the associated SE.

The licensee identified the implementation items in Attachment S, Table S-3 of the LAR. For each implementation item, the licensee and the NRC staff have reached a satisfactory resolution involving the level of detail and main attributes that each remaining change will incorporate upon completion. In addition, the licensee provided a date by which each implementation item will be completed.

Each implementation item will be completed prior to the deadline for implementation of the RI/PB FPP based on NFPA 805, as specified in the license condition and the letter transmitting the amended licenses.

The NRC staff, through an onsite audit or during a future fire protection inspection, may choose to examine the closure of the implementation items, with the expectation that any variations discovered during this review, or concerns with regard to adequate completion of the implementation item, would be tracked and dispositioned appropriately under the licensee's corrective action program.

## 3.0 TECHNICAL EVALUATION

The following sections evaluate the technical aspects of the requested license amendment to transition the FPP at CNP to one based on NFPA 805 in accordance with 10 CFR 50.48(c). While performing the technical evaluation of the licensee's submittal, the NRC staff used the guidance provided in NUREG-0800 Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection" (Reference 46), to determine whether the licensee had provided sufficient information in both scope and level of detail to adequately demonstrate compliance with the requirements of NFPA 805, as well as the other associated regulations and guidance documents discussed in Section 2.0 of this SE. Specifically:

- Section 3.1 provides the results of the NRC staff's review of the licensee's transition of the fire protection program from the existing deterministic guidance to that of NFPA 805, Chapter 3, "Fundamental Fire Protection Program and Design Elements."

- Section 3.2 provides the results of the NRC staff's review of the methods used by the licensee to demonstrate the ability to meet the nuclear safety performance criteria (NSPC).

- Section 3.3 provides the results of the NRC staff's review of the fire modeling methods used by the licensee to demonstrate the ability to meet the NSPC using a fire modeling performance-based approach.

- Section 3.4 provides the results of the NRC staff's review of the fire risk assessments used to demonstrate the ability to meet the NSPC using a FRE performance-based approach.

- Section 3.5 provides the results of the NRC staff's review of the licensee's NSCA results by fire area.

- Section 3.6 provides the results of the NRC staff's review of the methods used by the licensee to demonstrate an ability to meet the radioactive release performance criteria.

- Section 3.7 provides the results of the NRC staff's review of the NFPA 805 monitoring program developed as a part of the transition to an RI/PB FPP based on NFPA 805.

- Section 3.8 provides the results of the NRC staff's review of the licensee's program documentation, quality assurance, and configuration management.

In addition, Attachments A and B to this SE provide additional detailed information that was evaluated and/or dispositioned by the NRC staff to support the licensee's request to transition to an RI/PB FPP in accordance with NFPA 805 (i.e., 10 CFR 50.48(c)). These attachments are discussed as appropriate in the associated section of this SE.

## 3.1 NFPA 805 Fundamental FPP Elements and Minimum Design Requirements

NFPA 805, Chapter 3, contains the fundamental elements of the FPP and specifies the minimum design requirements for fire protection systems and features that are necessary to meet the standard. The fundamental FPP elements and minimum design requirements include necessary attributes pertaining to the fire protection plan and procedures, the fire prevention program and design controls, internal and external industrial fire brigades, and fire protection SSCs. However, 10 CFR 50.48(c) provides exceptions, modifications, and supplementations to certain aspects of NFPA 805, Chapter 3, as follows:

- 10 CFR 50.48(c)(2)(v) – *Existing cables.* In lieu of installing cables meeting flame propagation tests as required by Section 3.3.5.3 of NFPA 805, a flame-retardant coating may be applied to the electric cables, or an automatic fixed fire suppression system may be installed to provide an equivalent level of protection. In addition, the italicized exception to Section 3.3.5.3 of NFPA 805 is not endorsed.

- 10 CFR 50.48(c)(2)(vi) – *Water supply and distribution.* The italicized exception to Section 3.6.4 of NFPA 805 is not endorsed. Licensees who wish to use the exception to Section 3.6.4 of NFPA 805 must submit a request for a license amendment in accordance with 10 CFR 50.48(c)(2)(vii).

- 10 CFR 50.48(c)(2)(vii) – *Performance-based methods.* While Section 3.1 of NFPA 805 prohibits the use of performance-based methods to demonstrate compliance with the NFPA 805, Chapter 3, requirements, 10 CFR 50.48(c)(2)(vii) specifically permits that the FPP elements and minimum design requirements of NFPA 805, Chapter 3, may be subject to the performance-based methods permitted elsewhere in the standard.

Furthermore, Section 3.1 of NFPA 805 specifically allows the use of alternatives to the NFPA 805, Chapter 3, fundamental FPP requirements that have been previously approved by the NRC (which is the AHJ, as denoted in RG 1.205), and are contained in the currently approved FPP for the facility.

### 3.1.1 Compliance with NFPA 805, Chapter 3, Requirements

The licensee used the systematic approach described in NEI 04-02, Revision 2 (Reference 5), as endorsed by the NRC in RG 1.205 Revision 1 (Reference 1), to assess the proposed CNP FPP against the NFPA 805, Chapter 3 requirements.

As part of this assessment, the licensee reviewed each section and subsection of NFPA 805, Chapter 3, against the existing CNP FPP and provided specific compliance statements for each NFPA 805, Chapter 3, attribute that contained applicable requirements. As discussed below, some subsections of NFPA 805, Chapter 3, do not contain requirements, or are otherwise not applicable to CNP, and others are provided with multiple compliance statements to fully document compliance with the element.

The approach used by CNP for achieving compliance with the NFPA 805, Chapter 3, fundamental FPP elements and minimum design requirements is described as follows:

1. The existing FPP element directly complies with the requirement: noted in LAR Attachment A, "NEI 04-02 Table B-1, Transition of Fundamental Fire Protection Program and Design Elements (NFPA 805, Chapter 3)," also called the B-1 Table, as "Complies."

2. The existing FPP element complies through the use of an explanation or clarification: noted in the B-1 Table as "Complies with clarification."

3. The existing FPP element complies through the use of existing engineering equivalency evaluations (EEEEs) whose bases remain valid and are of sufficient quality: noted in the B-1 Table as "Complies with use of EEEEs."

4. The existing FPP element complies with the requirement based on prior NRC approval of an alternative to the fundamental FPP attribute and the bases for the NRC approval remain valid: noted in the B-1 Table as "Complies by previous NRC approval."

5. The existing FPP element does not comply with the requirement, but the licensee is requesting specific approval for a performance-based method in accordance with 10 CFR 50.48(c)(2)(vii): noted in the B-1 Table as "Submit for NRC Approval."

6. The existing FPP element does not comply with the requirement, but will be in direct compliance with the completion of a required action: noted in the B-1 Table as "Complies with Required Action."

Compliance approach #6, "Complies with Required Action," is a modification from the NEI 04-02 based approach in that it is a new category not included in NEI 04-02. The intent of this choice is to identify FPP elements that will comply after completion of an action by the licensee. The required actions are identified as implementation items in Attachment S, Table S-3 of the LAR.

The NRC staff has determined that, taken together, this constitutes an acceptable approach for documenting compliance with the NFPA 805, Chapter 3, requirements because the licensee has followed the compliance strategies identified in NEI 04-02 and the compliance approach #6 modification, as noted above, identifies a required action the licensee will perform and will bring the program into full compliance for those applicable elements.

The licensee stated in LAR Section 4.2.2, "Existing Engineering Equivalency Evaluation Transition," that it evaluated the EEEEs used to support compliance with the NFPA 805, Chapter 3, requirements in order to ensure continued appropriateness, quality, and applicability to the current plant configuration. The licensee determined that no EEEE used to support compliance with NFPA 805 required NRC approval.

EEEEs refer to "existing engineering equivalency evaluations" (previously known as Generic Letter 86-10 evaluations) performed for fire protection design variances such as fire protection system designs and fire barrier component deviations from the specific fire protection deterministic requirements. Once a licensee transitions to NFPA 805, future equivalency evaluations are to be conducted using a performance-based approach. The evaluation should demonstrate that the specific plant configuration meets the performance criteria in the standard.

Additionally, the licensee stated in LAR Section 4.2.3, "Licensing Action Transition," that the existing licensing actions used to demonstrate compliance have been evaluated to ensure that their bases remain valid. The results of these licensing action evaluations are provided in Attachment K of the LAR.

LAR Attachment A (the NEI 04-02 B-1 Table) provides further details regarding the licensee's compliance strategy for specific NFPA 805, Chapter 3, requirements, including references to where compliance is documented.

3.1.1.1 Compliance Strategy -- Complies

For certain NFPA 805, Chapter 3, requirements, as modified by 10 CFR 50.48(c)(2), the licensee determined that the RI/PB FPP complies directly with the fundamental FPP element using the existing FPP element. In these instances, based on the validity of the licensee's statements, the NRC staff concludes that the licensee's statements of compliance are acceptable.

3.1.1.2 Compliance Strategy -- Complies with Clarification

For certain NFPA 805, Chapter 3, requirements, the licensee provided additional clarification when describing its means of compliance with the fundamental FPP element. In these instances, the NRC staff reviewed the additional clarifications and concluded that the licensee will meet the underlying requirement for the FPP element as clarified.

The following NFPA 805 sections, identified in LAR Table B-1 as complying via this method, required additional review by the NRC staff:

- 3.3.1.2(1) Control of Combustible Materials – Use of Wood Within Power Block

The licensee identified a clarification to the stated requirement listed in Attachment A, Table B-1, Section 3.3.1.2(1) of the LAR that relied on alternate protection methods for untreated wood. However, in NRC Request for Additional Information (RAI) 10 and 10.01, dated January 27, 2012 (Reference 66), and October 11, 2012 (Reference 67), respectively, the NRC staff stated that there is no exception allowed for the covering of untreated wood as an alternative and the licensee was directed to provide a new compliance statement and compliance strategy that will satisfy the requirements of NFPA 805, Section 3.3.1.2(1). By letters dated April 27 and October 15, 2012 (References 8 and 11, respectively), I&M responded to the RAIs and agreed to revise LAR Attachment A, Table B-1 Section 3.3.1.2(1) to add "Complies with Required Action" and to retract the "Complies with clarification" statement. Additionally, a new Attachment S, Table S-3, implementation item S-3.16, has been added to revise procedures that control combustible materials and to require training on the requirements of NFPA 805, Section 3.3.1.2(1). The NRC staff concludes that this change in compliance strategy is acceptable, because the licensee changed its compliance to directly comply with the requirements of this NFPA 805, Chapter 3, element based on the completion of the required action discussed above.

- 3.3.1.3.1 Control of Ignition Sources – Hot Work

The licensee identified a clarification to the stated requirement listed in Attachment A Table B-1 Section 3.3.1.3.1 that fire watch personnel may have multiple duties. By letter dated January 27, 2012, in RAI 09 (Reference 66), the NRC staff requested CNP to describe what additional duties the fire watch may perform and what assures these duties will not impede the ability to perform the required fire watch duties. In response to the RAI dated April 27, 2012 (Reference 8), the licensee listed a single fire watch for multiple welding, burning or grinding activities, and the use of video cameras as sufficient fire watch monitoring for hot work in areas where radiation dose must be maintained "As Low As Reasonably Achievable" (ALARA). In RAI 09.01 (Reference 66), the NRC staff responded by indicating that the NRC does not accept these practices as providing an equivalent means of compliance with NFPA 805, NFPA 241, "Standard for Safeguarding Construction, Alteration, and Demolition Operations," or NFPA 51B, "Standard for Fire Prevention During Welding, Cutting, and Other Hot Work," and requested the licensee to provide a new compliance statement and compliance strategy that will satisfy the requirements of NFPA 805, Section 3.3.1.3.1. The licensee responded to RAI 9.01 (Reference 8) by revising LAR Attachment A, Table B-1 Section 3.3.1.3.1 to add the additional compliance method of "Complies with Required Action" with the action being to revise hot work procedures and require training on discontinuing the use of (1) video cameras for fire watch and (2) use of a single fire watch for multiple hot work activities. Completion of this action is included in LAR Attachment S, Table S-3, implementation item S-3.15. The NRC staff concludes that this change in compliance strategy is acceptable because the licensee changed its compliance and procedures, as discussed above, to no longer allow video cameras for fire watch which cannot detect a fire as well as a manned fire watch (e.g., manned fire watch can move around, smell a fire, view behind obstructions, etc.) in order to quickly detect a fire. In addition, the NRC staff concludes that the change in compliance strategy and the licensee's procedural changes, as discussed above, to no longer use a single fire watch for multiple hot work activities is acceptable because a single fire watch may not always be able to continuously monitor all assigned hot work activities including the applicable surrounding areas vertically and horizontally as stated in NFPA 51B. The NRC staff concludes that this change in compliance strategy, as discussed above, is acceptable because the intent of this NFPA 805, Chapter 3, element to have a dedicated fire watch able to continuously watch all hot work and applicable surrounding areas and quickly detect a fire is achieved.

- 3.6.3 Standpipe and Hose Stations – Proper Type of Hose Nozzle

The licensee identified a clarification to the stated requirement listed in Attachment A Table B-1 Section 3.6.3 that "The appropriate type of hose nozzle is provided to each power block area. All hose nozzles have shutoff capability and are able to control water flow from full open to full closed." By letter dated April 27, 2012 (Reference 8), in response to the NRC staff's RAI 07, CNP indicated that the hoses are equipped with adjustable spray nozzles that cannot be turned into the straight stream position. This prevents accidentally discharging a straight stream of water on energized electrical equipment and that inspection procedures specify "E-rated" (electrically safe) in the Auxiliary, Turbine, and Screen House buildings. However, straight stream nozzles are included on four fire brigade response carts staged at various plant locations to enhance response to hydrogen fires and some of the hose reels in the turbine room basements, mezzanine and main floors are equipped with foam nozzles for fighting flammable and/or combustible liquid fires.

The licensee identified that revision of fire pre-plans and subsequent training re-enforcing use of electrically safe fixed fog nozzles in high-voltage settings and use of straight stream nozzles for hydrogen fires would be included as part of LAR Attachment S, Table S-3, implementation item S-3.10. The NRC staff concludes that this clarification, as discussed above, is acceptable because the intent of this NFPA 805, Chapter 3, element to prevent firefighters from electrical shock during fire hose operation is achieved.

3.1.1.3 Compliance Strategy -- Complies with Use of EEEEs

For certain NFPA 805, Chapter 3, requirements, the licensee demonstrated compliance with the fundamental FPP element through the use of EEEEs. The NRC staff reviewed the licensee's statement of continued validity for the EEEEs, as well as a statement on the quality and appropriateness of the evaluations, and concludes that the licensee's statements of compliance in these instances are acceptable.

3.1.1.4 Compliance Strategy -- Complies via Previous NRC Approval

Certain NFPA 805, Chapter 3, requirements were supplanted by an alternative that was previously approved by the NRC. NRC approval was documented in (1) an SE dated July 31, 1979, supporting Amendments Nos. 31 and 12 to the CNP Unit 1 and 2, operating licenses (Reference 19); (2) an exemption dated December 23, 1983, approving the use of the reactor coolant pump (RCP) motor lube oil system that is not sized to contain the entire lube oil system inventory (Reference 23); (3) an SER dated June 16, 1988, approving the installation of carpet in the control room having a flame spread greater than that recommended by NRC staff guidance (Reference 29); (4) an SE dated April 9, 1991, approving the use of a minimum shift crew size of four members for up to 2 hours under certain conditions (Reference 68); (5) an

SER dated January 24, 1995, approving the installation and use of certain unsupervised circuits (Reference 39); (6) an exemption dated January 24, 1995, approving the use of 22 Auxiliary Building undampered ventilation duct penetrations (Reference 39); or (7) an SER dated April 26, 1990, approving the internal conduit seal program (Reference 35).

In each instance, the licensee evaluated the basis for the original NRC approval and determined that in all cases the bases were still valid. The NRC staff reviewed the information provided by the licensee and concludes that previous NRC approval had been demonstrated using suitable documentation that meets the approved guidance contained in RG 1.205, Revision 1 (Reference 1). Based on the licensee's statements for the continued validity of the previously approved alternatives to the NFPA 805, Chapter 3, requirements, the NRC staff concludes that the licensee's statements of compliance in these instances are acceptable.

The following NFPA 805 sections identified in LAR Table B-1 as complying via this method required additional review by the NRC staff:

- 3.3.12 Reactor Coolant Pumps

  NFPA 805, Section 3.3.12 provides requirements for reactor coolant pump (RCP) oil collection system. By letter dated October 11, 2012, the NRC staff issued RAI 57 (Reference 67) to correct a reference listed in the basis of the LAR Attachment K associated with Exemption 7.5. By letter dated November 9, 2012 (Reference 12), the licensee responded to the RAI and indicated that the licensing action described in LAR Attachment K, associated with Exemption 7.15, contains a typographical error. The discussion in the Basis section should have referred to NFPA 805, Section 3.3.12(2). The NRC staff concludes that the licensee's statements of compliance in this instance are acceptable.

- 3.4.1 On-Site Fire-Fighting Capability

  NFPA 805, Section 3.4.1, provides requirements for the fire brigade members including a minimum crew size of five. The licensee had originally claimed to comply with a clarification that there was a prior approval to use a reduced fire brigade crew size. During the NRC staff's review, NRC FAQ-12-0063, "Fire Brigade Make-Up" (Reference 69), was issued and provided additional guidance regarding the reduction in minimum crew size for up to 2 hours under certain conditions and provisions for the appropriate compliance strategy. The licensee's response dated April 27, 2012 (Reference 8), to RAI 10 dated January 27, 2012 (Reference 66), identified that its current FPP allows for the same minimum crew size under similar conditions and limitations. In response to the RAI, the licensee changed its compliance strategy from "complies with clarification" to "complies by previous NRC approval." The licensee quoted its Technical Requirements Manual (TRM) Section 10.1, which stated "the composition of the fire brigade may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence provided immediate action is taken to fill the required positions." This provision was approved by the NRC staff in Section 2.0(3) of the SE dated April 9, 1991 (Reference 68), for Amendment Nos. 154 and 138 for Unit 1 and Unit 2, respectively.

3.1.1.5 Compliance Strategy -- Submit for NRC Approval

For two of the NFPA 805, Chapter 3, requirements, the licensee requested approval for the use of a performance-based method to demonstrate compliance with a fundamental FPP element. In accordance with 10 CFR 50.48(c)(2)(vii), the licensee requested that specific approval be included in the license amendment approving the transition to NFPA 805 at CNP. The NFPA 805 sections identified in LAR Table B-1 as complying with this method are as follows:

- 3.2.3(1) Procedures – Inspection, Testing, and Maintenance

  In Attachment A, Table B-1 of the LAR, the licensee identified that EPRI Technical Report (TR)-1006756, "Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features," July 2003 (Reference 70), may be used to determine performance-based surveillance frequencies. EPRI TR-1006756 is published by the Electric Power Research Institute and provides guidance for licensees to follow in order to optimize their fire protection surveillance and testing practices and frequencies for fire protection systems, structures, and components based upon performance. In RAI 05 dated January 27, 2012 (Reference 66), and RAI 05.01 dated October 11, 2012 (Reference 67), the NRC staff identified this compliance strategy as unacceptable and stated that if the licensee intends to use a performance-based alternative program for managing inspection, testing, and maintenance, then a request for approval must be submitted to the NRC in accordance with 10 CFR 50.48(c)(2)(vii). The licensee's response dated October 15, 2012 (Reference 11), provided a new LAR Attachment L Approval Request #2 in accordance with 10 CFR 50.48(c)(2)(vii). See Section 3.1.4.1 of this SE for the NRC staff's evaluation of this request.

- 3.5.16 Water Supply – For Fire Protection Use Only

  Contrary to the requirements of NFPA 805, Section 3.5.16, and NFPA 24, Standard for the Installation of Private Fire Service Mains and their Appurtenances, Section 5-7, the fire protection water supply system at CNP may periodically be used to supply water for non-fire protection purposes. See Section 3.1.4.2 of this SE for the NRC staff's evaluation on this request.

As discussed in SE Section 3.1.4 below, the NRC staff concludes that the use of performance-based methods to demonstrate compliance with these fundamental FPP elements is acceptable.

3.1.1.6 Compliance Strategy -- Complies with Required Action

In certain NFPA 805, Chapter 3, requirements, as modified by 10 CFR 50.48(c)(2), the licensee determined that the RI/PB FPP will comply with the fundamental FPP element after completion of a required action. The required actions were identified as follows:

- 3.3.1.1 General Fire Prevention Activities

Initial General Employee Training (GET) to be verified and updated, after the LAR approval as part of the FPP transition to NFPA 805, to include the minimum FPP elements as discussed in Section K to NEI-04-02 (FAQ 06-0028). Completion of this action is identified as implementation item S-3.1 in Attachment S, Table S-3 of the LAR.

- 3.2.3(3) Procedures – Reviews of Fire Protection Program

The monitoring program required by NFPA 805, Section 2.6 will be implemented after the LAR approval as part of the FPP transition to NFPA 805, in accordance with NFPA 805 FAQ 10-0059, and will include a process that reviews the FPP performance and trends in performance. Completion of this action is identified as implementation item S-3.2 in Attachment S, Table S-3 of the LAR.

In RAI 59 dated October 11, 2012 (Reference 67), the NRC staff requested clarifications regarding implementation items listed in Attachment S, Table S-3 of the LAR that were not identified in the appropriate compliance strategy "Complies with Required Action" in LAR Attachment A Table B-1. In the licensee's response dated October 15, 2012 (Reference 11), the LAR Table B-1 compliance strategy was updated for several NFPA 805, Chapter 3 elements to add the compliance method "Complies with Required Action." The impacted elements in LAR Table B-1, and the associated actions for each, are as follows:

- 3.2.3 Procedures

Documents to be updated include technical documents and procedures that relate to the new RI/PB FP design and licensing basis (e.g., Fire Protection Program Manual, Technical Requirements Manual, Design Basis Document, maintenance and surveillance, configuration control, training and qualification guidelines, Quality Assurance Program Document) as needed for implementation of NFPA 805. Completion of this action is identified as implementation item S-3.10 in Attachment S, Table S-3 of the LAR.

- 3.3.1.2 Control of Combustible Materials

Transient combustible free zones will be established in high-risk fire areas AA40, AA43, AA48, AA50, AA51, and AA52. Completion of this action is identified as implementation item S-3.3 in Attachment S, Table S-3 of the LAR.

- 3.3.1.2 (1) Control of Combustible Materials – Use of Wood Within Power Block

The procedure for the control of combustibles will be revised and training on the revised procedure will be provided to CNP staff as required by NFPA 805, Section 3.3.1.2(1). Completion of this action is identified as implementation item S-3.16 in Attachment S, Table S-3 of the LAR.

- 3.3.1.3.1 Control of Ignition Sources – Hot Work

Hot work restriction zones will be established in high-risk fire areas AA40, AA43, AA48, AA50, AA51, and AA52. Completion of this action is identified as implementation item S-3.4 in Attachment S, Table S-3 of the LAR. Additionally, the procedure for welding, burning, and grinding activities will be revised and training will be required on discontinuing the use of (1) video cameras for fire watch and (2) use of a single fire watch for multiple hot work activities. Completion of this action is identified as implementation item S-3.15 in Attachment S, Table S-3 of the LAR.

- 3.3.1.3.4 Control of Ignition Sources – Portable Heaters

The licensee currently uses a fuel-fired ventilation fan for temporary ventilation of the Control Room. An alternate approach for Control Room temporary ventilation that is consistent with GDC-3 and NFPA 805, Section 3.3.1.3.4 will be developed and the technical evaluation for recovery action (RA) transition will be revised to reflect this alternate approach. This is further described in Section 3.5.1.9 of this SE. Completion of this action is identified as implementation item S-3.17 in Attachment S, Table S-3 of the LAR.

- 3.4.2 Pre-Fire Plans

Pre-fire plans will be revised to reflect changes required to meet the NFPA 805 radioactive release performance criteria. Completion of this action is identified as implementation item S-3.7 in Attachment S, Table S-3 of the LAR.

- 3.4.3 Training and Drills

The fire brigade training materials will be revised to reflect changes required to meet the NFPA 805 radioactive release performance criteria. Completion of this action is identified as implementation items S-3.7 in Attachment S, Table S-3 of the LAR.

The licensee provided mark-ups of the appropriate pages of the LAR Attachment A, Table B-1, in Reference 11. Based on the validity of the licensee's statements, the NRC staff concludes that the licensee's statements of compliance as supplemented are acceptable.

3.1.1.7 Compliance Strategy as supplemented -- Multiple Strategies

In certain compliance statements of the NFPA 805, Chapter 3, requirements, the licensee used more than one of the above strategies to demonstrate compliance with aspects of the fundamental FPP element.

In each of these cases, the NRC staff concludes that the individual compliance statements are acceptable, for the reasons outlined above, that the combination of compliance strategies are acceptable, and ensures holistic compliance with the fundamental FPP element.

3.1.1.8 Chapter 3 Sections Not Reviewed

Some NFPA 805, Chapter 3, sections either do not apply to the transition to an RI/PB FPP at CNP, or have no technical requirements. Accordingly, the NRC staff did not review these sections for acceptability. The sections that were not reviewed fall into one of the following categories:

- Sections that do not contain any technical requirements (e.g., NFPA 805, Chapter 3, Section 3.4.5 and Section 3.11).
- Sections that are not applicable to CNP because of the following:
  - The licensee stated that CNP does not have systems of this type installed (i.e., NFPA 805, Chapter 3, Section 3.9.1(3), NFPA 750 Standard on Water Mist Fire Protection Systems; Section 3.9.1(4), NFPA 16 Standard for the Installation of Foam-Water Sprinkler and Foam-Water Spray Systems; Section 3.10.1(3), NFPA 2001 Standard on Clean Agent Fire Extinguishing Systems).
  - The type of system, while installed at CNP, is not required under the RI/PB FPP (i.e., NFPA 805, Chapter 3, Section 3.10.4, which contains fire suppression system requirements for any area required to be protected by both primary and backup gaseous fire suppression systems).
  - The requirements are structured with an applicability statement (e.g., NFPA 805, Chapter 3, Section 3.4.1(a)(2) and Section 3.4.1(a)(3), wherein the determination of which NFPA code(s) apply to the fire brigade depends on the type of brigade specified in the FPP).

3.1.1.9 Compliance with Chapter 3 Requirements Conclusion

The NRC staff evaluated the results of the licensee's assessment of the proposed CNP RI/PB FPP against the NFPA 805, Chapter 3, fundamental FPP elements and minimum design requirements, as modified by the exceptions, modifications, and supplementations in 10 CFR 50.48(c)(2). Based on this review of the licensee's submittal, as supplemented, the NRC staff concludes that the RI/PB FPP is acceptable with respect to the fundamental FPP elements and minimum design requirements of NFPA 805, Chapter 3, as modified by 10 CFR 50.48(c)(2), because the licensee accomplished the following:

- Used an overall process consistent with NRC staff approved guidance to determine the state of compliance with each of the applicable NFPA 805, Chapter 3, requirements.

- Provided appropriate documentation of CNP's state of compliance with the NFPA 805, Chapter 3, requirements, which adequately demonstrated compliance in that the licensee was able to substantiate that it complied:

  - With the requirement directly.

  - With the intent of the requirement (or element) given adequate justification.

  - Via previous NRC staff approval of an alternative to the requirement.

  - Through the use of an engineering equivalency evaluation.

  - Through the use of a combination of the above methods.

  - Through the use of a performance-based method that the NRC staff has specifically approved in accordance with 10 CFR 50.48(c)(2)(vii).

  - With the requirement directly after the completion of an implementation item.

### 3.1.2 Identification of the Power Block

The NRC staff reviewed the CNP structures and fire areas identified in LAR Table I-1, "CNP Power Block Definition," as composing the "power block." The licensee stated that the methodology used to develop the list of Power Block structures is based on the guidance described in NEI 04-02, Revision 2, Section K.2 (Reference 5). The plant structures listed are established as part of the "power block" for the purpose of denoting the structures and equipment included in the CNP RI/PB FPP that have additional requirements in accordance with 10 CFR 50.48(c) and NFPA 805. As stated in the LAR, the power block includes all structures that have equipment required for nuclear plant operations. The identified structures include all of the equipment for SSCs required for the safe and reliable operation of the nuclear power plant. It includes all safety-related and balance-of-plant systems and components required for the operation of the station, including a large area called the Yard (Fire Area YD) that encompasses all locations inside the owner-controlled area with equipment required for nuclear plant operations, and that are not contained in another of the Nuclear Safety Capability Assessment (NSCA) fire areas. The equipment in Fire Area YD includes such items as offsite power distribution equipment (i.e., unit auxiliary and reserve transformers), portions of the non-safety power distribution system (i.e., main generator step-up transformer, 764-345, and 34.5 kV switchyard transformers), and the fire pump house.

LAR Table I-1, Power Block Definition, lists power block structures for the purpose of defining NFPA 805 applicability in accordance with the definitions and methodology of NEI 04-02, Revision 2, Section K. In RAI 56 (Reference 67), the NRC staff requested that the licensee clarify identification of structures in Fire Area YD. In the response to the RAI (Reference 11), the licensee identified modifications to LAR Table I-1 CNP Power Block Definition and provided a new LAR Table I-1. The structures added included Control Rooms, Cable Vaults & HVAC [Heating, Ventilation, and Air Conditioning] Equipment Areas, Service and Office Building, Fuel Handling Areas, Screen House, ESW Pump & Tunnel Areas and Water Intake and Discharge System, Containment Access, Fire Pump House. Additionally, the licensee provided a list of structures excluded from the power block because they are not required to meet either the NSPC or radioactive release performance goals. The response also included a more definitive description of offsite power distribution equipment and the Supplemental Diesel Generator Area.

Based on the information provided in the LAR, as supplemented, the NRC staff concludes that the licensee has appropriately evaluated the structures and equipment at CNP, and has adequately documented a list of those structures that fall under the definition of "power block" in NFPA 805.

3.1.3 Closure of Generic Letter 2006-03, "Potentially Nonconforming Hemyc™ and MT™ Fire Barrier Configurations" Issues

CNP does not use either the Hemyc™ or MT™ electrical raceway fire barrier systems (ERFBS). Therefore, the generic issue (Generic Letter (GL) 2006-03, Reference 60) related to these ERFBS is not applicable to CNP.

3.1.4 Performance-Based Methods for NFPA 805, Chapter 3, Elements

In accordance with 10 CFR 50.48(c)(2)(vii), a licensee may request NRC approval for use of the performance-based methods permitted elsewhere in the standard as a means of demonstrating compliance with the prescriptive FPP fundamental elements and minimum design requirements of NFPA 805, Chapter 3. Paragraph 50.48(c)(2)(vii) of 10 CFR requires that an acceptable performance-based approach accomplish the following:

(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).

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval (10 CFR 50.48(c)(2)(vii))," the licensee requested NRC staff review and approval of performance-based methods to demonstrate an equivalent level of fire protection for the requirements of the NFPA 805, Chapter 3, elements identified in Section 3.2.3(1), Inspection, Testing, and Maintenance, and Section 3.5.16, Fire Protection Water Supply. The NRC staff's evaluation of these proposed methods is provided below.

3.1.4.1 Section 3.5.16, Non-dedicated Use of the Fire Protection Water Supply

The licensee requested NRC staff review and approval of a performance-based method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Section 3.5.16 regarding dedicated use of the fire protection water supply. Specifically, the licensee has requested approval of a performance-based method to justify the periodic use of the fire protection water supply at CNP for non-fire protection purposes.

By letters dated April 27 and October 15, 2012, in response to NRC staff RAIs 11 dated January 27, 2012, and 11.01 dated October 11, 2012 (References 8 and 11), regarding the controls in place for uses of fire water for non-fire protection purposes, the licensee stated there are only two identified uses of non-fire protection purposes:

- Permanently piped cooling water (approximately 240 gallons per minute (gpm) maximum) to the Security Diesel Generator jacket water coolers.

- Tube sheet water lance cleaning of the Unit 1 or 2 Main Feed Pump Turbine Condensers via various fire hose stand pipe locations (approximately 100 gpm). This use is controlled as a proceduralized temporary plant modification in accordance with station procedures.

As described by the licensee, this usage is subject to the following conditions: (1) prior approval is obtained from the CNP fire protection staff and (2) personnel utilizing the fire protection water are in contact with the CNP Control Room. These controls are described as sufficient to ensure that the fire water system is not impaired and can be secured and restored to full capacity should a fire occur.

Additional controls in place for use of the fire water for non-fire use other than the two specific purposes described above are:

- Engineering evaluation has concluded that there is sufficient pumping capacity to supply the largest demand as well as non-fire use operation.

- Control Room Operations staff and fire brigade are notified of the non-fire uses, and contingencies established for prompt restoration in a fire event.

- Should non-fire water use of the fire water system impact operability of any of the three primary fire pumps, station procedure[s] provides guidance for establishing backup fire water capability from specified local township fire hydrants and for refilling the fire water storage tanks.

- A station procedure ensures that work activities on plant SSCs are conducted within the requirements of the station license, including TRM Section 8.7.5. All non-fire water uses of the fire water system are included in the Work Control

Process, which includes proper planning, scheduling, execution, and risk assessment.

The licensee stated that the use of the fire protection water for non-fire protection system water demands would have no adverse impact on the ability of the fire protection system to provide required flow and pressure based on the following:

- Controls in place, as described above, to cease the non-fire protection use should a fire condition occur.
- The system, which consists of one 2,500 gpm electric motor driven fire pump and two redundant 2,500 gpm diesel engine driven fire pumps connected by a common header to two 685,000 gallon fire protection water storage tanks, is designed to provide water in excess of that required to suppress a fire.
- During a largest demand fire scenario, a safety margin of approximately 1,300 gpm is maintained even with only two of the three pumps in operation.

The licensee concluded that the use of the fire protection water for non-fire protection uses does not impact the NSPC because the CNP fire water system has excess capacity to supply the combined demands of automatic and manual water-based fire suppression systems and non-fire protection uses in the event of a fire, even in the unlikely event of a delay in ceasing the use the fire protection water for non-fire protection purposes. For this same reason, the licensee concluded that the use of the CNP fire water system for non-fire protection uses has no impact on maintaining fire protection defense-in-depth because suppression systems are not affected.

The licensee also stated that this alternative will have no effect on the NFPA 805 radiological release performance goals, performance objectives, and performance criteria, since use of the CNP fire water system for non-fire protection uses has no impact on the radiological release.

The licensee further stated that the proposed alternative maintains the safety margins of the licensee's analyses related to fire water use functions, because the proposed alternative will not alter the methods, input parameters, and acceptance criteria used in fire water demand. The CNP fire water system has excess capacity to supply the combined demands of the automatic and manual water-based fire suppression systems and non-fire protection uses in the event of a fire. Finally, the licensee stated that fire protection defense-in-depth will be maintained because the fire water system pumps have excess capacity to supply demands of automatic and manual water-based fire suppression systems and non-fire protection uses in the event of a fire.

Based on its review of the information submitted by the licensee, and in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed performance-based method is an acceptable alternative to the corresponding NFPA 805, Section 3.5.16 requirement because it satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, maintains sufficient safety margin, and maintains adequate fire protection DID.

3.1.4.2 Section 3.2.3(1), Inspection, Testing, and Maintenance

As discussed in Section 3.1.1.5 of this SE, the NRC staff questioned the licensee's originally proposed compliance strategy for NFPA 805, Section 3.2.3(1). In response to RAI 05.01 dated October 15, 2012 (Reference 11), the licensee supplemented LAR Attachment L with Approval Request #2. In this supplement, the licensee requested NRC staff review and approval of a performance-based method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Section 3.2.3(1) regarding Inspection, testing, and maintenance for fire protection systems and features credited by the FPP. Specifically, the licensee requested NRC approval to use performance-based methods at CNP to establish the appropriate inspection, testing, and maintenance frequencies for fire protection systems and features required by NFPA 805.

As described by the licensee, performance-based inspection, testing, and maintenance frequencies would be established using the methods described in EPRI TR-1006756, "Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features," Final Report, July 2003 (Reference 70).

The licensee stated that the use of this method for establishing inspection, testing, and maintenance frequencies will have no adverse impact on the ability to provide assurance that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analyses.

The licensee stated that there will be no impact on the NFPA 805 nuclear safety performance goals, performance objectives, and performance criteria because the use of performance based test frequencies established per EPRI Technical Report TR-1006756 methods, combined with NFPA 805, Section 2.6, will provide assurance that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analyses. This will ensure that there is no impact on the ability of the fire protection systems and features to perform its function.

The licensee also stated that the radiological release performance goals, objectives, and criteria are satisfied based on the determination of limiting radioactive release (LAR Attachment E). Fire protection systems and features are credited as part of that evaluation. Use of performance based test frequencies established per EPRI Technical Report TR-1006756 methods combined with NFPA 805, Section 2.6 Monitoring Program will ensure that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analyses, which include those assumptions credited to meet the Radioactive Release performance criteria. Therefore, there will be no adverse impact to Radioactive Release performance criteria.

The licensee further stated that the proposed alternative maintains the safety margins of the licensee's analyses because it will provide assurance that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analyses, which includes those assumptions credited in the FRE safety margin discussions. In addition, the use of these methods in no way invalidates the inherent safety margins contained in the codes used for design and maintenance of fire protection systems and features. Therefore, the safety margin inherent and credited in the analyses will be preserved.

The three echelons of DID described in NFPA 805, Section 1.2 are (1) to prevent fires from starting (combustible/hot work controls); (2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans); and (3) provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, RAs). Echelon 1 is not affected by the use of EPRI Technical Report TR-1006756 methods. Use of performance-based test frequencies established per EPRI Technical Report TR-1006756 methods, combined with NFPA 805, Section 2.6, will provide assurance that the availability and reliability of the fire protection systems and features credited for DID are maintained to the levels assumed in the NFPA 805 engineering analyses. Therefore, there will be no adverse impact to Echelons 2 and 3 of DID.

Based on its review of the information submitted by the licensee, and in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed performance-based method is an acceptable alternative to the corresponding NFPA 805, Section 3.2.3(1) requirement because it satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, maintains sufficient safety margin, and maintains adequate fire protection DID.

## 3.2 Nuclear Safety Capability Assessment Methods

NFPA 805 (Reference 4) is an RI/PB standard that allows engineering analyses to be used to show that FPP features and systems provide sufficient capability to meet the requirements of 10 CFR 50.48(c).

NFPA 805, Section 2.4, "Engineering Analyses," states the following:

> Engineering analysis is an acceptable means of evaluating a fire protection program against performance criteria. Engineering analyses shall be permitted to be qualitative or quantitative in accordance with Figure 2.4. The effectiveness of the fire protection features shall be evaluated in relation to their ability to detect, control, suppress, and extinguish a fire and provide passive protection to achieve the performance criteria and not exceed the damage threshold defined in Section [2.5] for the plant area being analyzed.

Chapter 1 of the standard defines the goals, objectives, and performance criteria that the FPP must meet in order to be in accordance with NFPA 805.

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

> The nuclear safety goal is to provide reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition.

NFPA 805, Section 1.4.1, "Nuclear Safety Objectives":

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.

NFPA 805, Section 1.5.1, "Nuclear Safety Performance Criteria":

Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met.

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

(b) *Inventory and Pressure Control.* With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling coolant level such that subcooling is maintained for a PWR [pressurized-water reactor] and shall be capable of maintaining or rapidly restoring reactor water level above top of active fuel for a BWR [boiling-water reactor] such that fuel clad damage as a result of a fire is prevented.

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

(d) *Vital Auxiliaries.* 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.

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

3.2.1 Compliance with NFPA 805 Nuclear Safety Capability Assessment Methods

NFPA 805, Section 2.4.2, "Nuclear Safety Capability Assessment," states the following:

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

This section of the SE evaluates the first three of the topics listed above. Section 3.5 of this SE addresses the assessment of the fourth step.

RG 1.205, Revision 1 (Reference 1), endorses NEI 04-02, Revision 2 (Reference 5), and Chapter 3 of NEI 00-01, Revision 2, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," May 2009 (Reference 61), and promulgates the method outlined in NEI 04-02 for conducting an NSCA. This NRC-endorsed method documents in a table format (i.e., NEI 04-02, Table B-2, "NFPA 805, Chapter 2 – Nuclear Safety Transition – Methodology Review") the licensee's comparison of its post-fire safe shutdown analyses to the guidance in NEI 00-01, Chapter 3, which has been determined to address the related requirements of NFPA 805, Section 2.4.2. The NRC staff reviewed the LAR Section 4.2.1, "Nuclear Safety Capability Assessment Methodology," and Attachment B, "NEI 04-02 Table B-2 – Nuclear Safety Capability Assessment – Methodology Review," against these guidelines.

The endorsed guidance provided in NEI 00-01, Revision 2 provides a framework to evaluate the impact of fires on the ability to maintain post-fire safe shutdown. It provides detailed guidance for:

- Selecting systems and components required to meet the NSPC
- Selecting the cables necessary to achieve the NSPC
- Identifying the location of nuclear safety equipment and cables
- Appropriately conservative assumptions to be used in the performance of the NSCA

The licensee developed the CNP NFPA 805 LAR based on the guidance provided in the three guidance documents cited above. Although RG 1.205, Revision 1, endorses NEI 00-01, Revision 2, the licensee's review was performed using the guidance in NEI 00-01, Revision 1, January 2005 (Reference 63), as discussed below with regard to the NRC staff's RAI 19 dated January 27, 2012 (Reference 66). Based on the information provided in the licensee's submittal, as supplemented, I&M used a systematic process to evaluate the CNP post-fire safe shutdown analysis against the requirements of NFPA 805, Section 2.4.2, Subsections (1), (2), and (3), which meets the methodology outlined in the latest NRC-endorsed industry guidance.

FAQ 07-0039, "Lessons Learned – NEI 04-02 B-2 Table" (Reference 71), provides one acceptable method for documenting the comparison of the post-fire safe shutdown analysis against the NFPA 805 requirements. This method first maps the existing post-fire safe shutdown analysis to the NEI 00-01, Chapter 3 methodology, which in turn, is mapped to the NFPA 805, Section 2.4.2 requirements.

The licensee performed this evaluation by comparing the CNP post-fire safe shutdown analysis against the NFPA 805 NSCA requirements using the NRC-endorsed process in Chapter 3 of NEI 00-01, Revision 1, and documenting the results of the review in the B-2 Table in accordance with NEI 04-02, Revision 2, as modified by FAQ 07-0039.

The categories used by CNP to describe alignment with the NEI 00-01, Chapter 3, attributes are as follows:

1. The post-fire safe shutdown analysis directly aligns with the attribute: noted in LAR Attachment B, "NEI 04-02 Table B-2, NFPA 805, Chapter 2 – Nuclear Safety Transition – Methodology Review," also called the B-2 Table, as "Aligns."

2. The post-fire safe shutdown analysis aligns with the intent of the attribute: noted in the B-2 Table as "Aligns with Intent."

3. The post-fire safe shutdown analysis does not align with the attribute, but there is a prior NRC approval of an alternative to the attribute, and the bases for the NRC approval remain valid: noted in the B-2 Table as "Not in Alignment, but Prior NRC Approval."

As stated above, the licensee performed the review of the NSCA to the guidance of NEI 00-01, Revision 1 instead of Revision 2 as endorsed by RG 1.205, Revision 1. In RAI 19 dated January 27, 2012 (Reference 66), the NRC staff requested that the licensee perform a gap analysis to demonstrate the methodology applied at CNP meets the guidelines of NEI 00-01, Revision 2. In its response to RAI 19 dated April 27, 2012 (Reference 8), the licensee stated that:

> The NSCA was initiated during the period that NEI-00-01, Revision 1, was endorsed by the NRC. Subsequent to completion of the system and component selection, and the circuit analysis, Revision 2 to NEI-00-01 was issued in May 2009, and endorsed by the NRC. A review of NEI-00-01, Revision 2, was performed at that time and the differences from Revision 1 were determined to have no impact on the tasks completed at that time.

The licensee's response also identified the following clarifications to address differences in criteria and/or assumptions identified in NEI 00-01, Revision 2:

- Multiple Spurious Operations (MSOs) were evaluated consistent with the process outlined in FAQ 07-0038, "Lessons Learned on Multiple Spurious Operations" (Reference 72).

- With respect to NEI 00-01, Section 3.2.1.2, manual operation of valves must consider the effect of a fire on stem lubrication, where applicable. A review was conducted to address instances where this may be applicable at CNP. Additional discussion is provided in Section 3.2.1.1 below for this specific attribute regarding the assessment of manual operation of rising stem valves.

- With respect to NEI 00-01, Section 3.5.2.4, the potential for loss of DC control power adversely impacting the over-current trip capability on applicable switchgear was evaluated in accordance with CNP NFPA 805 transition project procedures.

The licensee's response stated that, based on its review against the endorsed criteria as described above, CNP aligns with the guidance provided in NEI 00-01, Revision 2.

Based on the documented review by the licensee against the detailed criteria in NEI 00-01, Revision 1 provided in the LAR, and the evaluation above of the additional attributes identified by the licensee in its gap assessment to Revision 2, the NRC staff concludes that the NSCA method review performed by the licensee addressed the required attributes from NFPA 805 Section 2.4.2. The NSCA method evaluated the selection of systems and components needed to meet the NSPC, selection of cables required to meet the NSPC, identification of equipment and cable locations, and conservative analysis assumptions to be used in the NSCA.

3.2.1.1 Attribute Alignment -- Aligns

For the majority of the NEI 00-01, Chapter 3, attributes, the licensee determined that the post-fire safe shutdown analysis aligns directly with the attribute. In these instances, based on the validity of the licensee's statements, the NRC staff concludes the licensee's statements of alignment are acceptable.

The following attribute identified in LAR Table B-2 as aligning via this method required additional review by the NRC staff:

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

The licensee stated in the LAR Attachment B that its methodology aligns with Section 3.2.1.2 of NEI 00-01, Revision 1. NEI 00-01, Revision 2, Section 3.2.1.2 provides additional guidance from that provided in NEI 00-01 Revision 1, regarding credit for post-fire operation of rising stem manual valves that have been exposed to fire conditions. In RAI 19 and RAI 19.01 dated January 27 and October 11, 2012 (References 66 and 67), the NRC staff requested additional information from the licensee regarding credit taken for post-fire operation of rising stem valves exposed to fire. In its responses dated April 27 and October 15, 2012 (References 8 and 11), the licensee identified three fire areas where a recovery action (RA) involving operation of a manual valve in the affected area may be necessary post-fire. The three fire areas are AA54 (Unit 1 charging pump area), AA55 (Unit 2 charging pump area), and AA36/42 (Auxiliary Building elevation 609 ft).

The RAs associated with AA54 and AA55 involve manual valve operations to cross-tie the Unit 1 and Unit 2 Chemical and Volume Control System (CVCS). The licensee stated that this cross-tie action is defense-in-depth. Thermal barrier cooling is unaffected for scenarios impacting these valves and would provide RCP seal cooling. Reactor coolant system boundary failures are not assumed for these scenarios and reactivity control is provided by insertion of the control rods. The FPRA determined the delta risk of the associated Variance From Deterministic Requirements (VFDRs) to be acceptably low and the actions would not take place for at least 24 hours following a fire allowing for evaluation and restoration of the valve. For fire area, AA36/42, a fire may impact the Volume Control Tank (VCT) suction valves. As stated in the licensee's response, there are multiple fire scenarios in this area that can impact cables for these valves; however, there is only one transient scenario that can directly expose the valves to thermal damage. Damage to the VCT outlet valves due to direct exposure or cable damage can lead to loss of the charging pumps. The PRA determined the availability of the safety injection pumps as an acceptable method for inventory makeup and the risk associated with failure of the valves was determined to be acceptably low. The licensee stated the RA would not take place for at least 24 hours following the fire allowing for evaluation and restoration of the valve as needed to support manual operation.

Based on the licensee's determination of low risk associated with these actions, the time available for repair before these valves are needed, or alternative means of accomplishing the function, the NRC staff concludes that the licensee's statement of alignment to the endorsed guidance in Section 3.2.1.2 of NEI 00-01 is acceptable. See Sections 3.2.4 and 3.4.4 of this SE for the NRC staff's conclusions regarding the licensee's crediting of RAs at CNP.

3.2.1.2 Attribute Alignment -- Aligns with Intent

For a few of the NEI 00-01, Chapter 3, attributes, the licensee determined that the post-fire safe shutdown analysis aligns with the intent of the attribute, and provided additional clarification when describing its means of aligning with the attribute. The NEI 00-01, Chapter 3, attributes identified in LAR Table B-2 as having this condition are 3.2.1.5, 3.2.2.1, 3.3.1.7, 3.3.3.3, 3.5.1.2, 3.5.1.4, 3.5.1.5, 3.5.2.1, 3.5.2.4, and 3.5.2.5.

Four of the attributes of NEI 00-01 (3.3.1.7, 3.3.3.3, 3.5.2.4, and 3.5.2.5) for which the licensee stated alignment with intent, address associated circuits by common power supply or common enclosure. The intent of Sections 3.3.1.7, 3.3.3.3, and 3.5.2.4 is that the licensee should either verify proper electrical circuit coordination or require inclusion of additional cables and performance of additional circuit analysis if proper breaker and fuse protection/coordination is not provided (normally referred to as common power supply concerns). The intent of section 3.5.2.5 is that electrical circuits should have adequate circuit protective devices to assure that secondary fires are not caused by fire-induced electrical faults (normally referred to as common enclosure concerns). As described in the LAR and reviewed by NRC staff, the licensee performed several circuit coordination studies for the CNP. NRC staff review of the licensee's circuit coordination documentation identified secondary fire concerns associated with fires impacting 250 Volts direct current (VDC) control power in the 4 kilovolt (kV) switchgear rooms, resulting in the potential for secondary fires upstream of the 4 kV bus and concerns associated with 250 VDC circuits, which are described in the licensee documentation but for which no resolution was identified. In RAI 15 dated January 27, 2012 (Reference 66), the NRC staff requested additional clarification of this secondary fire issue. In its response to the RAI dated April 27, 2012 (Reference 8), the licensee stated that it used undue conservatism in the evaluation of the cable protection requirements. Further review by the licensee of the issue in accordance with the guidance provided in NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Revision 2, has determined that a change in the cable protection strategy is not required.

The licensee determined the CNP has adequate cable protection and coordination, and can justify the conditions identified as deficiencies associated with electrical coordination (i.e., coordination between bus load protective devices and bus supply protective devices) in a site-specific calculation. The licensee completed an evaluation of those conditions and determined that there is adequate cable protection and coordination, and that secondary fires are not a concern. Therefore, there is no impact on the ability to achieve and maintain the NSPC, and no impact on the FPRA. Based on the licensee's revised evaluation, the condition discussed in RAI 15 (Reference 66) aligns with the intent of the NEI 00-01, Revision 2 guidance since the licensee performed detailed analyses demonstrating that secondary fires will not result from common enclosure concerns.

The revised LAR Attachment M was provided as Enclosure 3 to I&M letter dated June 21, 2013 (Reference 16).

Additionally, LAR Attachment S, "Plant Modifications and Items to be Completed during Implementation," Pages S-5 and S-6, Item S-2.3, regarding replacement of 250 VDC fuses and maintenance of compensatory measures was deleted. A new Attachment S, Table S-3, implementation item S-3.23 was added to revise the appropriate engineering calculations and update the following documentation to reflect the revised response to RAI 15(b) documented in I&M letters dated January 14, 2013 (Reference 13), February 1, 2013 (Reference 14), May 1, 2013 (Reference 15), and June 21, 2013 (Reference 16).

1. Revise Unit 1 and Unit 2 250 Volt DC calculations 1-E-N-ELCP-250-001, "Unit 1 250 VDC System Coordination Study," and 2-E-N-ELCP-250-001, "Unit 2 250 VDC System Coordination Study." These will be non-technical revisions to provide clarity regarding the CNP Licensing Basis and Design Requirements (specifically, the Safe Shutdown Capability Assessment Manual (SSCA) and NEI 00-01).

2. Revise 600 Volt alternating current (AC) calculations 1-E-N-PROT-BKR-007, "Unit 1 600V Switchgear Breaker 11A6, 11A7, 11B3, 11C3, 11C9, 11C18, 11D9 and OB2-1 setting." This will be a non-technical revision to provide clarity regarding the NFPA 805 requirements in the case of a potentially overloaded cable.

3. Create new Unit 1 and Unit 2 120 Volt AC calculations. A representative sample of cable data from buses identified in Technical Evaluation 12.5 was evaluated for adequate cable protection and coordination. Results of the survey revealed that all cables were acceptable with significant margin to preclude cable damage and secondary fires. The new calculations will document the assembled data for each 120 Volt AC cable and protective device. The new calculations by bounding analysis are to be completed no later than March 28, 2013.

4. Update Technical Evaluation 12.5. This update will provide justification for resolution of conditions currently identified as deficiencies with 250 VDC, 600 Volt AC and 120 Volt AC calculations."

The revised LAR Attachment S was provided as Enclosure 4 to I&M letter dated June 21, 2013 (Reference 16). Based on the licensee's analyses described in its response to NRC staff questions and the revised analyses already performed described in the RAI 15 response and the implementation items referenced in the NFPA 805 License Condition to complete the described analyses (items 1, 2, 3, and 4, above), the NRC staff concludes that the licensee's methods align with the intent of NEI 00-01 since common power supply and common enclosure concerns have been addressed in the NSCA.

The licensee also stated that the CNP methodology aligned with the intent of NEI 00-01 attribute 3.5.2.1, which addresses, in part, potential secondary fires associated with open circuit failures on circuits with current transformer (CT). As described in LAR Section 4.2.1.1, additional manufacturer and model data is needed to complete the evaluation of the CTs. The licensee identified these actions as implementation item S-3.12 in Attachment S, Table S-3 of the LAR. In RAI 16 dated January 27, 2012 (Reference 66), the NRC staff requested additional information from the licensee on the specific action to be taken, the status of the implementation, and how the resolution will affect the analysis, including delta CDF (ΔCDF) and delta LERF (ΔLERF). In its response to the RAI dated April 27, 2012 (Reference 8), the licensee described that the evaluation of CTs is continuing; however, based on the results to date, the actions to evaluate the CTs are expected to show that there is no detrimental effect on the NSCA, ΔCDF, or ΔLERF.

The licensee further stated that, should a detrimental effect be identified, strategies to resolve the condition will ensure no detrimental effect on the NSCA, ΔCDF, or ΔLERF, since the potential for secondary fires has not been postulated to occur in the FPRA. The licensee stated that the investigation will be completed and documentation updated prior to implementation of NFPA 805, as indicated by implementation item S-3.12 in Attachment S, Table S-3 of the LAR. Based on the licensee's decision to conduct an evaluation to completely resolve any secondary fire conditions associated with the evaluation of the CTs in a manner that does not impact the

NSCA or FPRA, and to complete the evaluation prior to implementation of NFPA 805, the NRC staff concludes that the licensee aligns with the intent of NEI 00-01 to preclude the generation of secondary fires as a result of open circuiting a CT under load.

The remaining attributes of NEI 00-01 for which the licensee stated that CNP aligns with intent are 3.2.1.5, 3.2.2.1, 3.5.1.2, 3.5.1.4, and 3.5.1.5. These attributes describe similar means or methods that were applied by the licensee to achieve the intended result of the NEI 00-01 guidance. NEI 00-01 attribute 3.2.1.5 contains criteria for assumed failure states of instrumentation. The licensee specifically describes how instrumentation providing control functions or spurious operation concerns were treated in the safe shutdown analysis, which conservatively addressed the appropriate failure modes. Attribute 3.2.2.1 describes annotation of piping and instrumentation drawings (P&IDs) to identify flow paths for shutdown paths. In addition to P&IDs, the licensee described a broader set of documents and diagrams used to identify flow paths. Attribute 3.5.1.4 involves criteria and assumptions for failures of cables/components in the same fire area that do not meet separation requirements (i.e., cables will fail in the worst-case configuration). The licensee provided additional detail regarding the specific methods applied to determining cable and component failure. This additional detail allowed the licensee to perform more detailed failure analyses beyond the conservative assumptions given in NEI 00-01. Based on the additional analyses performed, this meets and exceeds the guidance. Attribute 3.5.1.5 is associated with cable failure criteria and assumptions for spurious operations. The licensee described the specific approach to circuit analysis used including multiple spurious operations (MSOs). The process used by the licensee considered the effects of fire damage on both thermoplastic and thermoset cable, any possible combination of conductors shorting within intra-cable and does not limit the number of cables when addressing spurious operations due to inter-cable shorting. This exceeds the guidance in NEI 00-01. Although under this attribute the licensee described the previous licensing basis consideration of the "double break" design at CNP, which precluded the need to consider multiple cable-to-cable hot shorts, the approach used during transition considered the possibility of spurious actuations using risk-informed, performance-based analysis techniques. Based on the discussion provided above, the NRC staff concludes that the methods as described by the licensee are sufficiently similar to (and in several cases exceed) the specific methods in NEI 00-01, and therefore align with the intent of NEI 00-01.

3.2.1.3 Attribute Alignment -- Not in Alignment, but Prior NRC Approval

For one of the NEI 00-01, Chapter 3, attributes, the licensee determined that the post-fire safe shutdown analysis does not align with the attribute, but there is a prior NRC approval of an alternative to the attribute. The NEI 00-01, Chapter 3, attribute identified in LAR Table B-2 as complying via this method is 3.5.2.3.

This section provides guidance for analyzing the effects of a hot short on circuits for required safe shutdown equipment. A hot short is defined as a fire-induced insulation breakdown between conductors of the same cable, a different cable or some other external source resulting in an undesired impressed voltage on a specific conductor. The potential effect of the undesired impressed voltage would be to cause equipment to operate or fail to operate in an undesired manner.

The previously approved attribute is described in LAR Attachment B, Section 3.5.2.3, and is associated with the licensee's approach to evaluating hot shorts and specifically, the credit taken for "double-break" circuit design in mitigating spurious operations. In the Alignment Statement and Comments in Section 3.5.2.3, the licensee stated that this previous approval will not be carried forward in the transition to the new licensing basis and that circuit failures associated with inclusion of the cables added to the NFPA 805 analysis for these double-break valves will be captured, analyzed, and addressed using risk-informed, performance-based techniques and therefore aligns with NEI 00-01. Based on the statement that the licensee will not carry the previous approval forward in the transitioned licensing basis and cables for the double-break valves will be included in the NFPA 805 analysis, the NRC staff concludes that the licensee's analysis aligns with the NEI 00-01 guidance and is, therefore, acceptable.

3.2.1.4 Attribute Alignment -- Not in Alignment, but No Adverse Consequences

The licensee did not identify any attributes in this category in LAR Table B-2.

3.2.1.5 Attribute Alignment -- Not in Alignment

The licensee did not identify any attributes in this category in LAR Table B-2.

3.2.1.6 NFPA 805 Nuclear Safety Capability Assessment Methods Conclusion

The NRC staff reviewed the documentation provided by the licensee describing the process used to perform the NSCA required by NFPA 805, Section 2.4.2. The licensee performed this evaluation by comparing the CNP post-fire safe shutdown analysis against the NFPA 805 NSCA requirements using NEI 00-01, Revision 1, with a gap analysis to the NRC-endorsed process in Chapter 3 of NEI 00-01, Revision 2. The results of the review are documented in the B-2 Table in accordance with NEI 04-02, Revision 2, and the gap analysis of NEI 00-01, Revision 2, was addressed in the response to RAI 19 dated April 27, 2012 (Reference 8). Based on the information provided in the licensee's submittal, as supplemented, the NRC staff accepts the method the licensee used to perform the NSCA with respect to the selection of systems and equipment, selection of cables, and identification of the location of nuclear safety equipment and cables, as required by NFPA 805, Section 2.4.2. The NRC staff accepts the licensee's method because it either:

- Met the NRC-endorsed guidance directly, or
- Met the intent of the endorsed guidance with adequate justification, or
- Had a previous NRC staff approval of an alternative to the guidance.

3.2.2 Safe and Stable

The nuclear safety goals, objectives, and performance criteria of NFPA 805 allow more flexibility than the previous deterministic FPPs based on Appendix R to 10 CFR 50 and NUREG-0800, Section 9.5.1 (Reference 73), as well as, in part NEI 00-01, Chapter 3, since NFPA 805 only requires the licensee to maintain the fuel in a safe and stable condition rather than achieve and maintain cold shutdown.

The NRC staff notes that although NFPA 805 analytically allows the analysis to use an end state of safe and stable, potential fire damage to SSCs may result in the inoperability of numerous items required for operation in accordance with the unit's Technical Specifications (TSs). TS action statements may require the licensee to bring the unit to cold shutdown or other conditions; licensees are cautioned that TSs must still be met.

The licensee stated that the NFPA 805 licensing basis for CNP is to achieve and maintain safe and stable hot standby (Mode 3) conditions. The licensee stated that safe and stable conditions can be maintained for an initial 24-hour coping time with minimum plant operating shift staff and based on the design capacity of selected systems. The licensee further stated that the 24-hour coping period allows for the CNP Emergency Response Organization (ERO) to respond with adequate time to muster, assess the extent of fire damage, and assist plant staff with actions to sustain hot standby or, alternatively, to assist the plant operating staff with any necessary repairs and actions to transition and proceed to cold shutdown (Mode 5) if necessary.

In support of sustained Mode 3 operations, the licensee describes in Section 4.2.1.2 of the LAR that CNP design features and plant operating procedures provide the capability to sustain Mode 3 conditions beyond 24 hours. These features include a 7-day diesel fuel-oil supply for emergency generators; sustained decay heat removal through steam generators with Auxiliary Feedwater (AFW) makeup from essential service water (ESW) and nitrogen control of the steam generator (SG) Power-Operated Relief Valves (PORVs); reactivity control initially with control rod insertion and long-term through boron injection [from the Refueling Water Storage Tank (RWST) or Boric Acid Tank (BAT)]; and Reactor Coolant System (RCS) inventory control with makeup from the RWST using the CVCS.

In RAI 25 dated January 27, 2012 (Reference 66), the NRC staff requested that the licensee provide additional discussion of the actions necessary beyond 24 hours to meet the specific NSPC and to maintain safe and stable conditions. The NRC staff also requested the licensee to provide a qualitative or quantitative evaluation of the risk associated with the failure of actions and equipment necessary to extend safe and stable beyond 24 hours given the post-fire scenarios during which they may be required. In its response dated April 27, 2012 (Reference 8), the licensee summarizes the scope of actions, repairs, and restoration activities beyond 24 hours as follows:

- Core decay heat in Mode 3 would be rejected to the secondary plant through the SGs, and then to atmosphere through the main steam safety relief valves operating as spring relief valves. For sustained Mode 3 conditions, actions can be taken to operate the SG PORVs from the Control Room or locally at their nitrogen control station.

- CNP design features provide sufficient diesel fuel oil on-site for an emergency diesel generator (as necessary, for those fire areas where offsite power is not free of fire damage) to operate for 7 days.

- CNP has design features and procedures to ensure that an adequate source of inventory is provided for decay heat removal in sustained Mode 3 conditions (i.e., Condensate Storage Tank (CST) re-fill capability from Essential Service Water (ESW) for the AFW pumps).

- Gravity insertion of the control rods into the reactor core would ensure initial reactivity control is achieved and maintained for Mode 3. However, the CNP reactor core design does not ensure that there would not be a return to criticality with the plant in sustained Mode 3. Consequently, maintaining the "safe and stable" plant condition for NFPA 805 will require boration of the RCS. CNP has design features and procedures to ensure that an adequate source of borated inventory is provided to prevent a return to criticality in sustained Mode 3 (that is, operators can add borated water from the RWST or BAT) utilizing the CVCS system.

- Inventory makeup to the RCS may be required to account for expected RCS leakage and minimal RCS shrinkage. CNP has design features and procedures to ensure that an adequate source of borated inventory is provided for RCS inventory control in sustained Mode 3 utilizing the CVCS system.

- CNP has design features and repair procedures to ensure that an adequate source of heat input is maintained for RCS pressure control in sustained Mode 3 beyond 24 hours utilizing available combinations of backup Pressurizer Heaters. The backup Pressurizer Heaters are capable of being energized from Emergency Diesel Generator power.

- CNP has design features and repair procedures to ensure the ability to depressurize the RCS utilizing the Pressurizer PORVs from the Control Room for sustained Mode 3 operations.

The licensee further stated that recovery of equipment that may be required to maintain safe and stable conditions beyond 24 hours has been qualitatively evaluated and determined to have no significant measurable contribution to risk based on the following factors:

- The number of required RAs to support the 24-hour safe and stable coping period is limited and can be performed by existing minimum shift staff personnel;

- Procedures will be in place for each recovery action, and will be validated to assure that there is sufficient time available to complete them;

- The procedures will have indications of alarms and/or indications such that there are cues for the RAs;

- The staff will be trained in the use of the post-fire operating and long-term restoration;

- Required tools and replacement parts will be maintained onsite to support long-term restoration activities and will be routinely inventoried and inspected;

- The 24-hour coping period provides a reasonable assurance that adequate time is provided to augment plant staffing to commence and assist in repair and restoration activities, if needed; and

- The plant has long term coping measures as part of other programs such as those required by 10 CFR 50.54(hh)(2) that provide redundancy to installed SSCs.

Implementation item S-3.5 in Attachment S, Table S-3 of the LAR addresses update of post-fire operating procedures and associated training to include NSCA strategies.

As described in the LAR, the licensee has modeled the CNP capability to achieve and maintain safe and stable conditions for the initial 24 hours of the event. Beyond 24 hours, the licensee has described the means to maintain safe and stable conditions and determined that these post-24-hour actions have no significant contribution to risk. NFPA 805 Section 1.3.1 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.

The goal does not have a time limit, so the licensee must be able to demonstrate, using performance-based methods, that the risk of not maintaining safe and stable conditions after the initial 24 hours is acceptable. NFPA 805 allows the use of qualitative engineering analyses to demonstrate meeting performance criteria. On the basis of the licensee's initial 24-hour analysis and the qualitative engineering analysis of the post-24-hour actions indicating no significant contribution to risk, as described in the LAR, as supplemented, the NRC staff concludes that the licensee has provided reasonable assurance that CNP can be maintained in a safe and stable condition post-fire.

3.2.3 Applicability of Feed and Bleed

As stated below, 10 CFR 50.48(c)(2)(iii) limits the use of feed and bleed:

> In demonstrating compliance with the performance criteria of Sections 1.5.1(b) and (c), a high-pressure charging/injection pump coupled with the pressurizer power-operated relief valves (PORVs) as the sole fire-protected safe shutdown path for maintaining reactor coolant inventory, pressure control, and decay heat removal capability (i.e., feed-and-bleed) for (PWRs) is not permitted.

The NRC staff reviewed LAR Table 5-3, "10 CFR 50.48(c) – Applicability/Compliance References," and Attachment C, "NEI 04-02 Table B-3 – Fire Area Transition," to evaluate whether CNP meets the feed and bleed requirements. The licensee stated in LAR Table 5-3 that feed and bleed is not used as the sole fire protected safe shutdown path at CNP for any scenario. The NRC staff verified this by reviewing the designated safe shutdown path listed in

LAR Attachment C for each fire area. This review confirmed that all fire area analyses include the safe shutdown equipment necessary to provide decay heat removal without relying on feed and bleed. In addition, all fire areas either met the deterministic requirements of NFPA 805, Section 4.2.3, or the performance-based evaluation performed in accordance with NFPA 805, Section 4.2.4, demonstrated that the integrated assessment of risk, DID, and safety margins for the fire area was acceptable. Therefore, the NRC staff determined that, based on the information provided in LAR Table 5-3 as well as the fire area analyses documented in LAR Attachment C, the licensee meets the requirements of 10 CFR 50.48(c)(2)(iii) because feed and bleed is not used as the sole fire-protected safe shutdown path at CNP.

3.2.4 Assessment of Multiple Spurious Operations

NFPA 805, Section 2.4.2.2.1, "Circuits Required in Nuclear Safety Functions," states that:

> Circuits required for the nuclear safety functions shall be identified. This includes circuits that are required for operation, that could prevent the operation, or that result in the maloperation of the equipment identified in 2.4.2.1, ["Nuclear Safety Capability Systems and Equipment Selection"]. This evaluation shall consider fire-induced failure modes such as hot shorts (external and internal), open circuits, and shorts to ground, to identify circuits that are required to support the proper operation of components required to achieve the nuclear safety performance criteria, including spurious operation and signals.

In addition, NFPA 805, Section 2.4.3.2 states that the probabilistic safety assessment (PSA) evaluation shall address the risk contribution associated with all potentially risk-significant fire scenarios. Because the performance-based approach taken at CNP used FREs in accordance with NFPA 805, Section 4.2.4.2, "Use of Fire Risk Evaluation," adequately identifying and including potential MSO combinations is required to ensure that all potentially risk-significant fire scenarios have been evaluated.

Accordingly, the NRC staff reviewed LAR Section 4.2.1.4, "Evaluation of Multiple Spurious Operations," and Attachment F, "Fire-Induced Multiple Spurious Operations Resolution," to determine whether the licensee has adequately addressed MSO concerns at CNP. As described in the LAR, the licensee's process for identification and evaluation of MSOs used an expert panel and followed the guidance of NEI 04-02, RG 1.205, and FAQ 07-0038, "Lessons Learned on Multiple Spurious Operations," Revision 3. The expert panel used by the licensee consisted of subject matter experts with experience in electrical engineering, FPRA, PRA, safe shutdown analysis, fire protection, and plant operations.

Attachment F to the LAR stated the licensee conducted an initial expert panel review in 2007 and a second review in 2008. Prior to initial review, the panel was provided with training and was provided with a specific project instruction for conducting the review. The expert panel sources for identifying MSOs included the safe shutdown analysis, generic lists (e.g., from Owners Groups), self-assessment results, PRA insights, and operating experience. The results of the initial review were integrated into the NCSA and the FPRA. The second review panel dispositioned open items from the initial expert panel review and addressed new MSOs identified since the initial review. In 2009, following the PRA peer review, the MSO report was updated to include the generic MSO list developed by Westinghouse that was released in April 2009. An additional update to the MSO list in 2011 incorporated the results of the June 2010 update to the Westinghouse generic MSO report.

LAR Attachment F, "Fire-Induced Multiple Spurious Operations Resolution," describes the process the licensee used to address MSOs. That process includes 5 steps: 1. Identify potential MSOs of concern; 2. Conduct an expert panel to assess plant specific vulnerabilities; 3. Update the Fire PRA model and NSCA to include the MSOs of concern; 4. Evaluate for NFPA 805 Compliance; and, 5. Document Results. As described in LAR Attachment F, under the results for Steps 3, 4, and 5, the MSOs identified in Steps 1 and 2 were incorporated in the FPRA model and evaluated for inclusion in the NSCA. VFDRs were created where MSO combinations did not meet the deterministic requirements of NFPA 805, Section 4.2.3. These VFDRs were addressed using the performance-based approach of NFPA 805, Section 4.2.4. Based on the evaluations, components associated with the MSOs were added to the NSCA equipment list and logics, and cable tracing and circuit analysis was performed. The CNP FPRA quantified the fire-induced risk model containing the MSO pathways. The MSO contribution is included in the FPRA results, including those associated with VFDRs in the FREs.

The NRC staff reviewed the licensee's expert panel process for identifying circuits susceptible to multiple spurious operations as described above and concludes that the licensee adopted a systematic and comprehensive process for identifying MSOs to be analyzed utilizing available industry guidance. Furthermore, the process used provides reasonable assurance that the FRE appropriately identifies and includes risk-significant MSO combinations. Based on these conclusions, the NRC staff concludes that the licensee's approach for assessing the potential for MSO combinations is acceptable for use at CNP.

3.2.5 Establishing Recovery Actions

NFPA 805, Section 1.6.52, "Recovery Action," defines a recovery action (RA) as follows:

> Activities to achieve the nuclear safety performance criteria that take place outside the main control room or outside the primary control station(s) for the equipment being operated, including the replacement or modification of components.

NFPA 805, Section 4.2.3.1, states that:

> One success path of required cables and equipment to achieve and maintain the nuclear safety performance criteria without the use of recovery actions shall be protected by the requirements specified in either 4.2.3.2, 4.2.3.3, or 4.2.3.4, as applicable. Use of recovery actions to demonstrate availability of a success path for the nuclear safety performance criteria automatically shall imply use of the performance-based approach as outlined in 4.2.4.

NFPA 805, Section 4.2.4, "Performance-Based Approach," states the following:

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

The NRC staff reviewed LAR Section 4.2.1.3, "Establishing Recovery Actions," and Attachment G, "Recovery Actions Transition," to evaluate whether the licensee meets the associated requirements for the use of RAs per NFPA 805.

The licensee based its approach for transitioning operator manual actions (OMAs) into the 10 CFR 50.48(c) Risk-Informed, Performance-Based (RI/PB) FPP as RAs on NEI 04-02, Revision 2, Section 4.6, "Regulatory Submittal and Transition Documentation," as endorsed with exceptions by RG 1.205, Revision 1. The population of OMAs addressed during the NFPA 805 transition process at CNP included the existing OMAs in the deterministic FPP, as well as those being added based on the VFDRs identified in the individual fire area assessments.

OMAs are actions performed by plant operators to manipulate components and equipment from outside the main control room to achieve and maintain post-fire hot shutdown, not including "repairs." OMAs include an integrated set of actions needed to ensure that hot shutdown can be accomplished for a fire in a specific plant area. OMAs are transitioned to RAs under NFPA 805. Recovery actions are activities to achieve the NSPC that take place outside of the main control room or outside of the primary control station(s) for the equipment being operated, including the replacement or modification of components.

CNP does not have any locations designated as primary control stations (PCS) as defined in RG 1.205. For control room evacuation scenarios, the licensee uses Local Shutdown Indication (LSI) panels to monitor plant conditions; however, the essential control capability is provided at local control stations or components. Therefore, all OMAs included in the transition to NFPA 805 are treated as RAs.

OMAs meeting the definition of an RA are required to comply with the NFPA 805 requirements outlined above. Some of these OMAs may not be required to demonstrate the availability of a success path in accordance with NFPA 805, Section 4.2.3.1, but may still be required to be retained in the RI/PB FPP because of the DID considerations described in Section 1.2 of NFPA 805. Accordingly, the licensee defined a DID recovery action (DID-RA) as an action that is not needed to meet the NSPC, but has been retained to provide DID. In each instance, the licensee determined whether a transitioning OMA was an RA, a DID-RA, or not necessary for the post-transition RI/PB FPP.

The licensee stated that all credited RAs, as listed in LAR Attachment G (including DID-RAs) were subjected to a feasibility review. Attachment G, Table G-1, "Recovery Actions and Activities Occurring at the Primary Control Stations," describes each RA associated with the disposition of a VFDR from the fire area assessments as documented in LAR Attachment C, "Fire Area Transition." The feasibility review was based on documentation only, including previous feasibility evaluations for safe shutdown OMAs. The licensee has identified implementation items S-3.8 and S-3.9 in Attachment S, Table S-3 of the LAR to perform confirmatory field walkdowns to demonstrate the feasibility of identified RAs and to update the analyses based on the walkdown results.

RAs to cross-connect CNP Unit 1 and Unit 2 CVCS, AFW, and ESW are necessary for fire scenarios in several fire areas as described in LAR Attachment G. In RAI 17 and RAI 17.01 dated January 27 and October 11, 2012 (References 66 and 67, respectively), the NRC requested additional information from the licensee regarding the feasibility of these actions, including staffing, communication, operational interface between the units, the impact of the cross-connected systems on the unaffected unit post-fire, and the contribution to risk for each unit associated with the cross-connections. In its response to the RAIs dated April 27 and October 15, 2012 (References 8 and 11, respectively), the licensee stated that two dedicated operators, in addition to the normal operating crew, are available to perform the system cross-connects for CVCS and AFW. The ESW system is permanently cross-tied and is operable from the control room. With regard to the impact on the unaffected unit, the licensee stated in its response that when modeling cross-tie of AFW, the success criteria required that AFW be operable at both units, with the unaffected unit having priority. ESW has no effect on the unaffected unit because the systems have the capacity to meet the flow requirements of both units; however, cross-tying the CVCS requires shutdown of the unaffected unit per TS 3.0.3. The licensee also confirmed that the actions to cross-tie the systems were included in the feasibility analysis for CNP. The risk impacts to each unit were provided and are discussed in Section 3.4.4 of this SE.

Based on the above considerations, the NRC staff concludes that the licensee has followed the endorsed guidance of NEI 04-02 and RG 1.205 to identify and evaluate RAs in accordance with NFPA 805, thereby meeting the regulatory requirements of 10 CFR 50.48(c). The NRC staff concludes that the feasibility criteria applied to RAs are acceptable based on conformance with the endorsed guidance contained in NEI 04-02 and successful completion of identified implementation items S-3.8, S-3.9, S-3.14, and S-3.17 in Attachment S, Table S-3 of the LAR.

The additional risk of RAs is addressed in Section 3.4.4 of this SE.

3.2.6 Conclusion for Section 3.2

The NRC staff reviewed the licensee's LAR, as supplemented, for conformity with the requirements contained in NFPA 805, Section 2.4.2, regarding the process used to perform the NSCA at CNP. First, the NRC staff concluded that the safe and stable condition, proposed by the licensee, is acceptable. Second, pending completion of implementation item S-3.12, in Attachment S, Table S-3 of the LAR, the NRC staff concluded that the licensee's process is adequate to appropriately identify and locate the systems, equipment, and cables required to provide reasonable assurance of achieving and maintaining the fuel in a safe and stable condition, as well as to meet the NSPC of NFPA 805, Section 1.5.

The NRC staff verified, through review of the documentation provided in the LAR, that feed and bleed was not the sole fire-protected safe shutdown path for maintaining reactor coolant inventory, pressure control, and decay heat removal capability, in accordance with 10 CFR 50.48(c)(2)(iii).

The NRC staff reviewed the licensee's process to identify and analyze MSOs. Based on the information provided in the LAR, as supplemented, the process used to identify and analyze MSOs at CNP is considered comprehensive and thorough. Through the use of an expert panel process in accordance with RG 1.205, NEI 04-02, and FAQ 07-0038, potential MSO combinations were identified and included as necessary into the NSCA as well as the applicable FREs. The NRC staff also considers the licensee's approach for assessing the potential for MSO combinations is acceptable because it was performed in accordance with NRC-endorsed guidance.

The NRC staff concludes that, based on the information provided in the LAR, as supplemented, the process used by the licensee to review, categorize and address RAs during the transition from the existing deterministic fire protection licensing basis to an RI/PB FPP is consistent with the NRC-endorsed guidance contained in NEI 04-02 and RG 1.205. The licensee has identified the actions to be taken at a primary control station as well as identified those actions that meet the definition of RA provided in NFPA 805 Section 1.6.52. In accordance with license conditions 2.(4)(c)3 for Unit 1 and 2.(o)III.3 for Unit 2, the licensee must complete implementation items S-3.5, S-3.8, S-3.9, S-3.14, and S-3.17, in Attachment S, Table S-3 of the LAR by the end of the implementation period. Upon completion of these implementation items, this process meets the regulatory requirements of 10 CFR 50.48(c) and NFPA 805.

### 3.3 Fire Modeling

NFPA 805 (Reference 4) allows both Fire Modeling and Fire Risk Evaluations (FRE) as performance-based alternatives to the deterministic approach outlined in the standard. These two performance-based approaches are described in NFPA 805, Sections 4.2.4.1 and 4.2.4.2, respectively. Although Fire Modeling and FRE are presented as two different approaches for performance-based compliance, the FRE approach generally involves some degree of fire modeling to support engineering analyses and scenario development. NFPA 805, Section 1.6.18, defines a fire model as a "mathematical prediction of fire growth, environmental conditions, and potential effects on structures, systems, or components based on the conservation equations or empirical data."

The NRC staff reviewed LAR Section 4.5.2, "Performance-Based Approaches," which describes how the licensee used fire modeling as part of the transition to NFPA 805 at CNP, and LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," which describes how the licensee performed fire modeling calculations in compliance with the NFPA 805 performance-based evaluation quality requirements for fire protection systems and features at CNP, to determine whether the fire modeling used to support transition to NFPA 805 is acceptable.

In LAR Section 4.5.2, the licensee indicated that in lieu of the Fire Modeling approach (NFPA 805, Section 4.2.4.1), the FRE approach (NFPA 805, Section 4.2.4.2) was used for the transition to NFPA 805. In LAR Section 4.5.1.2, Subsection on "Fire Model Utilization in the Application," the licensee indicated that fire modeling was performed as part of the FPRA development. Therefore, the NRC staff reviewed the technical adequacy of the CNP FPRA, including the supporting fire modeling analyses, as documented in Section 3.4.2 of this SE, to evaluate compliance with the NSPC.

The licensee did not propose any fire modeling methods to support performance-based evaluations in accordance with NFPA 805, Section 4.2.4.1, as the sole means for demonstrating compliance with the NSPC.

## 3.4 Fire Risk Evaluations

This section addresses the licensee's Fire Risk Evaluation (FRE) performance-based method, which is based on NFPA 805, Section 4.2.4.2. The licensee chose to use only the FRE performance-based method in NFPA 805, Section 4.2.4.2.

NFPA 805, Section 4.2.4.2, "Use of Fire Risk Evaluations," states the following:

> Use of fire risk evaluation for the performance-based approach shall consist of an integrated assessment of the acceptability of risk, defense-in-depth, and safety margins.
>
> The evaluation process shall compare the risk associated with implementation of the deterministic requirements with the proposed alternative. The difference in risk between the two approaches shall meet the risk acceptance criteria described in NFPA 805, Section 2.4.4.1 ["Risk Acceptance Criteria"]. The fire risk shall be calculated using the approach described in NFPA 805, Section 2.4.3 ["Fire Risk Evaluations"].

### 3.4.1 Maintaining Defense-in-Depth and Safety Margins

NFPA 805, Section 4.2.4.2, requires that the "use of fire risk evaluation for the performance-based approach shall consist of an integrated assessment of the acceptability of risk, defense-in-depth, and safety margins."

#### 3.4.1.1 Defense-in-Depth (DID)

When implementing the performance-based approach, the licensee followed the general guidance contained in Section 5.3.5, "Acceptance Criteria," of NEI 04-02, which includes consideration of DID and safety margins as part of the FRE process. Each FRE includes an assessment to identify if any additional systems and features are necessary to maintain DID and an assessment whether sufficient safety margins are maintained. The results of these assessments are summarized for each VFDR by fire area in LAR Attachment C Table B-3, "Attachment C – NEI 04-02 Table B-3 – Fire Area Transition."

Defense-in-Depth (DID)

NFPA 805, Section 1.2, states the following:

> Protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations is paramount to this standard. The fire protection standard shall be based on the concept of defense-

in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:

- Preventing fires from starting.

- Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage.

- 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.

The NRC staff reviewed LAR Section 4.2.4, "Fire Area Transition," Section 4.5.2.2, "Fire Risk Approach," Section 4.8.1, "Results of the Fire Area Review," and Attachment C Table B-3, "Attachment C – NEI 04-02 Table B-3 – Fire Area Transition," as well as the associated supplemental information, in order to determine whether the principles of DID were maintained in regard to the planned transition to NFPA 805 at CNP.

The licensee developed a methodology for evaluating DID that defines each of the three DID elements identified in Section 1.2 of NFPA 805, referred to as Echelons 1, 2, and 3, respectively. In response to RAI 23 dated April 27, 2012 (Reference 8), the licensee provided a table where, for each of the three echelons, several examples of fire protection features that addressed that echelon are identified, along with a discussion of the considerations used in assessing those features. For the most part, the identified fire protection features are required to be in place in order to demonstrate compliance with the fundamental FPP and design elements of NFPA 805, Chapter 3. However, credit for some of the fire protection features for DID is taken based on the results of the performance-based analyses conducted during the NFPA 805 transition (e.g., ERFBS, use of fire-rated cable, use of RAs, etc.).

To augment this qualitative evaluation of DID and use it to systematically evaluate DID within the risk-informed evaluation, the licensee defines "potentially risk significant" fire scenarios for DID purposes as follows:

- CDF ≥ 1E-6/year (yr) and/or LERF ≥ 1E-7/yr

- CDF < 1E-6/yr and ≥ 1E-8/yr and/or LERF < 1E-7/yr and ≥ 1E-9/yr and DID Echelon 1 and 2 attributes contribute significantly to risk reduction

- CCDP > 1E-1

The licensee explained in Table RAI 23-1 dated April 27, 2012 (Reference 8), that if a VFDR could be impacted by a "potentially risk significant" fire scenario, then (1) manual suppression capability may not be adequate and additional automatic suppression systems (Echelon 2) should be considered, and/or (2) internal fire area separation may not be adequate and reliance on a recovery action, supplemental barrier, or other modification may be necessary (Echelon 3).

As part of the FRE process, this method for addressing DID was implemented in fire safety analyses (FSAs) which include the FREs performed on each performance-based fire area. The FREs evaluate VFDRs and identify "potentially risk-significant" fire scenarios. Accordingly, as described in the response to RAI 23 (Reference 8), each performance-based FSA includes a table documenting the review of DID. The table: (1) documents the fire protection systems/features required to either meet the deterministic criteria of NFPA 805, Section 4.2.3, or to support the FPRA (2) notes whether changes or improvements are necessary for each fire protection system/feature to maintain a balance among the DID echelons, and (3) provides a justification or basis for why the required fire protection systems/features are adequate for DID. As such, the table in the FSA is the licensee's internal record of the systems required to meet the NSPC and DID requirements of NFPA 805.

Based on its review of the response to RAI 23 dated April 27, 2012 (Reference 8), and the FSAs during its audit of the CNP NFPA 805 transition RI/PB FPP, the NRC staff concludes that the licensee has systematically and comprehensively evaluated fire hazards, area configuration, detection and suppression features, and administrative controls in each fire area and concludes that the methodology as proposed in its LAR adequately evaluates DID against fires as required by NFPA 805 and, therefore, the proposed RI/PB FPP adequately maintains DID.

3.4.1.2 Safety Margins

NFPA 805, Section 2.4.4.3, states the following:

> The plant change evaluation shall ensure that sufficient safety margins are maintained.

NEI 04-02, Section 5.3.5.3, "Safety Margins," lists two specific criteria that should be addressed when considering the impact of plant changes on safety margins:

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

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

LAR Section 4.5.2.2, "Fire Risk Approach," stated that safety margins were considered as part of the FRE process and that each retained VFDR was evaluated against the safety margin criteria of NEI 04-02 and RG 1.205. The site-specific FSA calculations contain the details of the licensee's review of safety margins for each performance-based fire area. In response to RAI 24 dated April 27, 2012 (Reference 8), the licensee described in detail the methodology used to evaluate safety margins at CNP.

The response to RAI 24 dated April 27, 2012 (Reference 8) identified codes and standards that were used during the evaluation supporting transition to NFPA 805 and that will be used in the subsequent FPP including:

- 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 by the NRC (i.e., FAQ 06-0008, FAQ 06-0004, and FAQ 07-0033).

- PRA modeling is performed using acceptable codes and standards or acceptable alternatives, such as NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," May 2007 (Reference 53), and ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 62).

The safety margin criteria described in NEI 04-02, Section 5.3.5.3 and the LAR, as supplemented, are consistent with the criteria as described in RG 1.174 and are, therefore, acceptable. Based on its review of the response to RAI 24 (Reference 8), and the FSAs during its audit of the CNP NFPA 805 transition RI/PB FPP, the NRC staff concludes that the licensee's approach has adequately addressed the issue of safety margins in the implementation process because the licensee used appropriate codes and standards (or NRC-approved alternatives) and, through the application of its FPRA in its FREs, provided sufficient margin to account for analysis and data uncertainty.

3.4.1.3 Conclusion for Section 3.4.1

Based on the information provided by the licensee in the LAR, as supplemented, the transition process included a detailed review of fire protection DID and safety margins. The individual FSAs, which include FREs, LAR Table 4-3, and LAR Attachment C Table B-3 document the results of the licensee's DID and safety margin review. The NRC staff concludes the licensee's evaluation in regard to DID and safety margins is acceptable because the licensee's process and results are consistent with the endorsed guidance in NEI 04-02, Revision 2 and are consistent with the NRC staff guidance in RG 1.205, Revision 1 (Reference 1), and RG 1.174, Revision 1 (Reference 43). Section 3.5 of this SE discusses the results of the individual fire area reviews, including the documentation of the required suppression and detection systems.

3.4.2 Quality of the Probabilistic Risk Assessment

The objective of the PRA quality review is to determine whether the plant-specific PRA used in evaluating the proposed LAR is of sufficient scope, level of detail, and technical adequacy for the application. The NRC staff evaluated the PRA quality information provided by the licensee in its NFPA 805 submittal, as supplemented, including industry peer review results and self-assessments performed by the licensee. The NRC staff reviewed LAR Section 4.5.1, "Fire PRA Development and Assessment," Section 4.7, "Program Documentation, Configuration Control, and Quality Assurance," Attachment C, "NEI 04-02 Table B-3 – Fire Area Transition," Attachment U, "Internal Events PRA Quality," Attachment V, "Fire PRA Quality," and Attachment W, "Fire PRA Insights."

The licensee developed its internal events PRA during the Individual Plant Examination process and continued to maintain and improve the PRA as RG 1.200, "An Approach For Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Reference 44), and supporting industry standards have evolved. The licensee developed its FPRA model using the guidance of NUREG/CR-6850, "EPRI/NRC-RES, Fire PRA Methodology for Nuclear Power Facilities" (References 49, 50, and 51). The model addresses both Level 1 (core damage) and partial Level 2 (large early release) PRA during at-power conditions. The licensee modified its internal events PRA model to capture the effects of fire, both as the initiator of an event and to characterize the subsequent potential failure modes for affected circuits or individual plant SSCs (targets), including fire-affected human actions.

The licensee did not identify any: (1) known outstanding plant changes that would require a change to the FPRA model, or (2) any planned plant changes that would significantly impact the PRA model, beyond those identified and scheduled to be implemented as part of the transition to an FPP based on NFPA 805.

The licensee identified administrative controls and processes used to maintain the FPRA model current with plant changes and to evaluate any outstanding changes not yet incorporated into the PRA model for potential risk impact as a part of the routine change evaluation process. Further, as described in Section 3.8.3 of this SE, the licensee has a program for ensuring that developers and users of these models are appropriately trained and qualified. Therefore, the NRC staff concludes that the PRA should be capable of supporting post-transition FREs to support, for example, the self-approval process because it will continue to model the as-operated plant and be used by qualified personnel.

3.4.2.1 Internal Events PRA Model

The licensee's evaluation of the technical adequacy of the portions of its internal events PRA model used to support development of the FPRA model included a combination of peer reviews and gap assessments as follows:

- A 2001 Westinghouse Owners Group (WOG) peer review pre-dating the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard and RG 1.200,

- A 2004 gap assessment, following resolution of all significant facts and observations (F&Os) from the 2001 peer review, performed using the ASME RA-Sa-2003 version of the PRA standard to identify any gaps to meeting the supporting requirements (SRs), and

- A 2009 focused-scope peer review, following a series of PRA model updates, using the ASME RA-Sb-2005 version of the PRA standard, as endorsed by RG 1.200, Revision 1.

Attachment U of the LAR[2] provides the licensee's dispositions to all 100 F&Os from the 2001 WOG peer review, 2004 gap self-assessment, and 2009 focused-scope peer review. In general, an F&O is written for any SR if that SR does not fully satisfy the Capability Category II requirements of the ASME standard. Consistent with RG 1.200, data used in the internal events PRA model should meet Capability Category II for use in a PRA, unless application specific acceptable justification for each individual SR of lesser capability is provided.

As described in the revised Attachment U, the licensee dispositioned each F&O by: a) providing a description of how F&Os were resolved, b) describing how modular accident analysis program analyses have been updated to support the NFPA 805 FREs, or c) clarifying how the conclusion that the issue only impacts the internal events PRA and not the FPRA were reached. This NRC staff's reviews of the licensee resolution of each of F&Os are summarized in the NRC's record of review dated April 22, 2013 (Reference 74), along with the staff's conclusions based upon those reviews. In some cases, the NRC staff requested supplemental information to assess the adequacy of the F&O disposition. In several cases, as described below, the RAI response raised issues that required further clarification.

In RAI 20 dated January 27, 2012 (Reference 66), the NRC staff noted that there are a number of differences between the SRs of ANS RA-Sa-2003 on which the 2004 gap assessment was based, ASME RA-Sb-2005 on which the limited 2009 focused-scope peer review was based, and the SRs of RA-Sa-2009 as endorsed by RG 1.200, Revision 2. In response to the RAI dated April 27, 2012 (Reference 8), the licensee performed a new gap self-assessment against the ASME/ANS RA-Sa-2009, as modified by RG 1.200, Revision 2. The licensee clarified that, in general, all new identified gaps in the internal events PRA model generally have no impact on the FPRA because they are either in portions of the model that are not used by the CNP FPRA or they had been corrected in the CNP FPRA. The exceptions were newly identified differences between the PRA and the Capability Category II for the LERF SRs in ASME/ANS RA-Sa-2009. As discussed in Section 3.4.7 of this SE, the licensee performed a sensitivity analysis and determined that the conclusions of the LAR are not changed by upgrading the LERF analysis to remove the non-conservatisms in the CNP LERF model. In response to RAI 61 dated May 1, 2013 (Reference 15), the licensee provided new risk results that corrected this and several other methods. Therefore, the NRC staff concludes that the quantitative fire risk results may be used to support the request to transition without a peer review of this FPRA upgrade. The self-approval acceptance guidelines are much smaller than the transition acceptance guidelines and the licensee has proposed a license condition that a focused-scope peer review of the new LERF methodology in the FPRA will be completed and all related F&Os resolved before the FPRA is used to support self-approval of future changes.

In RAI 45 dated October 11, 2012 (Reference 67), the NRC staff requested that the licensee identify any PRA upgrades since the last focused-scope peer review. In its response dated October 15, 2012 (Reference 11), the licensee identified a change in its internal events methodology since the last full-scope peer review that is considered a PRA model upgrade per the ASME/ANS PRA standard and which could impact the CNP FPRA risk calculations. This new methodology involved reducing the mission time for certain components in cutsets

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[2] In response to RAI 21 dated April 27, 2012, the licensee provided a revised LAR Attachment U. The revised Attachment U is the basis of this SE.

including test and maintenance (T&M) events. The NRC staff concurs with the licensee that this new method will often affect both the compliant model and the post-transition models and therefore have limited impact on the change in risk results. The NRC staff also notes that fire risk (and VFDR risk in particular) is generally dominated by unrecoverable loss of one or more redundant trains from fire damage so cutsets containing multiple redundant failures will not be dominant and the effect of this method on the fire risk results would be further reduced. Therefore, the NRC staff concludes that the quantitative fire risk results may be used to support the request to transition regardless of the acceptability of this method. The self-approval acceptance guidelines are much smaller than the transition acceptance guidelines and the licensee has proposed a license condition that a focused-scope peer review of the T&M methodology will be completed and all related F&Os resolved before the PRA is used to support self-approval of future changes.

As a result of its review of the LAR, as supplemented, the NRC staff concludes that the CNP internal events PRA is technically adequate such that its quantitative results, considered together with the sensitivity study results, can be used to demonstrate that the change in risk due to the transition to NFPA 805 meets the acceptance guidelines in RG 1.174. To reach this conclusion, the NRC staff has reviewed all F&Os provided by the peer reviewers and determined that the resolution of every F&O supports the determination that the quantitative results are adequate. Accordingly, the NRC staff concludes that the licensee has demonstrated that the internal events PRA meets the guidance in RG 1.200, Revision 2, that it is reviewed against the applicable SRs in ASME/ANS-RA-Sa-2009, and is technically adequate to support the FREs and other risk calculations required for the NFPA 805 application. The self-approval acceptance guidelines are much smaller than the transition acceptance guidelines. The NRC staff concludes that a focused-scope peer review be performed on the FPRA LERF and that the T&M upgrades in the PRA be performed and all related F&Os resolved before the FPRA may be used to support future self-approval. The licensee proposed a license condition that includes these requirements.

3.4.2.2 FPRA Model

The licensee evaluated the technical adequacy of the CNP FPRA model by conducting a peer review of the FPRA model using the SRs of RA-Sa-2009 as endorsed by RG 1.200, Revision 2. The peer review was performed by the PWR Owners Group (PWROG) in October 2009 and reviewed all SRs in the FPRA element. Table V-1 of Attachment V in the LAR provides the dispositions to all Facts and Observations (F&Os) against SRs that were met, not met, or met at Capability Category II. Table V-2 of Attachment V in the LAR provides the resolutions of all F&Os against SRs that were determined by the peer review to be met at Capability Category I.

The NRC staff reviewed the licensee's dispositions of all of the F&Ss to determine the technical adequacy of the fire events PRA for the NFPA 805 application. The NRC staff's review and conclusion of the licensee resolution of each of the F&Os is summarized in the NRC's record of review dated April 22, 2013 (Reference 75). In some cases, the NRC staff requested supplementary information to assess the adequacy of the F&O disposition. In several cases, the RAI response raised issues that required further clarification and these issues are discussed below. Further details regarding sensitivity analyses supporting the following conclusions are provided in Section 3.4.7 of this SE.

In RAI 30 dated January 27, 2012 (Reference 66), the NRC staff noted that utilization of the fire ignition frequencies in FAQ 08-0048 (Reference 76) should have an associated sensitivity study as part of the method. In response to the RAI dated April 27, 2012 (Reference 8), the licensee provided the sensitivity study.

In RAI 38 dated January 27, 2012 (Reference 66), the NRC staff noted that new information indicated that the reduction in hot short probabilities for circuits protected by control power transformers (CPT) identified in NUREG/CR-6850 could not be repeated in experiments and therefore may be too high and should be reduced. In response to RAI 38 dated April 27, 2012 (Reference 8), the licensee provided a sensitivity study that effectively increased the hot short probability from 0.33 to 0.5. Removal of all CPT credit generally yields a probability of 0.66. In response to RAI 61 dated May 1, 2013 (Reference 15), the licensee incorporated this new method into the PRA. The NRC staff concludes that the use of 0.5 is acceptable because it does not utilize the full reduction (from 0.66 to 0.33) that cannot be repeated experimentally, but does include some credit (from 0.66 to 0.5) for the protection afforded by the CPT and, therefore, this issue is resolved.

In RAI 31 dated January 27, 2012 (Reference 66), the NRC staff requested that the licensee perform a statistical propagation of parametric uncertainty and assess the impact on the risk results presented in the LAR. Finding UNC-A1-1 on SR UNC-A1 noted that the licensee had not performed a propagation of parametric uncertainty through the PRA model. In response to the RAI dated April 27, 2012 (Reference 8), the licensee provided a sensitivity analysis in which PRA model parameters having distributions were statistically propagated in the CNP FPRA model. The sensitivity analysis results show that propagating the parametric uncertainties through the PRA model could potentially significantly impact the results of decisions because of the little margin available before the proposed CNP RI/PB FPP potentially exceeds the risk acceptance guidelines for Region II (small change) in RG 1.174. See Section 3.4.7 of this SE for the NRC staff's evaluation of this sensitivity study. Consequently, the NRC staff concludes that the PRA is technically adequate with regard to SR UNC-A1, and associated SR QU-E3, to support the FREs and other risk calculations required for the NFPA 805 application, and the licensee has demonstrated the capability to a perform the propagation of parametric uncertainty through the PRA model as needed to support future self-approval.

In RAI 34d dated January 27, 2012 (Reference 66), the NRC staff asked the licensee to provide justification for using generic fire protection system unavailability from NUREG/CR-6850 without completing an evaluation for plant specific outlier behavior. Finding SR FSS-D7 noted that an evaluation of outlier behavior was being conducted but had not been completed. In response to the RAI 34d dated April 27, 2012 (Reference 8), the licensee explained that the evaluation of plant-specific outlier behavior had been completed and provided a sensitivity analysis discussed in Section 3.4.7 using the actual unavailability times indicating the correct values do not significantly impact the risk results in the LAR. Based on this, the NRC staff concludes that the PRA is technically adequate with regard to SR FSS-D7 to support the FREs and other risk calculations required for the NFPA 805 application. Updating PRA inputs to reflect current operating experience is a standard element of acceptable PRA maintenance procedures. The staff has determined that CNP has an acceptable PRA maintenance program and, therefore, the NRC staff concludes this issue is resolved.

In RAI 15 dated January 27, 2012 (Reference 66), the NRC staff requested that the licensee clarify potential secondary fire issues first identified in Finding CS-B1-1 on SR CS-B1. The Finding CS-B1-1 is that the licensee has not completed its evaluation of secondary fires for associated circuits. During the NRC staff's audit of the NFPA 805 LAR (Reference 6), additional issues regarding secondary fires were identified and included in RAI 15. The licensee provided an initial response to RAI 15 dated April 27, 2012 (Reference 8), indicating that further work was necessary for some direct current circuits. In its response to follow-up 15(b) (Reference 14), the licensee clarified that it had completed the evaluation of possible secondary fires and no deficiencies or technical inadequacies were identified so no fire PRA model was needed and, therefore, the NRC staff concludes this issue is resolved.

In RAI 34e dated January 27, 2012 (Reference 66), the NRC staff requested that the licensee provide sensitivity analyses in which the maintenance, occupancy, and storage influence factors were each assigned values consistent with the guidelines in NRC-endorsed FAQ 12-0064 (Reference 56). In response to RAI 34e dated April 27, 2012 (Reference 8), the licensee provided sensitivity analyses in which the maintenance, occupancy, and storage influence factors were each assigned values consistent with the guidelines in NRC-endorsed FAQ 12-0064. See Section 3.4.7 of this SE for the NRC staff's evaluation of these sensitivity studies. The results of these sensitivity studies by the licensee demonstrate that the risk calculations do not change significantly for this LAR. However, the NRC staff does not find the licensee's proposed method to be acceptable because it provided excessive flexibility to distribute transient fire frequency among different plant locations with no technical justification for modifying the acceptable method. In response to RAI 61 (Reference 15), the licensee provided an integrated analysis, which provided the risk results after apportioning the transient fire frequency according to the NRC method endorsed in FAQ 12-0064, and, therefore, the NRC staff concludes this issue is resolved.

In RAI 29 dated January 27, 2012 (Reference 66), the NRC requested that the licensee provide further justification for the use of the screening human error probability (HEP) value of 0.1, which is essentially the CCDP for main control room abandonment scenarios. Suggestion FSS-A6-2 suggested a refined treatment of main control room abandonment HEP CCDP and HEP estimates. Based on the response to RAI 29 dated August 9, 2012 (Reference 10), the NRC staff could not conclude that adequate justification for the 0.1 value was provided and issued follow-up RAI 29.01 dated February 1, 2013 (Reference 77), requesting additional evaluation of the HEPs. In response to RAI 29.01 dated May 1, 2013 (Reference 15), the licensee stated that it had completed an HEP evaluation using the HEP methods generally used in its PRA as adapted for fire specific scenarios as described in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," July 2012 (Reference 56). In response to RAI 61 dated May 1, 2013 (Reference 15), the licensee provided an integrated analysis, which provided the risk results after incorporating the new HEPs into the PRA and, therefore, the NRC staff concludes this issue is resolved.

During the audit, the NRC staff noted that, contrary to the NUREG/CR-6850 method of locating transient fires at pinch-points, transient fires had not been located at some pinch-points that were difficult to reach but not inaccessible. The NRC staff concludes this method as implemented is unacceptable because the licensee defined inaccessible as areas where a fire was improbable, not impossible, and did not postulate fires in these areas. The NRC staff believes that inaccessible applies to areas unable to be reached, not simply difficult to reach. In response to RAI 40 dated October 15, 2012 (Reference 11), discussed in Section 3.4.2.1 of this SE, the licensee provided several sensitivity studies where transient fire frequency was varied and conservative conditional risk estimates were used. These studies indicated that changes to the risk estimates are relatively small for all three sensitivity studies. In response to RAI 61 dated May 1, 2013 (Reference 15), the licensee provided new risk results using a PRA that used the most conservative method to assign conditional risk, and applied a reduced fire frequency compared a frequency based solely on the floor area. The NRC staff concludes that the incorporated method appropriately assigns a non-zero frequency of transient fires in areas that are difficult to access and applies a conservative (pinch point) conditional risk estimate. Based on the above, the incorporated method is consistent with NUREG/CR-6850 guidance that transient fires should be located at accessible pinch points, and is acceptable.

In response to NRC staff RAI 39 dated January 27, 2012 (Reference 66), the licensee affirmed that, other than the PRA methods identified and discussed above, there were no other methods that deviate from NUREG/CR-6850 or from NRC-approved FAQs. In response to RAI 36 dated April 27, 2012 (Reference 8), the licensee stated that no new methods are expected to be required to complete the implementation items identified in Attachment S, Table S-3 of the LAR. In response to RAI 35 dated April 27, 2012 (Reference 8), the licensee further stated that no new methods or plant modifications have been added to the PRA models that were not included in the CNP FPRA peer review. In response to RAI 45 dated October 15, 2012 (Reference 11), the licensee stated that no changes have been made to the CNP FPRA since the last full-scope peer review that would constitute a PRA upgrade per the ASME/ANS PRA standard. However, in response to subsequent RAI 61 dated May 1, 2013 (Reference 15), the licensee upgraded its PRA to correct several unacceptable methods and incorporated two new plant modifications into the PRA. The licensee also proposed license conditions that include focused-scope peer reviews of the application of the new methods and of the new models that will be completed before full implementation of the self-approval process. As described in RG 1.200, focused-scope peer reviews are an acceptable method to verify the technical adequacy of FPRAs provided, as specified in NFPA 805, that acceptable methods have been used. The total risk and change in risk results decreased substantively because of the new plant modifications to values well below the acceptance guidelines. Therefore, the NRC staff concludes that any modifications to the PRA that might arise from resolution of any new issues raised in the focused-scope peer reviews are not likely to change the results such that the currently acceptable changes in risk would become unacceptable.

Based on its review of the LAR, as supplemented, the NRC staff concludes that the CNP FPRA is technically adequate such that its quantitative results, considered together with the sensitivity study results, can be used to demonstrate that the change in risk due to the transition to NFPA 805 meets the acceptance guidelines in RG 1.174.

3.4.2.3 Fire Modeling in Support of the Development of an FPRA

The NRC staff performed detailed reviews of the fire modeling used to support the CNP FPRA in order to gain further assurance that the methods and approaches used for the application to transition to NFPA 805 (Reference 4) were technically adequate. NFPA 805 has the following requirements that pertain to fire modeling used in support of the development of an FPRA:

NFPA 805, Section 2.4.3.3: Acceptability

> The PSA approach, methods, and data shall be acceptable to the AHJ.

NFPA 805, Section 2.7.3.2: Verification and Validation

> Each calculational model or numerical method used shall be verified and validated through comparison to test results or comparison to other acceptable models.

NFPA 805, Section 2.7.3.3: Limitations of Use

> 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.

NFPA 805, Section 2.7.3.4: Qualification of Users

> Cognizant personnel who use and apply engineering analysis and numerical models (e.g., fire modeling techniques) 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.

NFPA 805, Section 2.7.3.5: Uncertainty Analysis

> An uncertainty analysis shall be performed to provide reasonable assurance that the performance criteria have been met.

The following sections discuss the results of the NRC staff's reviews of the acceptability of the fire modeling (first requirement). The results of the staff's reviews of compliance with the remaining requirements are discussed in Sections 3.8.3.2 through 3.8.3.5 of this SE.

3.4.2.3.1 Overview of Fire Models Used to Support the CNP FPRA

Fire modeling was used to develop the zone of influence (ZOI) around ignition sources in order to determine the thresholds at which a target would exceed the critical temperature or radiant heat flux. This approach provides a basis for the scoping or screening evaluation as part of the CNP FPRA. The following algebraic fire models and correlations were used for this purpose:

- Flame Height, Method of Heskestad (Reference 52, Chapter 3)
- Plume Centerline Temperature, Method of Heskestad (Reference 52, Chapter 9)

- Radiant Heat Flux, Point Source Method (Reference 52, Chapter 5)

- Ceiling Jet Temperature, Method of Alpert (Reference 78)

The first three algebraic models are described in NUREG-1805, "Fire Dynamics Tools (FDT$^s$): Quantitative Fire Hazard Analysis Methods for the US Nuclear Regulatory Commission Fire Protection Inspection Program" (Reference 52). Alpert's ceiling jet temperature correlation is described in FIVE, "EPRI Fire Induced Vulnerability Evaluation Methodology," Revision 1 (Reference 78), and serves as the basis for FDT$^s$ that are used to estimate sprinkler, smoke detector and heat detector response times as documented in NUREG-1805, Chapters 10, 11, and 12, respectively. Validation and Verification (V&V) of these algebraic models is documented in NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," Volumes 1-7 (Reference 53).

The algebraic fire models and correlations were implemented in a database and workbook referred to as the Fire Modeling Database (FMDB) and Transient Worksheet. The FMDB and Transient Worksheet also calculate the plume radius according to Heskestad's correlation described in FIVE, Revision 1 (Reference 78), but these calculations were not used in the ZOI determinations.

In addition, the licensee developed screening approaches for the evaluation of ignition sources to determine the potential for the generation of a hot gas layer (HGL) in the compartment or fire area being analyzed. The FPRA used these HGL screening approaches to further screen ignition sources, scenarios, and compartments that would not be expected to generate an HGL, and to identify the ignition sources that have the potential to generate an HGL for further analysis. The following correlations were used to determine the potential for the development of an HGL:

- Method of McCaffrey, Quintiere and Harkleroad (for naturally ventilated compartments)

- Method of Beyler (for closed compartments)

- Method of Foote, Pagni, and Alvares (for mechanically ventilated compartments)

- Method of Deal and Beyler (for mechanically ventilated compartments)

These HGL correlations are also described in NUREG-1805, Chapter 2, and implemented in the FMDB and Transient Worksheet.

In LAR Section 4.5.1.2 (Reference 6), the licensee also identified the use of the following empirical correlations that are not addressed in NUREG-1824, Volumes 3 and 4 (Reference 53).

- Sprinkler Activation Correlation (Reference 52, Chapter 10)

- Smoke Detection Actuation Correlation, Method of Heskestad and Delichatsios (Reference 52, Chapter 11)

- Heat Detection Actuation Correlation (Reference 52, Chapter 12)
- Corner and Wall Heat Release Rate (Reference 24)
- Correlation for Heat Release Rates of Cables (Reference 52, Chapter 7)
- Correlation for Flame Spread over Horizontal Cable Trays, FLASH-CAT, described in NUREG/CR-7010, "Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE), Volume 1: Horizontal Trays" (Reference 54)

The licensee's ZOI approach was used as a screening tool to distinguish between fire scenarios that required further evaluation and those that did not require further evaluation. Qualified personnel performed a plant walk-down to identify ignition sources and surrounding targets or SSCs in compartments and applied the empirical correlation screening tool to assess whether the SSCs were within the ZOI of the ignition source. Based on the fire hazard present, these generalized ZOIs were used to screen from further consideration those CNP-specific ignition sources that did not adversely affect the operation of credited SSCs, or targets, following a fire. The licensee's screening was based on the $98^{th}$ percentile fire heat release rate (HRR) from the NUREG/CR-6850 methodology (Reference 50).

The Consolidated Model of Fire and Smoke Transport (CFAST), Version 6 was used for

- Control room abandonment calculations
- Temperature sensitive equipment HGL Study

Finally, Fire Dynamics Simulator (FDS), Version 5 was used for

- HGL calculations in specific fire areas
- Temperature sensitive equipment ZOI study
- Plume/HGL interaction study
- Fire door closure calculations

Validation and Verification (V&V) of CFAST and FDS is documented in NUREG-1824, Volume 5 and Volume 7 (Reference 53).

The V&V of all correlations and fire models that were used to support the CNP FPRA is discussed in detail in Section 3.8.3.2 of this SE.

3.4.2.3.2 RAIs Pertaining to Fire Modeling in Support of the CNP FPRA

By letters dated January 27, 2012 (Reference 66) and October 11, 2012 (Reference 67), the NRC staff issued RAIs concerning the fire modeling conducted to support the CNP FPRA. By letters dated April 27, 2012 (Reference 8), and October 15, 2012 (Reference 11), the licensee provided a response to these RAIs. The following paragraphs describe selected RAI responses related to the acceptability of the fire models used.

- NRC staff issued RAI 04(a) (Reference 66) asking the licensee to explain how the input for the algebraic models was established for fires that involved multiple combustibles and justify the approach that was used.

  In response to RAI 04(a) (Reference 8), the licensee explained that the approach for fires involving multiple combustibles was to calculate the heat release rate of each individual combustible as a function of time, and then use the combined total heat release rate as the input to the algebraic models. Conservative heat release rates were determined from NUREG/CR-6850, Volume 2 (Reference 50), and the rules for propagation to cable trays and fire spread rates all followed the FLASH-CAT model found in NUREG/CR-7010 (Reference 54). The fire diameter used as the input to the algebraic models is equal to the fire diameter of the original source fire and remains unchanged throughout the burning duration of the fire. This is considered more severe for plume and flame height correlations, as the use of a small diameter results in a stronger plume and thus larger vertical ZOI values. The elevation of the fire is not changed after it propagates to secondary combustibles (i.e., at the top of cabinet ignition sources or two feet above the floor for transient ignition sources).

  Based on the explanation and justification provided in response to RAI 04(a), the NRC staff concludes that the licensee's approach to establish the algebraic model inputs for fires that involve multiple combustibles is acceptable.

- The NRC staff issued RAI 04(d) (Reference 66) asking the licensee to explain how the input parameters for fire damper fusible links were determined and how the uncertainty associated with the response characteristics (Response Time Index, or RTI, and activation temperature) was accounted for in the fire modeling analysis of scenarios involving dampers.

  In response to RAI 04(d) (Reference 8), the licensee explained that the activation temperatures are obtained from the CNP Fire Hazards Analysis. The RTI, which is normally derived experimentally, and is not readily available from the vendor. A best estimate RTI was selected based on engineering judgment and guidance provided in NUREG-1805 (Reference 52, Chapter 12). The uncertainty associated with the RTI and activation temperature of the fusible links was accounted for by running sensitivity analyses.

Based on the sensitivity analyses performed in response to RAI 04(d), the NRC staff concludes the input parameters for fire damper fusible links used in the fire modeling at CNP is acceptable.

- NRC staff issued RAI 46 (Reference 67) asking the licensee to justify the assumption that the peak HRR for transient combustibles in the Main Control Room (MCR) abandonment time study is reached in 10 minutes, instead of 8 minutes as specified in NRC FAQ 08-0052, "Transient Fire – Growth Rates and Control Room Non-Suppression" (Reference 79).

  In response to RAI 46 (Reference 11), the licensee revised the consolidated fire growth and smoke transport (CFAST) analysis for control room evacuation and demonstrated that the increased evacuation probability has a negligible effect on the risk.

  Based on the additional CFAST analysis performed in response to RAI 46, the NRC staff concludes that the abandonment time calculations for transient fires in the MCR are acceptable.

- NRC staff issued RAI 47 (Reference 67) asking the licensee to justify the assumption that, based on engineering judgment, transient combustibles need not be considered in the MCR back panel area.

  The licensee responded to RAI 47 (Reference 11) in two parts: one for the main control board (i.e., the "horseshoe"), and another for the other cabinets in the MCR (i.e., the back panels). In both cases, the additional fire frequency for a transient fire propagating into an electrical cabinet was found to be very small when compared to the probability of the cabinet fire. On that basis, the licensee concluded that any additional fire damage caused by propagation of transient fires is negligible with respect to cabinet fire frequency itself.

  Based on the response to RAI 47, the NRC staff concludes that the licensee's justification for not considering transient fires in the back panel area is acceptable.

- The MCR fire evacuation study is based on the assumption that MCR control boards do not extend to the top of the suspended ceiling. Field inspection determined that gaps are very small. The NRC staff therefore issued RAI 49 (Reference 67) asking the licensee to justify not using separate partitioned areas or obstructions in the CFAST fire modeling of the MCR.

  In response to RAI 49 (Reference 11), the licensee performed another walkdown of the MCR and confirmed that the gap is indeed significantly smaller than originally assumed. Based on this information, the MCR evacuation study was updated with the MCR separated into three separate volumes within CFAST.

As the effect on the risk of the revised MCR abandonment times calculated with the smaller gaps is found to be negligible, the NRC staff concludes that the CDF, ΔCDF, LERF, and ΔLERF originally determined on the basis of the MCR CFAST analyses with the larger gaps is acceptable.

- The MCR fire evacuation study only considers closed-door cabinets with unqualified cables. Field inspection determined that some MCR cabinets are open-backed. The NRC staff therefore issued RAI 51 (Reference 67), asking the licensee to determine the effect on plant risk of scenarios involving open cabinets.

  During the MCR walkdown referred to in the response to RAI 49 (Reference 11), the licensee determined that the main vertical control boards are open and approximately half of the "back panels" are open. Since 17 percent of the cables at CNP are unqualified, the licensee decided to perform additional calculations for MCR fire scenarios involving open cabinets with multiple bundles of qualified and unqualified cables. Fire propagation to adjacent cabinets was also modeled.

  Since it was not possible to determine the actual distribution of unqualified and qualified cable in the MCR, two cases were considered for the additional analysis: (a) 17 percent of the cabinets contain only unqualified cable and the remaining cabinets contain only qualified cable; and (b) all cabinets have 17 percent unqualified cable and conservatively use the parameters for unqualified cable for all cabinets. Case (a) increases the CDF and ΔCDF by less than 0.11 percent and less than 0.4 percent, respectively. In case (b), the increases in CDF and ΔCDF are 1 percent and 4 percent, respectively.

  Based on the results of the additional MCR abandonment calculations, the NRC staff concludes that the increase of the CDF and ΔCDF due to the consideration of fire scenarios involving open cabinets and unqualified cables in the MCR is acceptable.

- The MCR evacuation study assumes that cabinet fires propagate to adjacent cabinets after 10 minutes. This is based on the recommendations in NUREG/CR-6850 for cabinets that are separated by a single metal wall with cables in the exposed cabinet that are in direct contact with the separating wall. Field inspection revealed that the single metal partitions between cabinets have openings. The NRC staff therefore issued RAI 52 (Reference 67), asking the licensee to justify the 10-minute assumption.

  In the response to RAI 52 (Reference 11), the licensee stated that for open cabinets (separated by a single metal wall), it is unlikely that hot gases will accumulate inside the cabinet as they will flow out the openings in the back and/or top of the cabinet. Furthermore, during the MCR walkdown mentioned in the responses to RAIs 49 and 51, the licensee verified that there are no diagonal cable runs between open cabinets. Finally, the licensee referred to Appendix S of NUREG/CR-6850, which suggests that fires do not spread between cabinets when: (1) either the exposed or the exposing cabinet has an open top (2) there are internal walls, possibly with some openings, and (3) there are no diagonal cable runs between the exposing and exposed cabinets.

For closed cabinets, the licensee stated that NUREG/CR-6850 Appendix S suggests fire spread can be delayed by 15 minutes even when there is no internal barrier between the cabinets, but recommends a propagation time of 10 minutes if the cabinets are separated by a single metal wall and cables in the exposed cabinet are in contact with the wall.

Based on these observations and the guidance in NUREG/CR-6850, the licensee applied a fire propagation time of 10 minutes to all cabinets in the control room.

Based on the licensee's response to RAI 52, the NRC staff concludes that the assumption that fires propagate between cabinets in the MCR is acceptable.

- The NRC staff issued RAI 53(b) (Reference 67) asking the licensee to clarify how the wall and corner effects were accounted for in Heskestad's flame height correlation.

  The licensee responded to RAI 53(b) (Reference 11) and stated that the fire modeling at CNP did not apply the location factor to flame height calculations. This was justified on the basis that the flame height correlation was used as a reference height to determine if horizontal targets could be exposed to radiant heat. The flame height was not used to determine vertical target impacts since the plume temperature correlation provides a larger ZOI than the flame height. The radiant heat ZOI has been applied equally at all elevations of the flame height. In actuality, the point source model assumes that radiation originates from the midpoint of the flame. The radial distances to targets located above the midpoint of the flame are longer than the distances used for fire modeling at CNP. Thus the approach taken already applies conservative heat flux values. In conclusion, the licensee stated that the use of the point source flame radiation ZOI for targets at distances above the flame heights would apply unnecessary conservatism, since the emissive power of the flame at these heights would be much lower than at the midpoint of the flame.

  Based on the response to RAI 53(b), the NRC staff concludes that the licensee's justification for not applying a location factor in Heskestad's flame height correlation is acceptable.

- Based on Smokeview images of the gas temperature distribution in a horizontal slice plane at some distance below the ceiling, the reports that describes the FDS modeling in fire areas AA43 and AA44 concludes that the HGL extends a minimal distance beyond the calculated ZOI and that, therefore, whole room burnout is not achieved. Based on independent FDS runs of the non-ventilated case, the NRC staff found that the HGL seems to extend well beyond the ZOI calculated by the FDTs. The NRC staff therefore issued RAI 54(b)

(Reference 67) to request that the licensee perform a qualitative FPRA assessment using the ZOI created by the non-ventilated FDS analysis.

In response to RAI 54(b) (Reference 11), the licensee performed a sensitivity analysis to quantify the impact on the risk of using the HGL ZOI calculated with FDS for the non-ventilated case in the Unit 2 fire areas AA43 and AA44, and the corresponding Unit 1 fire areas AA40 and AA41. The total resulting increases in CDF and ΔCDF are 0.7 percent and 2.2 percent, respectively, for Unit 1 and 1.0 percent and 2.5 percent for Unit 2. LERF and ΔLERF increases, respectively, by 0.5 percent and 1.4 percent for Unit 1 and 1.0 percent and 2.6 percent for Unit 2.

Based on the results of the sensitivity study and given the conservative nature of the non-ventilated case, the NRC staff concludes that the increase of the plant CDF, ΔCDF, LERF, and ΔLERF due to the consideration of the HGL ZOI calculated with FDS is acceptable.

- The detailed fire modeling reports of several fire areas refer to the maximum expected fire scenario (MEFS) and the limiting fire scenario (LFS). The terms MEFS and LFS are typically used when fire modeling is performed to support performance-based evaluations in accordance with NFPA 805, Section 4.2.4.1. However, Section 4.5.1.2 in the LAR (Reference 6) states that "Fire modeling was performed as part of the FPRA development (NFPA 805, Section 4.2.4.2)." The NRC staff issued RAI 54(e) (Reference 67), asking the licensee to: (1) confirm that no fire modeling was performed to support compliance with NFPA 805, Section 4.2.4.1; and (2) explain how these terms were applied with regard to detailed fire modeling in support of the FPRA.

In response to RAI 54(e) (Reference 11), the licensee confirmed that no fire modeling was performed to support compliance with NFPA 805, Section 4.2.4.1. Furthermore, the licensee explained that the MEFS and LFS were used to assist in establishing safety margins, which did not directly affect the CDF and LERF calculations.

Based on the licensee's response to RAI 54(e), the NRC staff concludes that the application of MEFS and LFS in the fire modeling performed at CNP is acceptable.

- The detailed fire modeling report for fire area AA44 shows an image (FDS slice file image) of the ZOI at 120 s for the LFS and forced ventilation. The peak heat release rate in this scenario of 1462 kilowatt (kW) creates the worst-case ZOI at 1400 seconds. The NRC staff issued RAI 54(f) (Reference 67) asking the licensee to explain why the slice file image in the report pictures the ZOI at 120 seconds instead of at 1400 seconds.

In response to RAI 54(e) (Reference 11), the licensee stated that the LFS was considered solely to determine the safety margin and noted that at 120 seconds, the HRR is approximately 1200 kW. The licensee asserted that selecting 1200 kW as the LFS HRR still more than doubles the HRR of the MEFS and demonstrates an acceptable safety margin.

The NRC staff concludes that the lower safety margin for the FDS modeling in fire area AA44 of the case with forced ventilation is acceptable.

3.4.2.3.3 Conclusion for Section 3.4.2.3

Based on the licensee's description of the CNP process for performing fire modeling in support of the FPRA, the NRC staff concludes that the licensee's approach for meeting the requirements of NFPA 805, Section 2.4.3.3, is acceptable.

As discussed in Section 3.4.2.2, in response to RAIs 46, 47, 49, 51, and 54b, the licensee provided changes to the risk results caused by changes to the fire modeling assumption, which are discussed in Section 3.4.2.3.2. Taken individually, changing some of the assumptions had a negligible impact on the results while bounding assumptions for changing others increased the ΔCDF and ΔLERF by several percentage points. The NRC staff concludes that bounding assumptions that increase the quantitative risk results by several percentage points are assumptions that, if corrected, would not change the decision and, therefore, concludes that the current fire modeling evaluation is sufficient to support the requested license amendment.

3.4.2.4 Conclusions on PRA Quality

Utilizing the review process summarized in NUREG-0800, Section 19.2, Section III.2.2.4.1, the NRC staff concludes that the quality of the licensee's PRA satisfies the guidance in RG 1.174, Section 2.3, and RG 1.205, Section 4.3 regarding the technical adequacy of the PRA to support transition to NFPA 805.

The NRC staff concludes that the PRA approach, methods, and data are acceptable and Section 2.4.3.3 of NFPA 805 is satisfied for the request to transition to NFPA 805. The NRC staff based this conclusion on the findings that: (1) the PRA model for CNP meets the criteria that it adequately represents the current, as-built, as-operated configuration, and is therefore capable of being adapted to model both the post-transition and compliant plant as needed; (2) the PRA models conform sufficiently to the applicable industry PRA standards for internal events and fires at an appropriate capability category, considering the acceptable disposition of the peer review and NRC staff review findings; and (3) the fire modeling used to support the development of the CNP FPRA has been confirmed as appropriate and acceptable.

However, the self-approval acceptance guidelines are much smaller than the transition acceptance guidelines. Therefore, the NRC staff concludes that focused-scope peer reviews of the T&M mission time PRA evaluations and of the upgraded LERF FPRA models should be completed before the FPRA results are used to support risk-informed self-approval of changes to the FPP. The licensee has provided two license conditions that state that each of the reviews will be completed, and any comments resolved, before the PRA is used to support self-approval.

Finally, based on the licensee's administrative controls to maintain the PRA models current and assure continued quality, using only qualified staff and contractors (as described in Section 3.8.3 of this SE), the NRC staff concludes that PRA maintenance process is adequate to support self-approval of future risk-informed changes to the FPP following completion of the three PRA-related license conditions described in the updated Attachment M of the LAR.

### 3.4.3 Fire Risk Evaluation

For those fire areas for which the licensee used a performance-based approach to meet the NSPC, the licensee used FREs in accordance with NFPA 805, Section 4.2.4.2, to demonstrate the acceptability of the plant configuration. In accordance with the guidance in RG 1.205, Section C.2.2.4, "Risk Evaluations," the licensee used a risk-informed approach to justify acceptable alternatives to compliance with NFPA 805 deterministic criteria. The NRC staff reviewed the following information during its evaluation of CNP's FREs: LAR Section 4.5.2, "Performance Based Approaches," LAR Attachment C, "NEI 04-02 Table B-3 – Fire Area Transition," and LAR Attachment W, "Fire PRA Insights," as well as associated supplemental information.

Plant configurations that did not meet the deterministic requirements of NFPA 805, Section 4.2.3.1, were considered VFDRs. VFDRs that will be brought into deterministic compliance through plant modifications need no risk evaluation. The licensee identified 267 VFDRs in LAR Attachment C, Table B-3, "Attachment C – NEI 04-02 Table B-3 – Fire Area Transition," that it does not intend to bring into deterministic compliance under NFPA 805. For these VFDRs, the licensee performed evaluations using the risk-informed approach, in accordance with NFPA 805, Section 4.2.4.2, to address FPP non-compliances and demonstrate that the VFDRs are acceptable.

The VFDRs can generally be categorized into the following three types:

- inadequate separation resulting in fire-induced damage of process equipment or associated cables required for the identified success path
- inadequate separation resulting in fire-induced spurious operation of equipment that may defeat the identified success path
- inadequate separation resulting in fire-induced failure of process monitoring instrumentation or associated cables required for the identified success path

In response to RAI 43 dated October 15, 2012 (Reference 11), the licensee stated that none of the VFDRs involved performance-based evaluations of wrapped or embedded cables and that any such cables were credited in the FPRA as being protected from fire damage, commensurate with the fire barrier rating of the wrap or embedment.

In response to RAI 32 dated April 27, 2012 (Reference 8), the licensee explained that the delta risk associated with each VFDR is obtained by subtracting the CDF and the LERF of the post-transition plant configuration from the corresponding CDF and LERF results for a compliant plant which would be obtained if the facility was brought into compliance with Appendix R. The total delta fire risk for CNP Unit 1 and for CNP Unit 2 was obtained by summing the delta risk for each fire area in each unit and comparing the total for each unit to the acceptance guidelines contained in RG 1.174.

Some VFDRs have no risk implications (e.g., the NSCA calls for pressurizer tank heaters to help control RCS pressure but the PRA does not model operational pressure control and does not model the heaters). For each VFDR, the post-transition risk includes the risk from the fire-induced failures attributable to the existence of the VFDR. The risk of the compliant plant is defined by removing the fire-induced failures attributable to the existence of the VFDR. The post-transition risk may include the failure of recovery actions developed to mitigate the fire induced failures.

For process equipment failure VFDRs, the post-transition risk is obtained by setting the probability of failure for fire failure components to one (or True). The compliant plant risk is obtained by setting the probability of failure of this equipment back to the random failure probability. For instrumentation failure VFDRs, the compliant plant configuration is evaluated similarly if the instruments are modeled (e.g., support automatic actuation) but most instruments are not modeled in the PRA. Instrument failures are usually indirectly modeled by crediting their use to support recovery actions. In these cases, the post-transition risk is obtained by assigning 1 to any human error probability (HEP) associated with the action[3] that relied upon the instrumentation. The compliant plant risk is obtained by setting the HEP back to its original value. Other fire-induced failures that are not modeled in the PRA but have an indirect impact on risk are treated as analogous to instrument failures.

This method of calculating the delta risk was used for all fire areas except for AA46, "Unit 1 Control Room," AA47, "Unit 2 Control Room," and AA50, "Unit 1 Control Room Cable Vault and Hot Shutdown Panel Area," which utilize alternate shutdown (ASD) for fires in these fire areas. In these cases, the licensee used a bounding approach by setting the delta risk for each fire area to the total fire risk for each respective fire area (essentially setting the risk of the compliant plant to zero). This is conservative because it assumes the fire risk at a compliant plant is zero and a conservative estimate of the change in risk associated with a risk-informed change is acceptable, as described in RG 1.174.

RG 1.205, Section 2.2.4.1, and FAQ 08-0054 (Reference 91) contain guidance that directs that the change in risk between the post-transition and the compliant plant properly reflect both the post-transition and the compliant plant as-built and as-operated risks. To ensure that the risk accurately reflects the risk of both the post-transition and the compliant plant risk, the licensee noted that functions and systems (e.g., offsite power, turbine-driven auxiliary feedwater (TDAFW) pump, emergency core cooling, feed and bleed) that are not failed as a result of the fire are credited in both the post-transition and the compliant models. Therefore, the NRC staff concludes that the change in risk evaluations for each VFDR described above appropriately

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[3] These human actions are the actions to recover random (not fire-induced) failures (local and Control Room) to mitigate accident scenarios regardless of whether a fire or other initiating event caused the scenario. These human actions are modeled in the internal events PRA. Human actions taken only when the initiating event is a fire and directed toward mitigating the effects of fire-induced failures are termed recovery actions and are discussed in Section 3.4.4.

reflects the change in risk associated with retaining a VFDR instead of bringing the plant into compliance with the deterministic requirements.

3.4.4 Additional Risk Presented by Recovery Actions

For those fire areas for which the licensee used a performance-based approach to meet the NSPC, the licensee used FREs in accordance with NFPA 805, Section 4.2.4.2, to demonstrate the acceptability of the plant configuration. For many of these VFDRs, the licensee identified RAs to reduce the risk of the VFDR. NFPA 805, Section 4.2.4.2 further directs that, "when the use of recovery actions has resulted in the use of this approach, the additional risk presented by their use shall be evaluated."

The NRC staff reviewed LAR Attachment C, "NEI 04-02 Table B-3 – Fire Area Transition," LAR Attachment G, "Recovery Actions Transition," and Attachment W, "Fire PRA Insights," during its evaluation of the additional risk presented by the NFPA 805 RAs at CNP. Section 3.2.4 of this SE describes the identification and evaluation of RAs.

The licensee used the guidance in RG 1.205, Revision 1 to identify RAs. If the main control room must be abandoned (i.e., AA46, "Unit 1 Control Room," and AA47, "Unit 2 Control Room"), CNP has several Local Shutdown Indication (LSI) panels that can be used to support an approved ASD strategy. However, the licensee determined that the LSI panels did not meet the RG 1.205 definition of a Primary Control Station (PCS) since these panels only provide monitoring capability and not control capability. Therefore, the OMAs performed at the LSI panels are all RAs, and because such actions have also not been previously approved (in exemption requests), all operator actions were identified and evaluated by the licensee as new RAs.

The licensee identified RAs in the resolutions to 152 VFDRs, representing 31 of the 38 performance-based fire areas, that are necessary to meet the risk acceptance criteria (113 VFDRs) or maintain a sufficient level of DID (39 VFDRs). All RAs are described in LAR Attachment G. The licensee reviewed all of the RAs for adverse impact and dispositioned each action as stated in LAR Attachment G. None of the RAs listed in LAR Table G-1 were found to have an adverse impact on the FPRA.

For FREs, including those fire areas that utilize an ASD strategy, the post-transition risk scenarios that include an RA include the risk of failing to successfully complete the RA. The compliant plant does not include the risk of failing to complete the RA because the compliant plant would not need the action. Per NRC FAQ 07-0030 (Reference 80), one acceptable method to estimate the additional risk of RAs is to conservatively assign the total change in risk from transition for a VFDR with one or more RAs to be the additional risk of RAs in that area. The licensee chose to use this conservative method to report the additional risk of RAs.

Per the LAR, a majority of these RAs are currently credited under the existing FPP and have been verified to be feasible and reliable. However, in response to RAI 37 dated April 27, 2012 (Reference 8), the licensee clarified that CNP fire procedures will be changed during the implementation period to include changes such as "immediate actions" for time critical actions where needed and to address staffing and prioritization issues. Therefore, the licensee will perform a confirmatory demonstration of the feasibility of all RAs, including a field verification walk-through of transit times and execution times identified in implementation item S-3.8 in Attachment S, Table S-3 of the LAR. The new/revised procedures will be developed such that the RAs are reliable and the FPRA HRA will be updated following the field verification to ensure the risk metrics, which includes the additional risk of RAs, reported in the LAR, as supplemented, have not increased. The update to the HRA is identified as implementation item S-3.9 in Attachment S, Table S-3 of the LAR. The NRC staff concludes that the licensee's evaluation of NFPA 805 RAs is acceptable because additional confirmation during the implementation period will demonstrate that all RAs are feasible and reliable.

The NRC staff concludes that the licensee's approach for calculating the additional risk of RAs is acceptable because it is consistent with RG 1.205, Section 2.2.4.1, and FAQ 07-0030. The results of the delta risk calculations for each of the CNP units are summarized in Section 3.4.6 below. As described below, the results indicate that the final risk increase estimates slightly exceed the acceptance guidelines in RG 1.174, but are expected to be meet the guidelines if additional analytic effort was expended to improve the estimates. The total risk increase bounds the additional risk of operator actions and, therefore, the NRC staff concludes that the additional risk associated with RAs meets the acceptance guidelines in RG 1.174 and is acceptable.

3.4.5 Risk-Informed or Performance-Based Alternatives to NFPA 805

The licensee did not utilize any risk-informed or performance-based alternatives to compliance with NFPA 805, which falls under the requirements of 10 CFR 50.48(c)(4), at CNP.

3.4.6 Cumulative Risk and Combined Changes

The licensee identified the planned NFPA 805 transition modifications in LAR Attachment S and Section 5.4, as summarized in SE Section 2.8.1. The licensee included several modifications that were not needed to bring the facility into compliance in both the post-transition and the compliant risk estimates. Therefore, these modifications are not combined changes as discussed in Section C1-1 in RG 1.174 and the risk reduction from these changes need not be separately estimated.

The licensee reported in the LAR, as supplemented on May 1, 2013 (Reference 15), in response to RAI 61 dated February 1, 2013 (Reference 77), and as supplemented on September 16, 2013 (Reference 17), the total CDF and total LERF, which were estimated by adding the risk assessment results for internal events, fire events, and seismic events. The licensee reported that the other external hazards risks are negligible and are therefore not addressed in the total estimate. The CDF and LERF results are summarized below in Table 3.4.6-1. Based on these results, the total CDF after implementation of NFPA 805 is well below 1E-4/yr and the total LERF is well below 1E-5/yr.

Table 3.4.6-1: Total CDF and LERF Estimates for CNP after Transition to NFPA 805

| Hazard Group | Unit 1 | | Unit 2 | |
|---|---|---|---|---|
| | CDF (/year) | LERF (/year) | CDF (/year) | LERF (/year) |
| Internal Events | 1.33E-05 | 2.70E-06 | 1.32E-05 | 2.70E-06 |
| Fire Events (mean) | 3.13E-05 | 2.61E-06 | 2.64E-05 | 2.10E-06 |
| Seismic Events | 3.17E-06 | 9.82E-07 | 3.17E-06 | 9.82E-07 |
| TOTAL | 4.78E-05 | 6.29E-06 | 4.28E-05 | 5.78E-06 |

The licensee also provided in the LAR the ΔCDF and ΔLERF estimated for each fire area at each CNP unit that is not deterministically compliant in the LAR, in accordance with NFPA 805, Section 4.2.3, "Deterministic Approach." The risk estimates for these fire areas result from the completed and planned modifications and administrative controls that will be implemented as part of the transition to NFPA 805 at CNP, as well as RAs to reduce VFDR risk. The ΔCDF and ΔLERF results by fire area from the LAR are summarized in Table 3.4.6-2.

In response to RAI 61 dated May 1, 2013 (Reference 15), the licensee changed several PRA methods and added one new plant modification. The changed methods resulted in minor increases in risk and the modification resulted in a minor decrease in risk. In response to RAI 63 dated September 16, 2013 (Reference 17), the licensee provided final change in risk estimates. The estimates of ΔCDF (with the RCP seal modification removed) for Units 1 and 2 are reported as 1.7E-5/yr and 1.9E-5/yr respectively. In its letter dated May 1, 2013 (Reference 15), the licensee notes that the ΔLERF is less than 10 percent of the ΔCDF and, therefore, ΔLERF estimates one order of magnitude lower, or 1.7E-6/yr and 1.9E-6/yr respectively, can be used for the case with the RCP seal modification removed. These estimated risk increases slightly exceed the RG 1.174 guidelines of 1E-5/yr for ΔCDF and 1E-6/yr for ΔLERF. These risk values were further refined in the licensee's response to RAI 63.

In the RAI 63 response dated September 16, 2013 (Reference 17), the licensee described several plant configuration and method interactions that result in conservative results and which, if removed, are expected to reduce the increases below the acceptance guidelines. As described in the response to RAI 63, the dominant contributor to the risk increase arises from scenarios where component cooling water to the RCP seals is lost, the RCPs continue to run, and the capability to trip the RCPs from the main control room is lost. This requires an operator to go to a remote location and trip the pumps within 13 minutes and assumes failure to do so will lead directly to core damage. This short 13-minute window starts at ignition because all equipment is assumed to be failed when the fire starts although there would be some number of additional minutes before all equipment is failed while the fire is growing. During this time, the control room trip would be available and the operators would be responding to the fire. The human error probability for action at a remote location given total available time of only 13 minutes is quite high and some few extra minutes can reduce that probability substantially. Another assumption that has an impact on these scenarios is that there are a number of high likelihood failures which reduce risk (e.g., fire-induced spurious operation that remove power from the RCPs) but PRA does not traditionally include failures that reduce risk. Similarly, a lack of specific cable routing information in the cable spreading room requires the assumption that all unrouted cables fail immediately for every fire resulting in no credit being available for the two automatic suppression systems in the room although they were installed to mitigate these fires. Justifying new assumptions to address these issues would require additional analytic and review efforts and, instead, the licensee provided a sensitivity study that modified the above assumptions, which yielded change in risk estimate that meet the acceptance guidelines of RG 1.174.

The NRC staff concludes that further improvements to the fire PRA would reduce the change in risk estimates but that further reduction of the quantitative estimates from additional analytical efforts is not necessary. Based on the quantitative and qualitative evaluation performed by the licensee in the response to RAI 63, the NRC staff concludes that the risk increase associated with the transition to NFPA-805 is acceptable and meets the guidelines described in RG 1.174.

In response to RAI 61 dated May 1, 2013 (Reference 15), and RAI 63 dated September 16, 2013 (Reference 17), the licensee changed a number of methods in the LAR and developed final change in risk estimates that reflected all the changes. The impact of changing these methods individually was evaluated in earlier RAIs (identified in the response to RAI 61). NRC did not request, and the licensee did not provide, a new table of changes in risk for each area as part of the response to either RAI 61 or RAI 63. The largest increase in CDF reported in the LAR and Table 3.4.6-2 is less than 3.0E-6/yr. The largest increase in LERF reported in the LAR and Table 3.4.6-2 is less than 3.0E-7/yr. These are to be compared to acceptance guidelines of 1E-5/yr and 1E-6/yr, respectively.

The largest total increase in either CDF or LERF reported in the sensitivity studies was a 40 percent increase in LERF following modification of the LERF models. Increasing the CPT to 0.5 (and crediting one of the modifications) as reported in the response to RAI 38 dated April 27, 2012 (Reference 8), resulted in an increase in CDF of less than 15 percent and increase in LERF of less than 2 percent. The largest individual area increase in CDF and LERF reported in the sensitivity studies for RAI 34.01 dated January 14, 2013 (Reference 13), was about 1.2E-6/yr and 8E-8/yr, respectively, following the licensee's adoption of the transient fire frequency described in FAQ 12-0064 (Reference 92). The values are reported for Unit 2 area AA52. Summing these increases to those reported in Table 3.4.6-2 result in values less than 4E-6/yr and 3E-7/yr for increases in CDF and LERF, respectively. RG 1.174 provides guidelines for accepting combined change requests where risk increases larger than the guidelines may be acceptable if offset by risk decreases, such that the total change in risk is acceptable. Even in the unlikely event that synergism between sensitivity study effects within area AA52, for example, more than double the reported risk increase, the total increases in risk from transition to NFPA 805 of less than 5E-6/yr and 7E-7/yr for CDF and LERF, respectively, are less than the acceptance guidelines. The NRC staff concludes that it is unlikely that any given area would exceed the acceptance guidelines and, if any area did exceed the guidelines, the total risk increase is still less than the acceptance guidelines and therefore acceptable as part of the combined change request.

Table 3.4.6-2: Fire ΔCDF and ΔLERF Associated With Transition to NFPA 805[4]

| Fire Area | Description | Unit 1 | | Unit 2 | |
|---|---|---|---|---|---|
| | | ΔCDF (/year) | ΔLERF (/year) | ΔCDF (/year) | ΔLERF (/year) |
| AA2 | Unit 1 and Unit 2 Turbine Building, Main Steam Enclosures and Pipe Tunnels | ε[(a)] | ε[(a)] | 4.80E-07 | 1.75E-08 |
| AA3 | Unit 1 and Unit 2 Auxiliary Building and Fuel Handling Areas (El. 609 ft., 633 ft. and 650 ft.) | 1.13E-06 | 6.04E-08 | 8.51E-07 | 6.16E-08 |
| AA5/6 | Auxiliary Building (El. 587 ft.) | 2.73E-07 | 1.35E-08 | 1.86E-07 | 1.04E-08 |
| AA9 | Unit 1 Quadrant 3M & 3N Cable Tunnel (El. 596 ft.) | 3.13E-09 | 2.02E-09 | N/A[(b)] | N/A[(b)] |
| AA10 | Unit 1 Quadrant 3S Cable Tunnel (El. 596 ft.) | 5.68E-09 | 1.62E-09 | N/A[(b)] | N/A[(b)] |
| AA11 | Unit 1 Quadrant 2 Piping Tunnel (El. 591 ft.) | 2.12E-09 | 6.04E-10 | N/A[(b)] | N/A[(b)] |
| AA14 | Unit 1 CD Diesel Generator Room (El.587 ft.) | ε[(a)] | ε[(a)] | N/A[(b)] | N/A[(b)] |
| AA15 | Unit 1 AB Diesel Generator Room (El.587 ft.) | ε[(a)] | ε[(a)] | N/A[(b)] | N/A[(b)] |
| AA18 | Auxiliary Feedwater Pump Corridor (El. 591 ft.) | 1.55E-09 | 1.53E-10 | 3.52E-09 | 4.30E-10 |
| AA23 | Unit 2 CD Diesel Generator Room (El.587 ft.) | N/A[(b)] | N/A[(b)] | ε[(a)] | ε[(a)] |
| AA24 | Unit 2 AB Diesel Generator Room (El.587 ft.) | N/A[(b)] | N/A[(b)] | ε[(a)] | ε[(a)] |
| AA27 | Unit 2 Quadrant 2 Piping Tunnel (El. 591 ft.) | N/A[(b)] | N/A[(b)] | 2.12E-09 | 2.93E-10 |
| AA29 | Unit 2 Quadrant 3M & 3S Cable Tunnel (El. 596 ft.) | N/A[(b)] | N/A[(b)] | 1.61E-08 | 2.57E-09 |
| AA30 | Unit 2 Quadrant 4 Cable Tunnel (El. 596 ft.) | N/A[(b)] | N/A[(b)] | 1.66E-09 | 4.74E-11 |
| AA31 | Unit 2 Quadrant 1 Cable Tunnel (El. 596 ft.) | N/A[(b)] | N/A[(b)] | ε[(a)] | ε[(a)] |
| AA33 | Unit 2 Essential Service Water Pump Area (El. 591 ft.) | N/A[(b)] | N/A[(b)] | ε[(a)] | ε[(a)] |

4 The reported change in risk values in Table 3.4.6-2 are the values from the original LAR. In response to RAI 61 and RAI 63, the licensee changed several PRA methods and provided additional PRA analysis to remove modeling conservatisms. The changed methods resulted in minor increases in risk, and the removal of conservatisms resulted in a minor decrease in risk. The licensee stated that the risk profile as measured by the risk contributions of individual areas may change; however, the NRC staff concludes that the original results supported by the sensitivity studies demonstrate that the total increases in risk are small and acceptable.

| Fire Area | Description | Unit 1 | | Unit 2 | |
|---|---|---|---|---|---|
| | | ΔCDF (/year) | ΔLERF (/year) | ΔCDF (/year) | ΔLERF (/year) |
| AA34 | Unit 1 East Main Steam Valve Enclosure, Main Steam Line Non-Essential Service Water Valve Areas & Contractor Access Control Area (El. 612 ft.) | ε[(a)] | ε[(a)] | N/A[(b)] | N/A[(b)] |
| AA36/42 | Auxiliary Building (El. 609 ft.) | 6.07E-07 | 1.11E-08 | 1.32E-06 | 1.23E-08 |
| AA37 | Unit 1 Quadrant 2 Cable Tunnel (El. 612 ft.) | 2.77E-08 | 1.54E-09 | N/A[(b)] | N/A[(b)] |
| AA38 | Unit 2 Quadrant 2 Cable Tunnel (El. 612 ft.) | N/A | N/A | ε[(a)] | ε[(a)] |
| AA39A | Unit 1 AB Switchgear Room (El. 609 ft.) | ε[(a)] | ε[(a)] | N/A[(b)] | N/A[(b)] |
| AA40 | Unit 1 Engineered Safeguards Systems and Motor Control Center Room (El. 609 ft.) | 2.25E-06 | 1.03E-07 | N/A[(b)] | N/A[(b)] |
| AA41 | Unit 1 Emergency Power Systems Area (El. 609 ft.) | 6.28E-08 | 1.21E-08 | N/A[(b)] | N/A[(b)] |
| AA43 | Unit 2 Engineered Safeguards Systems and Motor Control Center Room (El. 609 ft.) | N/A | N/A | 6.66E-07 | 4.44E-08 |
| AA44 | Unit 2 Emergency Power Systems Area (El. 609 ft.) | N/A[(b)] | N/A[(b)] | 4.40E-08 | 8.80E-09 |
| AA45A | Unit 2 AB Switchgear Room (El. 609 ft.) | N/A[(b)] | N/A[(b)] | ε[(a)] | ε[(a)] |
| AA46 | Unit 1 Control Room (El. 633 ft.) | 1.11E-06[(c)] | 1.16E-07[(c)] | N/A[(b)] | N/A[(b)] |
| AA47 | Unit 2 Control Room (El. 633 ft.) | N/A | N/A | 1.11E-06[(c)] | 1.16E-07[(c)] |
| AA48 | Unit 1 Switchgear Rooms Cable Vault and Auxiliary Cable Vault (El. 625 ft. 10 in. and 620 ft 6 in.) | 1.19E-06 | 6.38E-08 | N/A[(b)] | N/A[(b)] |
| AA50 | Unit 1 Control Room Cable Vault and Hot Shutdown Panel Area (El. 624 ft. and 633 ft.) | 2.29E-06[(c)] | 2.95E-07[(c)] | N/A[(b)] | N/A[(b)] |
| AA51 | Unit 2 Control Room Cable Vault and Hot Shutdown Panel Area (El. 624 ft. and 633 ft.) | N/A[(b)] | N/A[(b)] | 1.43E-06 | 1.62E-07 |
| AA52 | Unit 2 Switchgear Room Cable Vault and Auxiliary Cable Vault (El. 625 ft. 10 in. and 620 ft 6 in.) | N/A[(b)] | N/A[(b)] | 2.35E-06 | 1.59E-07 |
| AA54 | Unit 1 Charging Pumps Area (El. 587 ft.) | ε[(a)] | ε[(a)] | N/A[(b)] | N/A[(b)] |

| Fire Area | Description | Unit 1 | | Unit 2 | |
|---|---|---|---|---|---|
| | | ΔCDF (/year) | ΔLERF (/year) | ΔCDF (/year) | ΔLERF (/year) |
| AA55 | Unit 2 Charging Pumps Area (El. 587 ft.) | N/A[b] | N/A[b] | ε[a] | ε[a] |
| AA56 | Unit 1 Containment | 4.93E-08 | 4.46E-12 | N/A[b] | N/A[b] |
| AA57A | Unit 1 Control Room HVAC Equipment and Computer Areas (El. 650 ft.) | 1.26E-08 | 3.58E-09 | N/A[b] | N/A[b] |
| AA57B | Unit 2 Control Room HVAC Equipment and Computer Areas El. 650 ft.) | N/A[b] | N/A[b] | 7.73E-09 | 2.20E-09 |
| AA58 | Unit 2 Containment | N/A[b] | N/A[b] | 4.51E-12 | 4.51E-12 |
| TOTAL | | 9.01E-06 | 6.85E-07 | 8.46E-06 | 5.97E-07 |

(a) In response to RAI 28 (Reference 8), the licensee explained that the symbol ε (epsilon) is used when, for all VFDRs in the fire area, the CNP FPRA does not produce a discernible change in the success criteria. In these cases, the VFDRs are evaluated to have a very small risk.
(b) In response to RAI 28 (Reference 8), the licensee explained that N/A is used when there are no VFDRs in a fire area for the unit in question. Therefore, there is no quantifiable delta risk in that fire area.
(c) For conservatism, a bounding approach was used by setting the delta risk for each fire area to the total fire risk for each respective fire area (essentially setting the risk of the compliant plant to zero).

Based on the results of the licensee's FREs, as summarized above, the risk increase for each fire area associated with transition to NFPA 805 at CNP, as well as the cumulative change in risk for all fire areas subject to a performance-based approach, is within the RG 1.174 risk acceptance guidelines of 1E-5/yr for ΔCDF and 1E-6/yr for ΔLERF for small changes. Sensitivity analyses discussed in Section 3.4.7 of this SE, which address key sources of uncertainty and the effects of PRA methods determined to be unacceptable to the NRC staff, indicate that the change in risk results will not change substantively for alternative reasonable assumptions or after replacing unacceptable with acceptable methods, respectively. Therefore, the NRC staff concludes that the licensee has satisfied RG 1.174, Sections 2.2.4 and 2.2.5, and NUREG-0800, Section 19.2 regarding acceptable changes in risk.

3.4.7 Uncertainty and Sensitivity Analyses

In response to several NRC staff RAIs, the licensee provided sensitivity analyses of the impact on the delta and total risk results of potentially key analysis assumptions. The sensitivity studies generally addressed the following types of issues: key assumptions that the NRC staff considers to have significant uncertainty, methods that the NRC staff has determined to be unacceptable or have been invalidated by recent research, the acceptability of a Capability Category I assessment for both internal events and FPRA SRs, and potential non-conservatisms in the risk analysis. The NRC staff's evaluation of each of the sensitivity analyses is provided below:

1. The fire ignition frequencies in NRC FAQ 08-0048 (Reference 76) were utilized in the CNP FPRA and reflected in the risk results reported in the LAR. In response to RAI 30 dated April 27, 2012 (Reference 8), the licensee provided a sensitivity analysis of the assumed fire ignition frequencies using the fire ignition frequency values in NUREG/CR-6850 (Reference 50) for those ignition frequency bins having an alpha factor less than one as described in NRC FAQ 08-0048. The sensitivity analysis was performed for only ΔCDF and ΔLERF because the licensee expressed "high confidence" that the total CDF and total LERF would not move above the risk acceptance guidelines for Region II (small change) in RG 1.174. The sensitivity analysis resulted in the ΔCDF for both CNP, Units 1 and 2, slightly exceeding 1E-5/year, the risk acceptance guideline for Region II (small change) in RG 1.174. In response to subsequent RAI 61 dated May 1, 2013 (Reference 15), the licensee reported new risk (and change in risk) estimates that corrected several methods and credited two new plant modifications. Given the new PRA model, the sensitivity study required by FAQ 08-0048 would not increase the change in risk results above the acceptance guidelines.

2. The CNP FPRA assumes the NUREG/CR-6850 (Reference 50) hot short probabilities, which are reflected in the risk results reported in the LAR. In the NRC staff's RAI 38 dated October 11, 2012 (Reference 67), the staff noted that recent fire testing has resulted in a Phenomena Identification and Ranking Table (PIRT) Panel conclusion that the NUREG/CR-6850 credit of a factor of two reduction in the hot short probabilities for circuits protected by a control power transformer (CPT) could not be reproduced. In response to the RAI dated April 27, 2012 (Reference 8), the licensee provided several sensitivity analyses of the assumed hot short probabilities. One of these analyses, where the hot short probability was increased to 0.5 for all circuits where a CPT was credited and a new plant modification was credited, increased the change in CDF by about 14 percent and the change in LERF by about 1 percent. In response to subsequent RAI 61 dated May 1, 2013 (Reference 15), the licensee reported new risk (and change in risk) estimates that used the 0.5 CPT credit, corrected several other methods, and credited two new plant modifications.

3. The licensee performed a gap assessment of the CNP Internal Events PRA against the requirements of the ASME RA-Sa-2009 PRA Standard. The

assessment determined that LERF SRs LE-B1, LE-B2, LE-C1, LE-C2, LE-C3, LE-C4, LE-C10, LE-C12, LE-E2, and LE-E3 meet Capability Category I because the CNP Internal Events PRA and FPRA use the generic containment event tree (CET) and containment failure probabilities in NUREG/CR-6595, Revision 0 (Reference 81), rather than plant-specific analyses and the CNP LERF model does not use the failure probabilities recommended in NUREG/CR-6595, Revision 1 (Reference 58), for ice condenser containments, which is non-conservative. Furthermore, the gap assessment determined that SR LE-D6 meets Capability Category-I because the steam generator tube conditional failure probabilities from NUREG-1570 (Reference 59) were used rather than plant-specific values and the CNP LERF model does not account for possible depressurized steam generators, which is a non-conservative treatment of thermally-induced steam generator tube rupture (TI-SGTR) events. In addressing these deficiencies, the licensee provided in response to RAI 20 (Reference 10), a sensitivity analysis that utilized the containment failure probabilities recommended by NUREG/CR-6595, Revision 1, for ice condenser containments and increased containment bypass probabilities to account for severe accident-induced TI-SGTR. While the sensitivity study resulted in increased total and ΔLERF values of 30-40 percent, the total LERF for both CNP, Units 1 and 2, remained well below 1E-5/yr and the ΔLERF for both CNP, Units 1 and 2, remained slightly below the risk acceptance guideline of 1E-6/yr for Region II (small change) in RG 1.174. In response to subsequent RAI 61 (Reference 15), the licensee reported new risk (and change in risk) estimates that used the new LERF estimates, corrected several other methods, and credited two new plant modifications.

4. Peer review Finding FSS-D7-1 evaluated SR FSS-D7 as meeting Capability Category I because generic values for fire protection system unavailability from NUREG/CR-6850 (Reference 50) were utilized in the CNP FPRA, rather than plant-specific data, and an assessment of plant-specific outlier behavior was not performed. In response to NRC staff RAI 34.d dated August 9, 2012 (Reference 10), the licensee provided a sensitivity analysis of actual unavailable times for credited fire detection and suppression systems where it was determined the actual times were greater than the generic values used in the FPRA. The results of the analysis showed an increase in total and ΔCDF and LERF of less than one percent, with one exception where the ΔCDF for CNP Unit 1 increased by 1.1 percent. Based on these results, the total CDF and LERF for both CNP, Units 1 and 2, reported in response to RAI 61 dated May 1, 2013 (Reference 15), remain well below 1E-4/yr for CDF and 1E-5/yr for LERF. In addition, the ΔCDF and ΔLERF for both CNP, Units 1 and 2, reported in response to RAI 61 (Reference 15) remain well below the risk acceptance guidelines of 1E-5/yr for CDF and 1E-6/yr for LERF for Region II (small change) in RG 1.174, which is acceptable to the NRC staff. The small increase in the change in risk results illustrated in this sensitivity study would not increase the change in risk results above the acceptance guidelines. Therefore, the licensee's proposal to update the uncertainties as part of its general PRA update process is acceptable.

5. Peer review Findings IGN-A7-1, IGN-A7-2, and IGN-A7-3 identified deviations to the NUREG/CR-6850 (Reference 50) methodology for assigning maintenance, occupancy, and storage influence factors in apportioning the transient fire frequency among fire zones. Findings (IGN-A7-1, IGN-A7-2, and IGN-A7-3) identified deviations to the NUREG/CR-6850 (Reference 50) methodology for assigning maintenance, occupancy, and storage influence factors in apportioning the cable fires caused by welding and cutting, transient fires caused by welding and cutting, and general transient fire frequencies among fire zones. In response to NRC staff RAI 34.e dated August 9, 2012 (Reference 10), the licensee provided several sensitivity analysis in which the maintenance, occupancy, and storage influence factors were modified. One sensitivity study used the apportionment process endorsed by the NRC in FAQ 12-0064. In response to subsequent RAI 61 dated May 1, 2013 (Reference 15), the licensee reported new risk (and change in risk) estimates that utilized the FAQ 12-0064 method, corrected several other methods, and credited two new plant modifications.

6. During the NRC staff audit of the licensee's NFPA 805 LAR, the NRC staff observed based on plant walkdowns that transient fires were not postulated in certain plant locations based on the determination that the space was inaccessible and therefore transient fires were highly improbable. The NRC staff requested in RAI 40 dated October 11, 2012 (Reference 67), that the licensee postulate transient fires in all locations where a pinch point can be threatened. In response to RAI 40 dated October 15, 2012 (Reference 11), the licensee provided a sensitivity analysis in which additional transient fires were postulated in locations in fire areas AA48 and AA52 where the CNP FPRA assumed transient fires were improbable due to inaccessibility, and conservative (pinch point) conditional risk values were postulated. Three sensitivity cases are presented and, in all cases, the increase in risk and change in risk was no larger than 10 percent. In response to RAI 61 dated May 1, 2013 (Reference 15), the licensee reported new risk (and change in risk) estimates that incorporates the method that yields results between the two extremes, corrected several other methods, and credited two new plant modifications.

7. In response to RAI 46 dated October 15, 2012 (Reference 11), the licensee provided a sensitivity analysis of the impact of assuming transient combustible fires reach the peak heat release rate (HRR) after 10 minutes rather than the recommended 8 minutes in FAQ 08-0052 (Reference 79). The licensee stated that this assumption was only used in determining the probability of control room abandonment (due to habitability) for postulated transient fire in the main control room (MCR). The sensitivity study resulted in a factor of 8 increase in the control room evacuation probability, which negligibly impacts the total and ΔCDF and LERF results reported in the LAR. In response to subsequent RAI 61 dated May 1, 2013 (Reference 15), the licensee reported new risk (and change in risk) estimates that used 8 minutes instead of 10 minutes, corrected several other methods, and credited two new plant modifications.

8. In response to RAI 51 dated October 15, 2012 (Reference 11), the licensee provided a sensitivity analysis of the impact of assuming closed door cabinets with unqualified cables when modeling electrical cabinet fires. The licensee stated that the issue of open/closed cabinets is only relevant to MCR fire scenarios where plant walkdowns determined that the main control boards (MCBs) and approximately half of the back panels are open. In the sensitivity study, the licensee added fire scenarios involving open cabinet fires (for both MCBs and MCR back panels). In addition, the sensitivity study considered a bounding case scenario in which all of the cables in all of the cabinets were conservatively assumed to be unqualified. This bounding sensitivity study resulted in a total CDF increase of 1 percent and a 4 percent increase in ΔCDF. In response to subsequent RAI 61 dated May 1, 2013 (Reference 15), the licensee reported new risk (and change in risk) estimates that used different HRRs for qualified versus unqualified cables, corrected several other methods, and credited two new plant modifications.

9. In response to RAI 54.b dated October 15, 2012 (Reference 11), the licensee provided a sensitivity analysis of the impact of assuming that hot gas layer (HGL) formation in fire areas AA40, AA41, AA43, and AA44 do not significantly extend the zone of influence (ZOI) for these fire areas. In the sensitivity study, the licensee extended the ZOI assumed in the CNP FPRA for these fire areas, which was calculated by the fire dynamics tools (FDTs), to use the HGL ZOI from non-ventilated Fire Dynamics Simulator (FDS) simulations. The sensitivity study resulted in a total CDF and LERF increase of 1 percent or less and a maximum ΔCDF and LERF increase of 2.6 percent. Therefore, the total CDF and LERF for both CNP, Units 1 and 2, remain well below 1E-4/yr for CDF and 1E-5/yr for LERF. In addition, the ΔCDF and ΔLERF for both CNP, Units 1 and 2, remain below the risk acceptance guidelines of 1E-5/yr for CDF and 1E-6/yr for LERF for Region II (small change) in RG 1.174, which is acceptable to the NRC staff.

10. During the NRC staff audit of the licensee's NFPA 805 LAR, the NRC staff observed through viewing the large models that the FPRA did not fail emergency core cooling system (ECCS) for fire scenarios that resulted in loss of both trains of RWST indication, but rather assumed that ECCS via switchover to containment sump recirculation was available using other indications that were not credited in the FPRA. In response to related RAIs 26 and 27 dated April 27, 2012 (Reference 8), and to subsequent RAI 61 dated May 1, 2013 (Reference 15), the licensee reported new risk (and change in risk) estimates that appropriately incorporated the loss of RWST water level indication into the model.

Based on the results of the above sensitivity analyses, the licensee has demonstrated that most unacceptable assumptions have a negligible to small impact on the risk analyses. In most cases, the licensee has modified its analyses as described above to incorporate acceptable models into its PRA. The results of the modified analysis reported in the response to RAI 61 dated May 1, 2013 (Reference 15), demonstrates that the total CDF and LERF for both CNP, Units 1 and 2, remain below 1E-4/yr for CDF and 1E-5/yr for LERF. In addition, the ΔCDF and

ΔLERF for both CNP, Units 1 and 2, are expected to be below the risk acceptance guidelines of 1E-5/yr for CDF and 1E-6/yr for LERF for Region II (small changes) in RG 1.174, which demonstrates that any change in risk is small and acceptable.

3.4.8 Conclusion for Section 3.4

Based on the NRC staff's review of the information provided by the licensee in the LAR, as supplemented, the NRC staff concludes that the PRA approach, methods, tools, and data are acceptable and, therefore, NFPA 805, Sections 2.4.3, 2.4.4, and 4.2.4.2 are satisfied. Specifically:

- The evaluations with regard to DID and safety margins are acceptable because the licensee's process and results are consistent with the endorsed guidance in NEI 04-02, Revision 2. They are also consistent with the NRC staff guidance in RG 1.205, Revision 1, and RG 1.174, Revision 1.

- The internal events and FPRA models for CNP represent the current, as built, as operated configuration, and are capable of being adapted to model both the post-transition and compliant plant as needed. The licensee has a program for ensuring that developers and users of these models are appropriately trained and qualified and therefore the licensee should be capable of maintaining the FPRA to support post-transition FREs in support of the self-approval process.

- The licensee's FPRA results, supported by the sensitivity study results, are technically adequate to support the FREs required for the NFPA 805 application.

- The NRC PRA maintenance process is adequate to support self-approval of future risk-informed changes to the FPP following completion of the two PRA-related licensee conditions described in the updated Attachment M of the LAR.

- The evaluations of each VFDR appropriately reflect the change in risk associated with retaining a VFDR instead of bringing the plant into compliance with the deterministic requirements. The analyses, assumptions, and approximations used to map the cause-effect relationship associated with the NFPA 805 application are technically adequate and acceptable.

- The licensee's approach for calculating the additional risk of RAs is acceptable because the approach is consistent with that described in RG 1.205, Section 2.2.4.1, and FAQ 07-0030. The results demonstrate that the total risk of transition is less than the risk acceptance guidelines in RG 1.174. Therefore, the NRC concludes that the additional risk associated with RAs is acceptable.

- The changes in risk (i.e., ΔCDF and ΔLERF) associated with the proposed alternatives to comply with the deterministic criteria of NFPA 805 (FREs) are acceptable for this application. The licensee has demonstrated that it could satisfy the guidance regarding acceptable risk contained in RG 1.205, Revision 1, RG 1.174, Sections 2.2.4 and 2.2.5, and NUREG-0800, Section 19.2, regarding acceptable risk. Therefore, the NRC staff concludes that the changes in risk are acceptable and meet the requirements of NFPA 805.

- The cumulative risk of the bundled plant changes for NFPA 805 transition, including all the RAs, meets the acceptance criteria in RG 1.205, Revision 1.

- The licensee did not utilize any risk-informed or performance-based alternatives to compliance to NFPA 805 that would fall under the requirements of 10 CFR 50.48(c)(4) and, therefore, any additional justification is not required.

## 3.5 Nuclear Safety Capability Assessment Results

NFPA 805, Section 2.2.3, "Evaluating Performance Criteria," states the following:

> To determine whether plant design will satisfy the appropriate performance criteria, an analysis shall be performed on a fire area basis, given the potential fire exposures and damage thresholds, using either a deterministic or performance-based approach.

NFPA 805, Section 2.2.4, "Performance Criteria," states the following:

> The performance criteria for nuclear safety, radioactive release, life safety, and property damage/business interruption covered by this standard are listed in Section 1.5 and shall be examined on a fire area basis.

NFPA 805, Section 2.2.7, "Existing Engineering Equivalency Evaluations," states:

> When applying a deterministic approach, the user shall be permitted to demonstrate compliance with specific deterministic fire protection design requirements in Chapter 4 for existing configurations with an engineering equivalency evaluation. These existing engineering evaluations shall clearly demonstrate an equivalent level of fire protection compared to the deterministic requirements.

### 3.5.1 Nuclear Safety Capability Assessment Results by Fire Area

NFPA 805, Section 2.4.2, "Nuclear Safety Capability Assessment (NSCA)," states the following:

> The purpose of this section is to define the methodology for performing a NSCA. The following steps shall be performed:
>
> (1) Selection of systems and equipment and their interrelationships necessary to achieve the nuclear safety performance criteria (NSPC) in Chapter 1
>
> (2) Selection of cables necessary to achieve the NSPC in Chapter 1

(3) Identification of the location of nuclear safety equipment and cables

(4) Assessment of the ability to achieve the nuclear safety performance criteria given a fire in each fire area.

This section of the SE addresses the last topic regarding the ability of each fire area to meet the NSPC of NFPA 805. Section 3.2.1 of this SE addresses the first three topics.

NFPA 805, Section 2.4.2.4, "Fire Area Assessment," also states the following:

> An engineering analysis shall be performed in accordance with the requirements of Section 2.3 for each fire area to determine the effects of fire or fire suppression activities on the ability to achieve the NSPC of Section 1.5.

In accordance with the above, the process defined in NFPA 805, Chapter 4, provides a framework to select either a deterministic or a PB approach to meet the NSPC. Within each of these approaches, additional requirements and guidance provide the information necessary for the licensee to perform the engineering analyses necessary to determine which fire protection systems and features are required to meet the NSPC of NFPA 805.

NFPA 805, Section 4.2.2, "Selection of Approach," states the following:

> For each fire area either a deterministic or performance based approach shall be selected in accordance with Figure 4.2.2. Either approach shall be deemed to satisfy the nuclear safety performance criteria. The performance based approach shall be permitted to utilize deterministic methods for simplifying assumptions within the fire area.

This section of the SE evaluates the approach used to meet the NSPC on a fire area basis, as well as what fire protection features and systems are required to meet the NSPC.

The NRC staff reviewed the LAR (Reference 6) Section 4.2.4, "Fire Area Transition," Section 4.8.1, "Results of the Fire Area Review," Attachment C, "NEI 04-02 Table B-3 – Fire Area Transition," Attachment G, "Recovery Actions Transition," Attachment K, "Existing Licensing Action Transition," and Attachment S, "Plant Modifications and Items to be Completed During Implementation," during its evaluation of the ability of each fire area to meet the NSPC of NFPA 805.

CNP is a two-unit plant with unit-specific as well as common fire areas. The plant is divided into 57 fire areas, including the yard, and each fire area is composed of multiple fire zones. Based on the information provided by the licensee in the LAR, the licensee performed the (NSCA) on a fire area basis for each of the 57 fire areas. LAR Attachment C provides the results of these analyses on a fire area basis and also identifies the individual fire zones within the fire areas.

For each fire area, whether deterministic or PB, the licensee documented the following:

- The approach used in accordance with NFPA 805 (i.e., the deterministic approach in accordance with NFPA 805, Section 4.2.3, or the PB approach in accordance with NFPA 805, Section 4.2.4).
- The SSCs required to meet the NSPC.
- The fire detection and suppression systems required to meet the NSPC.
- An evaluation of the effects of fire suppression activities on the ability to achieve the NSPC.
- The previously approved licensing actions that are being transitioned into the NFPA 805 RI/PB FPP.
- The licensee's existing engineering equivalency evaluations (EEEEs) used to demonstrate an equivalent level of fire protection to specific NFPA 805, Section 4.2.3 deterministic requirements.

In addition to the above, the licensee documented the following for each performance-based fire area:

- The disposition of each variance from the deterministic requirements (VFDR) using FREs.
- The RAs credited in the resolution to VFDRs to meet either the risk acceptance guidelines or defense-in-depth criteria.
- Modifications performed to bring the VFDRs into compliance with the deterministic requirements or to reduce risk.

Table 3.5-1 below identifies the compliance basis for each fire area or, in other words, those fire areas that were analyzed using either the deterministic (19 fire areas) or PB approach (38 fire areas) in accordance with NFPA 805, Sections 4.2.3 and 4.2.4, respectively. This table is based on the information provided in LAR Attachment C Table B-3, "Fire Area Transition."

Table 3.5-1: CNP Fire Areas and Compliance Strategy

| Fire Area | Area Description | NFPA 805 Compliance Basis |
|---|---|---|
| AA1 | Unit 1 and Unit 2 Residual Heat Removal and Containment Spray Pump Area (El. 573 ft.) | Deterministic |
| AA2 | Unit 1 and Unit 2 Turbine Building, Main Steam Enclosures and Pipe Tunnels | Performance-Based |

| Fire Area | Area Description | NFPA 805 Compliance Basis |
|---|---|---|
| AA2C | Unit 1 and Unit 2 Sub-Basement and Essential Service Water Pipe Tunnels | Deterministic |
| AA3 | Unit 1 and Unit 2 Auxiliary Building and Fuel Handling Areas (El. 609 ft., 633 ft. and 650 ft.) | Performance-Based |
| AA5/6 | Auxiliary Building (El. 587 ft.) | Performance-Based |
| AA7 | Unit 1 Quadrant 1 Cable Tunnel (El. 596 ft.) | Deterministic |
| AA8 | Unit 1 Quadrant 4 Cable Tunnel (El. 596 ft.) | Deterministic |
| AA9 | Unit 1 Quadrant 3M & 3N Cable Tunnel (El. 596 ft.) | Performance-Based |
| AA10 | Unit 1 Quadrant 3S Cable Tunnel (El. 596 ft.) | Performance-Based |
| AA11 | Unit 1 Quadrant 2 Piping Tunnel (El. 591 ft.) | Performance-Based |
| AA12 | Unit 1 Diesel Generator Oil Pump Room (El. 587 ft.) | Deterministic |
| AA13 | Unit 1 Transformer Room (El. 591 ft.) | Deterministic |
| AA14 | Unit 1 CD Diesel Generator Room (El. 587 ft.) | Performance-Based |
| AA15 | Unit 1 AB Diesel Generator Room (El. 587 ft.) | Performance-Based |
| AA16 | Unit 1 West Motor Driven Auxiliary Feedwater Pump Room (El. 591 ft.) | Deterministic |
| AA17 | Unit 2 West Motor Driven Auxiliary Feedwater Pump Room (El. 591 ft.) | Deterministic |
| AA18 | Auxiliary Feedwater Pump Corridor (El. 591 ft.) | Performance-Based |
| AA19 | Unit 1 East Motor Driven Auxiliary Feedwater Pump Room (El. 591 ft.) | Deterministic |
| AA20 | Unit 1 Turbine Driven Auxiliary Feedwater Pump Room (El. 591 ft.) | Deterministic |
| AA21 | Unit 2 Turbine Driven Auxiliary Feedwater Pump Room (El. 591 ft.) | Deterministic |
| AA22 | Unit 2 East Motor Driven Auxiliary Feedwater Pump Room (El. 591 ft.) | Deterministic |
| AA23 | Unit 2 CD Diesel Generator Room (El. 587 ft.) | Performance-Based |
| AA24 | Unit 2 AB Diesel Generator Room (El. 587 ft.) | Performance-Based |
| AA25 | Unit 2 Transformer Room (El. 591 ft.) | Deterministic |
| AA26 | Unit 2 Diesel Generator Oil Pump Room (El. 587 ft.) | Deterministic |
| AA27 | Unit 2 Quadrant 2 Piping Tunnel (El. 591 ft.) | Performance-Based |
| AA29 | Unit 2 Quadrant 3M & 3S Cable Tunnel (El. 596 ft.) | Performance-Based |
| AA30 | Unit 2 Quadrant 4 Cable Tunnel (El. 596 ft.) | Performance-Based |
| AA31 | Unit 2 Quadrant 1 Cable Tunnel (El. 596 ft.) | Performance-Based |
| AA32 | Unit 1 Essential Service Water Pump Area and Unit 1 and Unit 2 Basement Motor Control Center Room (El. 591 ft. and 575 ft.) | Deterministic |
| AA33 | Unit 2 Essential Service Water Pump Area (El. 591 ft.) | Performance-Based |
| AA34 | Unit 1 East Main Steam Valve Enclosure, Main Steam Line Non-Essential Service Water Valve Areas & Contractor Access Control Area (El. 612 ft.) | Performance-Based |
| AA35 | Unit 2 East Main Steam Valve Enclosure, Main Steam Line Non-Essential Service Water Valve Areas & Contractor Access Control Area (El. 612 ft.) | Deterministic |

| Fire Area | Area Description | NFPA 805 Compliance Basis |
|---|---|---|
| AA36/42 | Auxiliary Building (El. 609 ft.) | Performance-Based |
| AA37 | Unit 1 Quadrant 2 Cable Tunnel (El. 612 ft.) | Performance-Based |
| AA38 | Unit 2 Quadrant 2 Cable Tunnel (El. 612 ft.) | Performance-Based |
| AA39A | Unit 1 AB Switchgear Room (El. 609 ft.) | Performance-Based |
| AA39B | Unit 1 CD Switchgear Room (El. 609 ft.) | Deterministic |
| AA40 | Unit 1 Engineered Safeguards Systems and Motor Control Center Room (El. 609 ft.) | Performance-Based |
| AA41 | Unit 1 Emergency Power Systems Area (El. 609 ft.) | Performance-Based |
| AA43 | Unit 2 Engineered Safeguards Systems and Motor Control Center Room (El. 609 ft.) | Performance-Based |
| AA44 | Unit 2 Emergency Power Systems Area (El. 609 ft.) | Performance-Based |
| AA45A | Unit 2 AB Switchgear Room (El. 609 ft.) | Performance-Based |
| AA45B | Unit 2 CD Switchgear Room (El. 609 ft.) | Deterministic |
| AA46 | Unit 1 Control Room (El. 633 ft.) | Performance-Based |
| AA47 | Unit 2 Control Room (El. 633 ft.) | Performance-Based |
| AA48 | Unit 1 Switchgear Rooms Cable Vault and Auxiliary Cable Vault (El. 625 ft. 10 in. and 620 ft 6 in.) | Performance-Based |
| AA50 | Unit 1 Control Room Cable Vault and Hot Shutdown Panel Area (El. 624 ft. and 633 ft.) | Performance-Based |
| AA51 | Unit 2 Control Room Cable Vault and Hot Shutdown Panel Area (El. 624 ft. and 633 ft.) | Performance-Based |
| AA52 | Unit 2 Switchgear Room Cable Vault and Auxiliary Cable Vault (El. 625 ft. 10 in. and 620 ft 6 in.) | Performance-Based |
| AA54 | Unit 1 Charging Pumps Area (El. 587 ft.) | Performance-Based |
| AA55 | Unit 2 Charging Pumps Area (El. 587 ft.) | Performance-Based |
| AA56 | Unit 1 Containment | Performance-Based |
| AA57A | Unit 1 Control Room HVAC Equipment and Computer Areas (El. 650 ft.) | Performance-Based |
| AA57B | Unit 2 Control Room HVAC Equipment and Computer Areas (El. 650 ft.) | Performance-Based |
| AA58 | Unit 2 Containment | Performance-Based |
| YD | Yard | Deterministic |

3.5.1.1 Plant Systems and Equipment required to meet NSPC

The licensee performed an NSCA in accordance with NFPA 805, Section 2.4.2 to identify systems and components that establish a success path, free of fire damage that is necessary to achieve and maintain the NSPC as required by NFPA 805, Sections 1.5.1 and 4.2.1. The licensee's methodology for performance of the NSCA is reviewed in Section 3.2 of this SE. Based on the licensee's analysis, LAR Attachment C identifies on a fire area basis, the major systems or equipment that define the success path necessary to accomplish the NSPC, which include: (1) reactivity control, (2) inventory and pressure control, (3) decay heat removal,

(4) vital auxiliaries, and (5) process monitoring. Where systems and components necessary to meet the NSPC do not meet the deterministic criteria of NFPA 805, Section 4.2.3, VFDRs are identified for the specific NSPC and each VFDR is evaluated for impact on risk, DID, and safety margins. VFDRs are evaluated in Section 3.5.1.8 of this SE.

Based on the statements provided in LAR Attachment C, as supplemented, the NRC staff concludes that the CNP treatment of this issue is acceptable because the licensee has adequately identified the systems and equipment associated with the success path for each fire area that is necessary to achieve and maintain the NSPC of NFPA 805.

3.5.1.2 Fire Detection and Suppression Systems Required to meet the NSPC

A primary purpose of NFPA 805, Chapter 4 is to determine, by analysis, what fire protection features and systems need to be credited to meet the NSPC. Four sections of NFPA 805, Chapter 3, have requirements dependent upon the results of the engineering analyses performed in accordance with NFPA 805, Chapter 4: (1) fire detection systems, in accordance with Section 3.8.2; (2) automatic water-based fire suppression systems, in accordance with Section 3.9.1; (3) gaseous fire suppression systems; in accordance with Section 3.10.1; and (4) passive fire protection features, in accordance with Section 3.11. The features and systems addressed in these sections that are only required when the analyses performed in accordance with NFPA 805, Chapter 4 indicate the features and systems are required to meet the NSPC.

The licensee performed a detailed analysis of fire protection features and identified the fire suppression and detection systems required to meet the NSPC for each fire area. LAR Table 4-3, "Summary of NFPA 805, Compliance Basis and Required Fire Protection Systems and Features," lists the fire areas and fire zones at CNP, identifies if automatic fire suppression and detection systems are installed in these areas/zones, and identifies the fire detection and suppression systems that are required in each area/zone and the basis for why the system is required. Fire detection and suppression systems were determined to be required by the licensee based on one or more of the following: (1) required to meet the separation criteria in NFPA 805, Section 4.2.3; (2) required for acceptability of previously NRC-approved licensing actions; (3) required for acceptability of existing engineering equivalency evaluations (EEEEs); (4) required to meet the risk criteria in the performance-based approach of NFPA 805, Section 4.2.4; or (5) required to maintain an adequate balance of DID in the performance-based approach of NFPA 805, Section 4.2.4.

Based on the statements provided in LAR Attachment C, as supplemented, the NRC staff concludes that the CNP treatment of this issue is acceptable because the license adequately identified the fire detection and suppression systems in each fire area/zone that are required to meet the NSPC of NFPA 805.

3.5.1.3 Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

Section 4.2.4.2 of NFPA 805 requires an assessment of each fire area that includes an analysis of the effects of fire suppression activities on the ability to achieve the NSPC. For each fire area in LAR Attachment C, the licensee provided a summary discussion of its analysis of the impact of manual and (where provided) fixed suppression effects on plant equipment as well as the mitigating features such as cabinet seals, cabinet design, floor drains, and fire brigade training. No impacts on the ability to achieve the NSPC were identified.

Based on the information provided by the licensee in LAR Attachment C, the licensee has evaluated fire suppression effects on meeting the NSPC and determined that fire suppression activities will not adversely affect achievement of the NSPC. The NRC staff has reviewed the information provided by the licensee in the LAR and, on this basis, concludes that the licensee's evaluation of the suppression effects on the NSPC is acceptable.

3.5.1.4 Plant Fire Barriers and Separations

Passive fire protection features (e.g., fire barriers, through penetration fire stops, and penetration seals) and active fire protection features (e.g., doors, dampers, and water curtains) include the fire barriers and the associated elements used to form fire area boundaries and barriers separating success paths necessary to meet the NSPC. The fire barrier fire-resistance rating necessary for separation between fire areas under NFPA 805 (i.e., 3 hours) is the same as that necessary under the plant's pre-NFPA 805 licensing basis. Where the fire barriers do not meet the required fire-resistance rating, the licensee has performed EEEEs on the acceptability of the barrier relative to the hazards in the fire area, as discussed in Section 3.5.1.7 of this SE.

In addition to those established fire barriers and separations that define the plant fire areas, passive fire protection features may include such design elements or features as radiant energy shields, flame impingement shields, high-energy arcing fault (HEAF) shields, and electrical raceway fire barrier systems (ERFBS) that are credited with protecting cables, electrical components, and equipment within a fire area from the effects of fire or high-energy faults.

LAR Table 4-3, "Summary of NFPA 805, Compliance Basis and Required Fire Protection Systems and Features," identifies equipment or passive fire protection features that are required to meet NFPA 805 separation criteria. With the exception of ERFBS, which are addressed in Section 3.5.1.5 below, the only equipment or passive fire protection features identified in Table 4-3 are radiant energy shields in fire areas AA56 and AA58, Unit 1 and Unit 2 Containments, respectively that provide protection of instrumentation in the Containment Instrumentation Rooms for each unit. These radiant energy shields were installed under the pre-NFPA 805 licensing basis and are identified in LAR Table 4-3, as necessary, to meet risk and separation requirements for compliance with NFPA 805. VFDRs AA56-002 and AA58-002 credit the radiant energy shields with protection of process monitoring capability.

LAR Table 4-3 also identifies an "Intra-Fire Area Barrier" in fire area AA36/42, Fire Zone 44S, as a required feature to meet separation and risk criteria for redundant component cooling water (CCW) pumps. This fire barrier is a 3-hour rated, partial height (6-ft) wall separating the redundant CCW pumps, was previously approved by the NRC as part of the basis for approval of an Appendix R exemption request. While this previously-approved exemption is not being transitioned, the fire barrier feature continues to be credited in the performance-based analysis for the fire area.

The acceptability of fire barriers and separations is evaluated as part of the NRC staff's review of LAR Attachment A, Table B-1. Section 3.1 of this SE also provides the results of the NRC staff's evaluation of the acceptability of CNP fire barriers and separations against the NFPA 805, Section 3.11, minimum design requirements for these fire protection features.

3.5.1.5 Electrical Raceway Fire Barrier Systems (ERFBS)

LAR Table 4-3 identifies the fire areas that credit ERFBS as a fire protection feature. Fire areas utilizing ERFBS include AA2, AA14, AA24, AA32, AA39A, and AA45A. These fire areas were evaluated using the performance-based approach of NFPA 805, Section 4.2.4.2, with the exception of fire area AA32, which meets the deterministic criteria of NFPA 805, Section 4.2.3.3(c). NFPA 805, Section 3.11.5, "Electrical Raceway Fire Barrier Systems (ERFBS)," requires that ERFBS be capable of resisting the fire effects of the hazards in the area. The ERFBS must also be tested in accordance with, and meet the acceptance criteria of Supplement 1, "Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Safe Shutdown Trains Within the Same Fire Area," dated March 25, 1994 (Reference 82), to GL 1986-10, "Implementation of Fire Protection Requirements," dated April 24, 1986 (Reference 83). CNP does not use either the Hemyc™ or MT™ ERFBS. Therefore, the generic issue (GL 2006-03) (Reference 60) related to these ERFBS is not applicable to CNP.

Each fire area utilizing ERFBS, as identified in LAR Attachment A and C includes a discussion of the VFDR analysis used to evaluate the acceptability of this feature and indicates that the fire area is in deterministic compliance. The results of the NRC staff's evaluation of the acceptability of the ERFBS at CNP against the NFPA 805, Chapter 3, Element 3.11.5 minimum design requirements for ERFBS is provided in Section 3.1 of this SE.

3.5.1.6 Licensing Actions

Based on the information provided in the LAR Attachment C, the licensee identified two exemptions from the deterministic requirements that were previously approved by the NRC and that are being transitioned with the NFPA 805 RI/PB FPP. The engineering evaluations that form the safety basis for approval of these two previously approved exemptions are being used as qualitative engineering evaluations with respect to the deterministic requirements of NFPA 805. Each of the exemptions being transitioned is summarized in each applicable fire area in LAR Attachment C and described in further detail in LAR Attachment K, "Existing Licensing Action Transition." The licensee stated in LAR Section 4.2.3, as required by NEI 04-02 (Reference 5), the review of these existing licensing actions included a determination of the basis of acceptability and a determination that the basis of acceptability is still valid. The licensing actions being transitioned are summarized in Table 3.5-2.

Table 3.5-2: Previously Approved Licensing Actions Being Transitioned

| Licensing Action Description | Applicable Fire Areas | Basis and Continuing Validity | NRC Staff Evaluation |
|---|---|---|---|
| Appendix R Exemption, Screenhouse Auxiliary Motor Control Center (MCC) Room Lack of Automatic Suppression (Criteria III.G.2.c) – Exemption 7.7 (Reference 23)<br><br>The original exemption was for lack of automatic suppression in the fire area as required by Paragraph III.G.2.c of Appendix R to 10 CFR 50. The transitioned compliance basis for the applicable fire area is NFPA 805, Section 4.2.3.3(c), which is similar to the original Appendix R requirement. | AA32 | The basis for approval as described by the licensee in LAR Attachment K is the ceilings and walls are 3-hour rated; the stairway and exhaust ventilation preclude buildup of a hot gas layer where the cables penetrate the fire zone; the ESW cables have 1-hour barriers; combustible loading is low and 3-hour dampers are installed in the Unit 2 pump cubicle supply ducts. In addition, detection is installed in the area. The licensee stated in Section 4.2.3 of the LAR that the basis for the previous NRC staff approval of the exemption has been verified and remains valid. | Based on the previous staff approval of the engineering justification for this exemption and the statement by the licensee that the basis remains valid, the NRC staff concludes that the underlying condition allowed by this licensing action is acceptable as a performance-based qualitative engineering analysis. |
| Appendix R Exemption, Reactor Coolant Pump (RCP) Lube Oil Collection System (Criteria III.O) – Exemption 7.15 (References 23 and 84)<br><br>The original exemption was for reactor coolant pump (RCP) oil collection capacity that did not meet Paragraph III.O of Appendix R to 10 CFR 50. The transitioned compliance basis for the applicable fire areas is NFPA 805, Section 3.3.12.(2), which is similar to the original Appendix R requirement. | AA58, AA56 | The exemption was approved based on the RCP motor lube oil system being capable of withstanding the safe shutdown earthquake, and the oil collection tank being provided with sufficient capacity to hold the total lube oil inventory of one reactor coolant pump with margin and is designed so that any overflow will be drained to a safe location. The licensee stated in Section 4.2.3 of the LAR that the basis for the previous NRC staff approval of the exemption has been verified and remains valid. | Based on the previous staff approval of the engineering justification for this exemption and the statement by the licensee that the basis remains valid, the NRC staff concludes that the underlying condition allowed by this licensing action is acceptable as a performance-based qualitative engineering analysis. |

The NRC staff reviewed the description of the previously approved exemptions from the deterministic requirements, the basis for and continuing validity of the exemptions, and the NRC staff's original evaluation or basis for approval of the exemptions. The NRC's staff's evaluation of each exemption is provided in Table 3.5-2.

Based on the NRC staff's review of the licensing actions identified and described in LAR Attachments C and K, the NRC staff concludes that the Licensing Actions are identified by applicable fire area and remain valid to support the proposed license amendment because the licensee used the process described in NEI 04-02, as endorsed by RG 1.205, which requires a determination of the basis of acceptability and a determination that the basis is still valid. Based on the previous staff approval of the exemptions and the statement by the licensee that the basis remains valid as presented in each appropriate fire area, the NRC staff concludes that the engineering evaluations being carried forward supporting the NFPA 805 transition, as identified in Table 3.5-2, are acceptable. See Section 2.5 of this SE for further discussion.

3.5.1.7 Existing Engineering Equivalency Evaluations (EEEEs)

The existing engineering equivalency evaluations (EEEEs) that support compliance with NFPA 805 Chapter 4 were reviewed by the licensee using the methodology contained in NEI 04-02 (Reference 5). The methodology for performing the EEEE review included the following determinations:

- The EEEE is not based solely on quantitative risk evaluations,
- The EEEE is an appropriate use of an engineering equivalency evaluation,
- The EEEE is of appropriate quality,
- The standard license condition is met,
- The EEEE is technically adequate,
- The EEEE reflects the plant as-built condition, and
- The basis for acceptability of the EEEE remains valid

In LAR Section 4.2.2, the licensee states the guidance in RG 1.205 (Reference 1), Regulatory Position 2.3.2, and FAQ 07-0054 (sic), "Demonstrating Compliance with Chapter 4 of NFPA 805" (Reference 91), was followed. EEEEs that demonstrate a fire protection system or feature is "adequate for the hazard" are to be addressed in the LAR as follows:

- If not requesting specific approval for "adequate for the hazard" EEEEs, then the EEEE is referenced where required and a brief description of the evaluated condition is provided.

- If requesting specific NRC approval for "adequate for the hazard" EEEEs, then the EEEE is referenced where required to demonstrate compliance and is included in Attachment L for NRC review and approval.

The licensee identified and summarized the EEEEs for each fire area in LAR Attachment C, as applicable. The licensee did not request the NRC staff to review and approve any of these EEEEs.

Based on the NRC staff's review of the licensee's methodology for review of EEEEs and identification of the applicable EEEEs in LAR Attachment C, the NRC staff concludes that the use of EEEEs meets the requirements of NFPA 805 and the guidance of RG 1.205 and FAQ 08-0054, and is acceptable.

3.5.1.8 Variances from Deterministic Requirements (VFDRs)

For those fire areas where deterministic criteria were not met, VFDRs were identified and evaluated using performance-based methods. VFDR identification, characterization, and resolutions were identified and summarized in LAR Attachment C for each fire area. The VFDRs can generally be categorized as: (1) inadequate separation resulting in fire-induced damage of process equipment or associated cables required for the identified success path (2) inadequate separation resulting in fire-induced spurious operation of equipment that may defeat the identified success path, and (3) inadequate separation resulting in fire-induced failure of process monitoring instrumentation or associated cables required for the identified success path. A total of 267 VFDRs are identified in LAR Attachment C, of which two of the VFDRs (all in fire area AA36/42) were stated to "have been removed" and are no longer VFDRs. The licensee's approach to resolution of the remaining 265 VFDRs and the NRC staff's review of the resolutions are described in the paragraphs below:

For 98 of the VFDRs, which are located in 21 of the performance-based fire areas, the licensee determined that the FRE results were acceptable with no further action. In these instances, based on the NRC staff's review of the licensee's methods for assessing risk, DID, and safety margin (contained in SE Section 3.4) as described in the LAR, the NRC staff concludes that this resolution of the applicable VFDRs is acceptable.

For 111 of the VFDRs, which are located in 14 of the performance-based fire areas, the licensee determined that the FRE results were acceptable with an RA credited. In these instances, based on the NRC staff's review of the licensee's methods for assessing risk, DID, and safety margin as described in the LAR, and the feasibility of the RAs associated with the VFDRs as documented in LAR Attachment G, the NRC staff concludes that this resolution of the applicable VFDRs is acceptable.

For 27 of the VFDRs, which are located in 20 of the performance-based fire area, the licensee determined that the FRE results were acceptable with a DID-RA credited. In these instances, based on the NRC staff's review of the licensee's methods for assessing risk, DID, and safety margin as described in the LAR, and the DID-RAs associated with the VFDRs as documented in LAR Attachment G, the staff concludes that this resolution of the applicable VFDRs is acceptable.

For 11 of the VFDRs, which are located in two of the performance-based fire areas (AA40 and AA43), the licensee determined that the FRE results were acceptable with an RA credited and plant modifications to implement transient combustible free areas and an automatic carbon dioxide ($CO_2$) suppression system. The modification to implement transient combustible free areas is identified as implementation item S-3.3 in Attachment S, Table S-3 of the LAR. The modification to modify the $CO_2$ system from manual to automatic actuation is identified as modification item S-2.2 in Attachment S, Table S-2 of the LAR. In these instances, based on the NRC staff's review of the licensee's methods for assessing risk, DID, and safety margin as described in the LAR, the RAs associated with the VFDRs as documented in LAR Attachment G, and the proposed implementation item S-3.3 and modification item S-2.2, the staff concludes that this resolution of the applicable VFDRs is acceptable.

For one of the VFDRs located in performance-based fire area AA40, the licensee determined that the FRE results were acceptable with a DID-RA credited and plant modifications to implement transient combustible free areas and an automatic $CO_2$ suppression system. The modification to implement transient combustible free areas is identified as implementation item S-3.3 in Attachment S, Table S-3 of the LAR. The modification to modify the $CO_2$ system from manual to automatic actuation is identified as modification item S-2.2 in Attachment S, Table S-2 of the LAR. In this instance, based on the NRC staff's review of the licensee's methods for assessing risk, defense-in-depth and safety margin as described in the LAR, the DID-RA associated with the VFDR as documented in LAR Attachment G, and the proposed implementation item S-3.3 and modification item S-2.2, the staff concludes that this resolution of the applicable VFDR is acceptable.

For eight of the VFDRs, which are located in two of the performance-based fire areas (AA40 and AA43), the licensee determined that the FRE results were acceptable with plant modifications to implement transient combustible free areas and an automatic $CO_2$ suppression system. The modification to implement transient combustible free areas is identified as implementation item S-3.3 in Attachment S, Table S-3 of the LAR. The modification to modify the $CO_2$ system from manual to automatic actuation is identified as modification item S-2.2 in Attachment S, Table S-2 of the LAR. In these instances, based on the NRC staff's review of the licensee's methods for assessing risk, DID, and safety margin as described in the LAR, and the proposed implementation item S-3.3 and modification item S-2.2, the staff concludes that this resolution of the applicable VFDRs is acceptable.

For four of the VFDRs, which are located in two of the performance-based fire areas (AA41 and AA44), the licensee determined that the FRE results were acceptable with a plant modification to implement transient combustible free areas. The modification to implement transient combustible free areas is identified as implementation item S-3.3 in Attachment S, Table S-3 of the LAR. In these instances, based on the NRC staff's review of the licensee's methods for assessing risk, DID, and safety margin as described in the LAR, and the proposed implementation item S-3.3, the staff concludes that this resolution of the applicable VFDRs is acceptable.

For one of the VFDRs, which is located in performance-based fire area AA41, the licensee determined that the FRE results were acceptable with an RA credited and a plant modification to implement transient combustible free areas. The modification to implement transient combustible free areas is identified as implementation item S-3.3 in Attachment S, Table S-3 of the LAR. In this instance, based on the NRC staff's review of the licensee's methods for assessing risk, DID, and safety margin as described in the LAR, the RA associated with the VFDR as documented in LAR Attachment G, and the proposed implementation item S-3.3, the staff concludes that this resolution of the applicable VFDR is acceptable.

For three of the VFDRs, which are located in two of the performance-based fire areas (AA41 and AA44), the licensee determined that the FRE results were acceptable with a DID-RA credited and a plant modification to implement transient combustible free areas. The modification to implement transient combustible free areas is identified as implementation item S-3.3 in Attachment S, Table S-3 of the LAR. In these instances, based on the NRC staff's review of the licensee's methods for assessing risk, DID, and safety margin as described in the

LAR, the DID-RAs associated with the VFDRs as documented in LAR Attachment G, and the proposed implementation item S-3.3, the staff concludes that this resolution of the applicable VFDRs is acceptable.

Where RAs or DID-RAs are credited for disposition of a VFDR, these actions are described in LAR Attachment G. The NRC staff's evaluation of the identified RAs is provided in Sections 3.5.1.9 and 3.5.1.10 below.

For all fire areas where the licensee used the PB approach to meet the NSPC, each VFDR and the associated disposition has been described in LAR Attachment C. The NRC staff reviewed these VFDRs and the licensee's disposition of each. Based on the review of the VFDRs and associated resolutions as described in LAR Attachment C, as supplemented, the NRC staff concludes that the licensee's identification and resolution of the VFDRs is acceptable. The NRC staff's evaluation of the acceptability of the change in risk reported for each fire areas as well as the cumulative change in risk reported for each CNP unit is provided in Section 3.4.6 of this SE.

3.5.1.9 Recovery Actions

LAR Attachment G lists the RAs identified in the resolutions to the VFDRs delineated in LAR Attachment C for each fire area. The RAs identified include both actions credited to meet the risk acceptance guidelines as well as actions relied upon as DID (see SE Section 3.5.1.10 below). The NRC staff reviewed the licensee's documentation related to the identified RAs.

LAR Attachment G identifies actions to provide temporary control room ventilation by opening the control room door and installing a gasoline-powered portable fan. In RAI 18 dated January 27, 2012 (Reference 66), the NRC staff requested additional information on the use of this fan including the timing of the action; the means to exhaust combustion gases; and any special controls needed to safely handle and address fuel handling inside the power block. In its response dated April 27, 2012 (Reference 8), the licensee stated the action is required in 30 minutes as validated by calculation. The fan and 3 hours of fuel are stored in a non-flammable cabinet near the control room entrance door of each unit. Simulation testing conducted by the licensee indicated that carbon monoxide reading taken over a 60-minute period were approximately one-third of the allowable levels of the plant, State, and Federal values. In RAI 18.01 dated October 11, 2012 (Reference 67), the NRC staff informed the licensee that the use and refueling of gas-powered blowers for temporary ventilation of the Control Room is inconsistent with GDC-3 for fire protection of SSCs important to safety and presented a hazard to equipment important to nuclear safety that is similar to the hazard of portable fuel-fired heaters, which are prohibited by NFPA 805, Section 3.3.1.3.4. The NRC staff requested that the licensee provide an alternative approach to providing temporary Control Room ventilation. In its response dated October 15, 2012 (Reference 11), the licensee agreed to provide an alternative approach to providing Control Room ventilation and to revise the analyses accordingly. The licensee's proposed changes in its response included a revision to implementation item S-3.14 and new implementation item S-3.17 in Attachment S, Table S-3 of the LAR.

In response to several RAIs, the licensee identified an operator action for restoring the TDAFW pump N train battery charger after it is tripped following loss of offsite power or after a safety injection signal. The N train battery bus provides power for the TDAFW pump and discharge valves. While the licensee does not consider this an RA, this operator action is credited in the FPRA to provide a reduction in the risk by mitigating failures in the AFW system (see Section 3.4.4 of this SE). The licensee provided an implementation item S-3.14 in Attachment S, Table S-3 of the LAR, to revise CNP operating procedures to include this operator action and to include it in the RI/PB FPP.

The NRC staff reviewed LAR Section 4.2.1.3, "Establishing Recovery Actions," and Attachment G, "Recovery Actions Transition," to evaluate whether the licensee meets the associated requirements for the use of RAs per NFPA 805. The NRC staff's evaluation of the licensee's process for identifying RAs and assessing their feasibility is provided in SE Section 3.2.4, "Establishing Recovery Actions." The NRC staff's evaluation of the additional risk of RAs credited to meet the risk acceptance guidelines is provided in Section 3.4.4 of this SE.

3.5.1.10 Recovery Actions Credited for Defense in Depth (RA-DID)

The licensee stated in LAR Attachment G that DID measures have been conservatively maintained to provide plant operations with written guidance where such actions will enhance Echelon 3 of DID, to provide some assurance that one success path of safe shutdown capability can be restored in the event that Echelon 1 and Echelon 2 of DID are somehow degraded or rendered ineffective. The NRC staff's evaluation of the licensee's process for maintaining a balance among the DID echelons is provided in SE Section 3.4.1, "Maintaining Defense-in-Depth and Safety Margins."

The licensee stated that the nuclear safety and radioactive release performance goals, objectives, and criteria of NFPA 805, including the risk acceptance guidelines, are met without these DID actions. However, DID-RAs are credited to meet the requirements for DID and are therefore considered part of the RI/PB FPP, which necessitates that these actions would be subject to a plant change evaluation if subsequently modified or removed.

The NRC staff reviewed LAR Section 4.2.1.3, "Establishing Recovery Actions," and Attachment G, "Recovery Actions Transition," to evaluate whether the licensee meets the associated requirements for the use of RAs per NFPA 805. The NRC staff's evaluation of the licensee's process for identifying RAs and assessing their feasibility is provided in SE Section 3.2.4, "Establishing Recovery Actions."

3.5.1.11 Conclusion for Section 3.5.1

For those fire areas that used a deterministic approach in accordance with NFPA 805, Section 4.2.3, as described in LAR Attachment C, the NRC staff concludes that each of the fire areas analyzed using the deterministic approach meet the associated criteria of NFPA 805 as demonstrated by the following:

- The licensee's documented compliance with NFPA 805, Section 4.2.3;

- Transitioned exemptions from the existing fire protection licensing basis were reviewed for applicability, as well as continued validity, and found acceptable by the NRC staff;

- The licensee's assertion that the success path will be free of fire damage without reliance on RAs;

- The licensee's assessment that the suppression systems in the fire area will have no impact on the ability to meet the NSPC; and

- The licensee's appropriate determination of the fire suppression and detection systems required to meet the NSPC.

For those fire areas that used the PB approach in accordance with NFPA 805, Section 4.2.4, as described in LAR Attachment C, the NRC staff concludes that each fire area has been properly analyzed and is compliance with the NFPA 805 requirements as demonstrated by the following:

- Transitioned exemptions from the existing fire protection licensing basis were reviewed for applicability, as well as continued validity, and found acceptable by the NRC staff.

- VFDRs were evaluated and found to be acceptable by the NRC staff based on an integrated assessment of risk, DID, and safety margins (see Sections 3.4.1 and 3.4.6 of this SE). Where credited in the disposition of the VFDRs, modifications and RAs were identified. Implementation items address the modifications and other actions as applicable.

- RAs used to demonstrate the availability of a success path to achieve the NSPC were evaluated and the additional risk of their use determined, reported, and found to be acceptable by the NRC staff (see Section 3.4.4 of this SE).

- The licensee's analysis appropriately identified the fire protection SSCs required to meet the NSPC, including fire suppression and detection systems, as well as required fire protection features (ERFBS, radiant energy shields, etc.).

- ERFBS that are credited in meeting the requirements of NFPA 805 are documented on a fire area basis and verified to meet the criteria of NFPA 805.

On this basis, the NRC staff concludes that each fire area at the CNP has been appropriately evaluated in accordance with the deterministic or PB requirements of NFPA 805 to demonstrate that CNP can achieve and maintain the NSPC of NFPA 805.

3.5.2 Clarification of Prior NRC Approvals

As stated in LAR Attachment T, there are no elements of the current FPP for which NRC clarification is needed.

### 3.5.3 Fire Protection During Non-Power Operational Modes

NFPA 805, Section 1.1 "Scope," states the following:

> This standard specifies the minimum fire protection requirements for existing light water nuclear power plants during all phases of plant operation, including shutdown, degraded conditions, and decommissioning.

NFPA 805, Section 1.3.1, "Nuclear Safety Goal," states the following:

> 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.

The NRC staff reviewed LAR Section 4.3, "Non-Power Operational Modes" and Attachment D, "NEI 04-02, Table F-1, Non-Power Operational Modes Transition," to evaluate the licensee's treatment of potential fire impacts during non-power operations (NPO). The licensee followed the guidance used in the process described in NEI 04-02, Revision 2 (Reference 5), and as modified by FAQ 07-0040, "Non-Power Operations Clarification," Revision 4 (Reference 85), for demonstrating that the NSPC are met for high-risk evolutions (HREs) during NPO modes.

#### 3.5.3.1 NPO Strategy and Analysis Process

LAR Section 4.3, "Non-Power Operational Modes," and Attachment D, "NEI 04-02 – Non-Power Operations Modes Transition," describe the licensee's implementation of the FAQ 07-0040 (Reference 85) process. The licensee stated that its goal is to ensure that contingency plans are established when the plant is in an NPO condition where the risk is high. The licensee's strategy for control and protection of equipment during NPO modes follows the process flow in LAR Figure 4-6, which considers the availability of Key Safety Function (KSF) equipment, if the plant is in an HRE, if the KSF may be lost due to fire, and contingency plans to mitigate the risks. If KSFs are unaffected, normal risk management controls and fire prevention/protection processes and procedures are used at CNP.

As described in LAR Attachment D, the licensee's procedure defines HRE as "outage activities, plant configurations, or conditions during shutdown where the plant is more susceptible to an event causing the loss of a KSF." The procedure contains specific actions to address reduced inventory conditions that result in a short time to boil, limited methods for decay heat removal, and low RCS inventory. The considerations are consistent with those described in FAQ 07-0040.

The process to demonstrate that the NSPC are met during NPO involved the following steps as described in LAR Section 4.3.1 and depicted in Figures 4-5 and 4-6 of the LAR:

- Reviewed the existing Outage Management Processes
- Identified Equipment/Cables:

  - Reviewed plant systems to determine success paths that support each of the defense-in-depth KSFs, and

  - Identified cables required for the selected components and determined their routing.

- Performed Fire Area Assessments (identify "pinch-points," i.e., plant locations where a single fire may damage all success paths of a KSF).

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

The NPO process described and documented by the licensee in LAR Section 4.3 and Attachment D follows the guidance of NEI 04-02. On this basis, the NRC staff concludes that the process is acceptable.

3.5.3.2 NPO System, Component, and Cable Identification

To identify components and cables associated with KSFs, the licensee considered the Plant Operating States (POS): Hot Shutdown, Cold Shutdown, and Refueling.

The KSFs evaluated against the POS include, shutdown cooling, inventory control, reactivity control, containment, 4kV electric power sources, electric power distribution, service water systems, and spent fuel pit cooling. The evaluation resulted in the exclusion of Containment Control and Spent Fuel Pool Cooling KSFs from further consideration. Containment access and closure was determined to be adequately controlled administratively and sufficient time and alternative methods were determined to be available to mitigate spent fuel pool cooling and reactivity control. The remaining KSFs were explicitly modeled in the plant NPO analysis database.

As described in the LAR Attachment D, the licensee identified equipment and cables necessary to support the KSF success paths. Additional cable selection was performed for those components evaluated for at-power but whose functional requirements may have been different for the non-power analysis. The operational modes and functional requirements for the systems and components were reviewed, the equipment and cables were logically tied and related to the applicable KSF success paths, and power supplies and other supporting components such as interlocks were also identified, listed, and tied with their component and KSF success paths in the analysis database.

The licensee's process to define POS and identify NPO systems, components, and cables, as described in LAR Attachment D is consistent with FAQ 07-0040. NPO systems, components, and cables logically related to KSFs in the NPO analysis database. On this basis, the licensee's approach to identification of NPO systems, components, and cables is acceptable to the NRC staff.

3.5.3.3 NPO Fire Area Assessments

Following identification of KSF equipment and cables, the licensee performed an analysis on a fire area basis to identify areas where redundant equipment and cables credited for a given KSF fail due to fire damage (i.e., pinch-points). The licensee used a deterministic fire separation approach to identify pinch-points by assuming fire would impact all KSF components and cables in the fire area. As stated in Section 4.3.2 of the LAR, fire modeling was not used to eliminate any fire area from being a pinch-point.

The licensee evaluated 57 fire areas to identify pinch-points and associated loss of KSFs. Of the areas evaluated, 11 fire areas were found to have an adequate number of KSF success paths survive the entire loss of the fire area, and 46 fire areas were found to have pinch-points resulting in loss of one or more KSF success paths. Pinch-points were resolved using engineering justifications including the recommended use of RAs or fire prevention and protection controls consistent with the risk management strategies described in FAQ 07-0040. The NRC staff requested in RAI 14 dated January 27, 2012 (Reference 66), that the licensee identify the pinch-points by fire areas. In its response dated April 27, 2012 (Reference 8), the licensee identified the individual fire areas, if pinch-points were identified in the fire area, and the KSFs impacted by the pinch-points. The licensee states in its response that the analysis will be used to identify the pinch-point locations to the applicable plant organizations. Implementation item S-3.6 in Attachment S, Table S-3 of the LAR, will incorporate the NPO analysis in plant technical and administrative procedures. On the basis of the NPO analysis as described in the LAR and the response to RAI 14 (Reference 8), the NRC staff concludes that the licensee's method to perform NPO fire area assessments as described in the LAR is acceptable.

3.5.3.4 NPO Pinch-Point Resolutions and Outage Risk Management

The guidance in NEI 04-02 as modified by FAQ 07-0040, describes a method for managing risks associated with fire-induced vulnerabilities during outages. The FAQ describes the normal fire protection DID measures considered adequate to manage the risk of fires that may cause minor losses of system capability or redundancy, but will not result in total loss of the KSF. For HREs, the FAQ identifies additional DID measures and strategies for managing risk in areas with known pinch-points or where pinch-points may arise as a result of equipment removed from service.

The licensee's approach to managing NPO risks is described in LAR Attachment D and follows the guidance of FAQ 07-0040 to protect KSFs during normal and HREs. The LAR cites the licensee's existing outage procedure and describes the procedure as considering hazards such as fire in establishing compensatory measures and controls as appropriate to the scope of work. The licensee further describes the current procedures for ensuring the Unit 1/Unit 2 cross-tie capabilities relied on for safe shutdown are maintained when either unit is in an outage condition.

Based on a review of LAR Section 4.3 and Attachment D, the NRC staff requested in RAI 14 dated January 27, 2012 (Reference 66), that the licensee provide additional information and discussion regarding procedure changes, protective strategies to be used in preventing fire-related events that impact KSFs, any actions credited to minimize the impact of spurious operations during NPO, types of compensatory actions when certain NPO-credited equipment is removed from service, and any locations where KSFs are achieved solely via RAs or for which instrumentation is needed to support RAs required to maintain a safe and stable state, including the feasibility of these actions. In its response to RAI 14 dated April 27, 2012 (Reference 8), the licensee summarized the approach to incorporating NPO into the plant outage, operating, and administrative procedures and identified several of the procedures that would likely be updated.

Procedural controls to address the potential impact of a fire on certain valves (i.e., pinch-points) during higher risk evolutions are listed, as follows:

- Disable power to residual heat removal (RHR)-related motor-operated valves 1(2)-ICM-111, 1(2)-IMO-390.
- Fail ECCS-related air-operated valves 1-IRV-50, 1(2)-IRV-60 closed by isolation of the air supply.
- De-energize VCT isolation when the RWST is aligned as suction source for charging pumps.

Management of KSFs includes the following attributes:

a) The Outage Schedule establishes SSCs to provide backup for KSF. The backup capabilities provided should be commensurate with plant conditions.

b) The Outage Schedule optimizes safety system availability. Systems are returned to service, either operable when required by TSs, or available as soon as practicable following completion of scheduled work.

c) The operability of systems and components as defined by TSs is assured. This is accomplished through post maintenance testing, plant modification acceptance testing, surveillance testing, monitoring of key parameters with the system in service, verification of system alignment, and administrative control by operations personnel.

d) SSCs identified to provide DID are controlled such that they remain available during the outage window specified.

The use of RAs, or RAs with alternate indication, is recommended as the means for resolving KSF pinch-points in the following fire areas:

AA2C: Alternate indication credited for Decay Heat Removal KSF.

AA11: Recovery action with alternate indication credited for Inventory, Reactivity, and Decay Heat Removal KSFs.

AA13: Alternate indication credited for Decay Heat Removal KSF.

AA14: Recovery action credited for Reactivity and Support KSFs. Recovery action with alternate indication credited for Decay Heat Removal KSF.

AA15: Alternate indication credited for Decay Heat Removal KSF. Recovery action credited for Support KSF.

AA24: Alternate indication credited for Decay Heat Removal KSF.

AA32: Recovery action credited for Inventory, Reactivity and Decay Heat Removal KSFs.

AA34: Recovery action credited for Support KSF. Alternate indication credited for Decay Heat Removal KSF.

AA57A: Alternate indication credited for Reactivity and Decay Heat Removal KSF. Recovery action credited for Support KSF.

The licensee also identified recommended preventive or mitigating strategies to be incorporated in the procedures, including actions to configure the plant systems to prevent or mitigate potential spurious operations during HREs, and the possible use of RAs. The strategies, as described in the RAI response, are consistent with FAQ 07-0040 (Reference 85). The licensee states these recommended actions from the NPO analysis will be evaluated for inclusion in plant procedures as part of implementation item S-3.6 in Attachment S, Table S-3 of the LAR.

3.5.3.5 Conclusion for Section 3.5.3

Based on its review of the information provided in the LAR, as supplemented, the NRC staff concludes that the licensee used methods consistent with the guidance provided in FAQ 07-0040, Revision 4, and RG 1.205, Revision 1, to identify the equipment required to achieve and maintain the fuel in a safe and stable condition during NPO modes. Furthermore, the licensee has a process in place to ensure that fire protection DID measures will be implemented to achieve the KSFs during plant outages.

NFPA 805 requires that the NSPC be met during any operational mode or condition, including NPO. As described above, the licensee has performed the following engineering analyses to demonstrate that it meets this requirement:

- Identified the KSFs required to support the NSPC during non-power operations.
- Identified the POSs where further analysis is necessary during non-power operations.
- Identified the SSCs required to meet the KSFs during the POSs analyzed.
- Identified the location of these SSCs and their associated cables.
- Performed analyses on a fire area basis to identify pinch-points where one or more KSF could be lost as a direct result of fire-induced damage.

- Planned/implemented modifications to appropriate station procedures in order to employ one or more fire protection strategy for reducing risk at these pinch-points during HREs.

Accordingly, based on the information provided in the LAR as supplemented, and subject to completion of implementation item S-3.6 in Attachment S, Table S-3 of the LAR, the NRC staff concludes that the licensee has provided reasonable assurance that the NSPC are met during NPO modes and HREs at CNP.

3.5.4 Conclusion for Section 3.5

The NRC staff reviewed the licensee's RI/PB FPP, as described in the LAR and its supplements, to evaluate the NSCA results. The licensee used a combination of the deterministic approach in accordance with NFPA 805, Section 4.2.3, and the PB approach in accordance with NFPA 805, Section 4.2.4, to perform this assessment at CNP.

For those fire areas that used a deterministic approach, the NRC staff concluded that:

- The engineering evaluations for exemptions from the existing CNP FPP were evaluated and found to be valid and acceptable for meeting the deterministic requirements of NFPA 805, as allowed by NFPA 805, Section 2.2.7.

- Fire suppression effects were evaluated and found to have no adverse impact on the ability to achieve and maintain the NSPC for each fire area.

- All DID-RAs were documented for each fire area.

- The required automatic fire suppression and fire detection systems were appropriately documented for each fire area.

Accordingly, the NRC staff concludes that there is reasonable assurance that each fire area utilizing the deterministic approach meets the deterministic requirements of NFPA 805, Section 4.2.3.

For those fire areas that used a PB approach, the NRC staff concluded that:

- The engineering evaluations for exemptions from the existing CNP FPP were evaluated and are valid and acceptable for meeting the deterministic requirements of NFPA 805 as allowed by NFPA 805, Section 2.2.7.

- Fire suppression effects were evaluated and have no adverse impact on the ability to achieve and maintain the NSPC for each fire area.

- All VFDRs were evaluated using the FRE performance-based method (in accordance with NFPA 805, Section 4.2.4.2) to address risk impact, DID, and safety margin, and were found to be acceptable.

- All RAs necessary to demonstrate the availability of a success path were evaluated with respect to the additional risk presented by their use and were found to be acceptable in accordance with NFPA 805, Section 4.2.4 (see Section 3.4.4 of this SE).

- All DID-RAs were properly documented for each fire area.

- The required automatic fire suppression and fire detection systems were appropriately documented for each fire area.

The required automatic fire suppression and automatic fire detection systems were appropriately documented for each fire area.

Based on the analyses performed by the licensee and described in the LAR as supplemented, the NRC staff concludes that each fire area utilizing the PB approach, in accordance with NFPA 805, Section 4.2.4, is able to achieve and maintain the NSPC. Furthermore, there is reasonable assurance that the associated FREs meet the requirements for risk, DID, and safety margin.

The NRC staff's review of the licensee's analysis and outage management process during NPO modes concludes that there is reasonable assurance that the NSPC will be met during NPO modes and HREs and that the licensee used methods consistent with the guidance in FAQ 07-0040 and RG 1.205. The NRC staff's review also concluded that the normal FPP DID actions are credited for addressing the risk impact of those fires which potentially affect one or more trains of equipment that provide a KSF required during NPO modes, but would not be expected to cause the total loss of that KSF. The NRC staff concludes that this overall approach for fire protection during NPO modes is acceptable.

## 3.6 Radioactive Release Performance Criteria

NFPA 805, Chapter 1, defines the radioactive release goals, objectives, and performance criteria that must be met by the FPP in the event of a fire at a nuclear power plant.

Radioactive Release Goal

The radioactive release goal is to provide reasonable assurance that a fire will not result in a radiological release that adversely affects the public, plant personnel, or the environment.

Radioactive Release Objective

Either of the following objectives shall be met during all operational modes and plant configurations.

(1) Containment integrity is capable of being maintained.

(2) The source term is capable of being limited.

Radioactive Release Performance Criteria

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

In order to assess whether the CNP FPP to be implemented under NFPA 805 meets the above requirements, the licensee performed a review of the current CNP FPP using the methodology contained in NEI 04-02 (Reference 5) and FAQ 09-0056 (Reference 86). Each fire zone was first screened to determine the potential for generating radioactive effluents during firefighting operations. The screening process considered input from CNP Radiation Protection personnel and evaluated the fire zone's potential for radioactive effluent release during all modes of operation. Fire zones where there is no possibility of radioactive materials being present (e.g., those outside of the Radiologically Controlled Area) were screened from further review. For all other fire zones, engineering controls afforded by plant design features, fire pre-plans, and fire brigade training materials were reviewed to ascertain whether existing CNP FPP is adequate to ensure that radioactive materials (contamination) generated as a direct result of fire suppression activities are contained and monitored before release to unrestricted areas, such that the release would meet the NFPA 805 radioactive release performance criteria. Table E-1 of the LAR provides a detailed summary, on a fire zone by fire zone basis, of the licensee's qualitative assessment.

The licensee's review determined that the current FPP is compliant with the radiological release requirements of NFPA 805 and the guidance in RG 1.205. With the exception of those fire zones discussed below, the licensee's qualitative review determined that CNP buildings and structures provide sufficient capacity to contain the liquid and gaseous firefighting effluents such that there are no offsite releases. The licensee's review did not identify any plant design features (such as roll-up doors, windows, or drains) that would divert the liquid or gaseous effluents from being collected/processed as credited. The fire pre-plans have been revised to assure that manual actions are taken to prevent offsite releases in those fire areas where there is a potential for such effluent diversions. In addition, the licensee updated each of the fire pre-plans addressing fire areas where radioactive materials may be present to include provisions for containment and monitoring of smoke and fire suppression agent runoff should the effectiveness of the installed engineering controls be challenged or impacted by fire suppression activities. In general, the Reactor Containment and Auxiliary Building ventilation systems are credited for the capture and monitoring of airborne products of firefighting. In those fire areas where no monitored ventilation is provided, or where normal ventilation is not available, gaseous effluents will be manually ventilated to the outside (if Radiation Protection personnel have verified the radioactive concentrations are within the Technical Specification (TS) limits) or to an area that has operating normal ventilation. Subsequent release of these effluents will be within the TS limits as determined by the CNP effluent release program. Consistent with the guidance in RG 1.205, as discussed in Regulatory Issue Summary 2007-19 (Reference 87), there is reasonable assurance that the annual dose limits of 10 CFR 20 are met if the concentrations of radioactive materials in airborne and liquid releases are maintained below the instantaneous release limits in the CNP TS.

Table E-1 of the CNP LAR identifies several fire areas, designated as the "yard," where radioactive materials are stored but where neither a monitored liquid drain, nor monitored ventilation, is provided. In response to the NRC staff's RAI dated June 29, 2012 (Reference 9), the licensee provided Technical Evaluation 11.76, Revision 0, "NFPA 805 Airborne and Liquid Effluents Offsite Dose Analysis," to demonstrate that firefighting effluents in these areas will meet the radiological release performance criteria. For airborne releases, the licensee back-calculated, using the methods and parameters in its Offsite Dose Calculation Manual (ODCM), the quantity of radioactive material that would have to be released from a container fire, to exceed the TS airborne particulate release limit. The expected dose rates for the types of containers used to store radioactive materials onsite were then calculated, assuming that they contained these bounding amounts of radioactive material. In all cases the calculated container contact dose rates exceed the current CNP administrative limit (10 millirem per hour on contact) for any individual container of radioactive material stored in an outdoors area. Therefore, administratively limiting the source term in each container provides reasonable assurance that the maximum offsite dose resulting from a fire that releases the radioactive contents of a storage container, will be within the limits of 10 CFR 20 and will be consistent with the radiological release objectives. For liquid releases from such a fire, the licensee has determined that there is no potential for dose to a member of the public due to liquids released from a storage container during firefighting activities. In its June 29, 2012, response to the staff, the licensee stated that precautions will be taken using standard industry methods (such as creating berms, using sandbags and tarps to cover drains) to prevent contaminated water runoff to Lake Michigan. Liquid effluent captured onsite will be monitored and released in accordance with the CNP ODCM so that they are within the TS limits. The hydrology of the site is such that any liquid effluent that reaches the sub-surface ground water will migrate west towards the lake. CNP has restrictive covenants in place, effectively preventing the use of potentially contaminated groundwater as a drinking water source. As discussed in the licensee's ODCM Land Use Census, there is no drinking water pathway to a member of the public.

The licensee also reviewed the fire brigade training materials to ensure they are consistent with the pre-fire plans in terms of containment and monitoring of potentially contaminated smoke and fire suppression water. Table E-1 of the LAR identifies several fire zones where the associated fire pre-plans, and training materials, will provide instructions for communication with Radiation Protection personnel, and describe precautions to be undertaken for safe removal of contaminated smoke and water runoff in potentially contaminated areas. As indicated in LAR Attachment S, Table S-3, Item S-3.7, the licensee plans to complete these revisions within 12 months from issuance of the NFPA 805 SE.

NFPA 805 requires the licensee to address the nuclear safety and radioactive release goals, objectives and performance criteria in any operational mode. As noted above, the licensee's radioactive release review considered all plant operating modes (including power and non-power operations), since fire suppression activities, as defined in the pre-fire plans and fire brigade firefighting instruction operating guidelines, are written for any plant operating mode.

3.6.1 Conclusion for Section 3.6

Based on: (1) the information provided in the LAR, as supplemented (2) the licensee's use of fire pre-plans (3) the results of the NRC staff's evaluation of the identified engineered controls used to manage suppression water and combustion products, and (4) the development and implementation of newly revised fire brigade training procedures, the NRC staff concludes that the licensee's RI/PB FPP provides reasonable assurance that radiation releases to any unrestricted area resulting from the direct effects of fire suppression activities at CNP Unit 1 and Unit 2 are as low as reasonably achievable and are not expected to exceed the radiological dose limits in 10 CFR Part 20. Therefore, the NRC staff concludes that the licensee's RI/PB FPP complies with the requirements specified in NFPA 805, Sections 1.3.2, 1.4.2, and 1.5.2. Accordingly, the NRC staff concludes that this approach is acceptable.

## 3.7 NFPA 805 Monitoring Program

For this section of the SE, the following NFPA 805, Chapter 2, requirements are applicable to the NRC staff's review of the licensee's LAR:

NFPA 805 Section 2.6, "Monitoring":

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

NFPA 805 Section 2.6.1, "Availability, Reliability, and Performance Levels":

> Acceptable levels of availability, reliability, and performance shall be established.

NFPA 805 Section 2.6.2, "Monitoring Availability, Reliability, and Performance":

> Methods to monitor availability, reliability, and performance shall be established. The methods shall consider the plant operating experience and industry operating experience.

NFPA 805 Section 2.6.3, "Corrective Action":

> If the established levels of availability, reliability, or performance are not met, appropriate corrective actions to return to the established levels shall be implemented. Monitoring shall be continued to ensure that the corrective actions are effective.

The NRC staff reviewed LAR Section 4.6, "Monitoring Program" (Reference 6), the program that the licensee developed to monitor availability, reliability, and performance of CNP FPP systems and features after transition to NFPA 805. The focus of the NRC staff review was on critical elements related to the monitoring program, including the selection of FPP systems and features to be included in the program, the attributes of those systems and features that will be monitored, and the methods for monitoring those attributes. Implementation of the monitoring program will occur on the same schedule as the NFPA 805 RI/PB FPP implementation, which the NRC staff concluded was acceptable (implementation item S-3.2 in Attachment S, Table S-3 of the LAR).

Based on the information provided by the licensee in the LAR and RAI response (References 8 and 11), the NRC staff concludes that the licensee's NFPA 805 monitoring program development and implementation process provides reasonable assurance that an effective program for monitoring risk-significant fire SSCs will be implemented at CNP because it:

1. Establishes the appropriate scope of SSCs to be monitored;
2. Uses an acceptable screening process for determining the SSCs to be included in the program;
3. Establishes availability, reliability, and performance criteria for the SSCs being monitored; and
4. Requires corrective actions when SSC availability, reliability, or performance criteria targets are exceeded, to bring performance back within the required range.

However, since the final values for availability and reliability, as well as the performance criteria for the SSCs being monitored, have not been established for the NFPA 805 monitoring program as of the date of this SE, completion of the NFPA 805 monitoring program is an implementation item, as noted previously. Implementation of the monitoring program will occur on the same schedule as the implementation of NFPA 805, which the NRC staff concludes is acceptable.

#### 3.7.1. Conclusion for Section 3.7

The NRC staff reviewed the licensee's RI/PB FPP and RAI responses for Section 3.7 of this SE. The NRC staff concludes that, upon successful closure of the implementation item in this area, there is reasonable assurance that the licensee's monitoring program meets the requirements specified in Sections 2.6.1, 2.6.2, and 2.6.3 of NFPA 805.

### 3.8 Program Documentation, Configuration Control, and Quality Assurance

For this section of the SE, the requirements from NFPA 805, Section 2.7, "Program Documentation, Configuration Control and Quality," are applicable to the NRC staff's review of the LAR in regard to the appropriate content, configuration control, and quality of the documentation used to support the transition to NFPA 805 at CNP.

#### 3.8.1 Documentation

The NRC staff reviewed LAR Section 4.7.1, "Compliance with Documentation Requirements in Section 2.7.1 of NFPA 805." CNP's FPP design basis is a compilation of multiple documents (such as analyses, calculations, and engineering evaluations), databases, and drawings that are identified in Figure 4-9 of the LAR. The licensee stated that analyses performed to support NFPA 805 transition were performed in accordance with CNP processes for ensuring that assumptions are clearly defined, that results are easily understood, that results are clearly and consistently described, and that sufficient detail is provided to allow future review of the entire analysis, which meets or exceeds the requirements for documentation in Section 2.7.1 of NFPA 805.

The licensee stated in the LAR that documentation associated with the CNP RI/PB FPP will be maintained for the life of the plant and organized to facilitate review for accuracy and adequacy by independent reviewers and by NRC staff. Based on the description of the content of the CNP FPP design basis and supporting documentation, and the licensee's plans to maintain this documentation throughout the life of the plant, the NRC staff concludes that the licensee's approach meets the requirements of NFPA 805, Sections 2.7.1.1, 2.7.1.2, and 2.7.1.3 to develop and maintain FPP design basis documentation.

3.8.2 Configuration Control

The NRC staff reviewed LAR Section 4.7.2, "Compliance with Configuration Control Requirements in Sections 2.7.2 and 2.2.9 of NFPA 805." To support the many other technical, engineering and licensing programs at CNP, the licensee has existing configuration control processes and procedures for establishing, revising, or utilizing program documentation. The RI/PB FPP design basis and supporting documentation is being integrated into these configuration control processes and procedures. These processes and procedures require that all plant changes be reviewed for impact on the various CNP licensing programs, including the FPP. The licensee stated in the LAR that the configuration control process includes provisions for appropriate design and engineering reviews and approvals and that approved analyses are considered controlled documents available through the CNP document control system. The licensee also stated that analyses based on the PRA program, which includes the FPRA, are issued as formal analyses and subject to these same configuration control processes, and are additionally subjected to the PRA peer review process specified in the ASME/ANS PRA standard RA-Sa-2009. Configuration control of the FPP during the transition period is maintained by the CNP change evaluation process defined in existing CNP configuration management and configuration control procedures. CNP will revise these existing procedures, as necessary, for application to the NFPA 805 FPP.

Note that the NRC staff reviewed the licensee's process for updating and maintaining the FPRA to reflect plant changes made after the transition to NFPA; this review is documented in Section 3.4.1 of this SE.

Based on the description in the LAR of the CNP configuration control process, and the licensee's statements that the CNP RI/PB FPP design basis and supporting documentation are controlled documents and that plant changes are reviewed for impact on the FPP, the NRC staff concludes that there is reasonable assurance that the requirements of NFPA 805, Sections 2.7.2.1 and 2.7.2.2, will be met.

3.8.3 Quality

The NRC staff reviewed LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," which focuses on the quality of engineering analysis. The licensee stated that the RI/PB FPP Quality Assurance (QA) program will be included within and implemented by the CNP nuclear QA program. The relevant criteria of that program have been applied to the FPP. The following discussion addresses the application of that program to the CNP NFPA 805 FP. Further, the licensee is obligated to revise the QA program to reflect the applicable requirements of NFPA 805, Section 2.7.3 as an implementation item (Attachment S, Table S-3, implementation item S-3.10).

3.8.3.1 Review

The licensee stated that its procedures require independent review of analyses, calculations, and evaluations, including those performed in support of compliance with 10 CFR 50.48(c). CNP stated in the LAR that the analyses, calculations, and evaluations performed in support of transition to NFPA 805 requirements were independently reviewed and that analyses, calculations, and evaluations to be performed post-transition will be independently reviewed as required by CNP procedures.

Based on the licensee's description of the CNP process for performing independent reviews of analyses, calculations, and evaluations, the NRC staff concludes that the licensee's approach to meeting the requirements of NFPA 805, Section 2.7.3.1, is acceptable.

3.8.3.2 Verification and Validation (V&V)

The licensee stated in the LAR that calculational models and numerical methods used in support of transition to NFPA 805 requirements were verified and validated and that calculational models and numerical methods used post-transition will be verified and validated. CNP also stated that processes and procedures will be revised to include NFPA 805 quality requirements for post-transition FPP changes, including those for verification and validation. Revision of post-transition processes and procedures to include NFPA 805 requirements for verification and validation is identified as implementation item S-3.10 in Attachment S, Table S-3, of the LAR.

Based on the licensee's description of the CNP process for V&V of calculational models and numerical methods, the NRC staff concludes that the licensee's approach to meeting the requirements of NFPA 805, Section 2.7.3.2, is acceptable.

3.8.3.2.1 General

NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," May 2007 (Reference 53), documents the V&V of five selected fire models commonly used to support applications of risk-informed, performance-based fire protection at nuclear power plants. The seven volumes of this NUREG series report provide technical documentation concerning the predictive capabilities of a specific set of fire dynamics calculation tools and fire phenomenological models that may be used for the analysis of fire hazards in postulated nuclear power plant scenarios. When used within the limitations of the fire models and considering the identified uncertainties, these models may be employed to demonstrate compliance with the requirements of 10 CFR 50.48(c).

Accordingly, for those fire modeling elements performed by the licensee using the V&V applications contained in NUREG-1824 to support the transition to NFPA 805 at CNP, the NRC approves the use of these models, provided that the intended application is within the appropriate limitations, as identified in NUREG-1824.

In LAR Section 4.5.2 (Reference 6), the licensee also identified the use of several empirical correlations that are not addressed in NUREG-1824. The NRC staff reviewed the empirical correlation screening tool methodology, as well as the related material provided in the LAR, in order to determine whether the licensee adequately demonstrated alignment with specific portions of the applicable NUREG-1824 guidance.

Table 3.8.3.2-1, "V&V Basis for Fire Modeling Correlations Used at CNP," in Attachment A to this SE identifies these empirical correlations and models for the screening tool, as well as a NRC staff disposition for each.

The NRC staff concludes that the theoretical bases of the models and empirical correlations used in the fire modeling calculations that were not addressed in NUREG-1824 were identified and submitted to peer reviewed journals, authoritative publications such as *The SFPE Handbook of Fire Protection Engineering* (Reference 88). SE Table 3.8.3.2-2, in Attachment B, summarizes these additional fire models and the NRC staff's evaluation of the acceptability of each of the additional methods.

As reflected in Table 3.8.3.2-1 and Table 3.8.3.2-2, of Attachments A and B to this SE, the fire modeling employed by the licensee in the development of the CNP FPRA used either: (1) empirical correlations that provide bounding solutions for the ZOI, or (2) conservative input parameters in the application of the other models, which produced conservative results for the fire modeling analysis.

Based on the above, the NRC staff concludes that this approach provides reasonable assurance that the fire modeling used in the development of the fire scenarios for the CNP FPRA is appropriate and, therefore, is acceptable for use in this application (i.e., transition to NFPA 805).

3.8.3.2.2 Discussion of Selected RAI Responses

By letter dated January 27, 2012 (Reference 66), the NRC staff requested additional information related to the fire modeling used in support of the CNP FPRA in regard to: (1) identification of the specific fire models, tools, and correlations used at CNP, including the specific version of any fire modeling software used; (2) assurance that the fire models and empirical correlations used in the associated analyses were applied within their appropriate scopes and limitations; (3) providing a detailed description of the V&V status for the applied models and correlations; and (4) providing the methods, input data, models, and V&V used for special purposes to analyze several different compartments and fire areas at CNP.

By letter dated April 27, 2012 (Reference 8), the licensee provided a response to these RAIs related to V&V of fire modeling. The following paragraphs describe selected RAI responses related to V&V of fire modeling tools.

During the audit conducted the week of November 7, 2011 (Reference 89), the NRC staff noted that fire modeling was performed in support of the CNP NFPA 805 LAR in the form of a plant-specific Fire Modeling Database (FMDB) and "Transient Analysis Worksheets (TAWs)." The FMDB and TAWs were developed in lieu of using NUREG-1805, "Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," or EPRI Fire Induced Vulnerability Evaluation Methodology, Revision 1 (FIVE Rev. 1).

In RAI 01(a) dated January 27, 2012 (Reference 66), the NRC staff requested additional information from the licensee to ensure that the FMDB and TAWs were coded correctly and that the fire modeling calculation solutions obtained with this tool are the same as those achieved with the FDTs or FIVE Rev1. In response to RAI 01(a) dated April 27, 2012 (Reference 8), the licensee described how the FMDB and TAWs were verified and where this verification was documented. Based on its review and the above explanation, the NRC staff concludes that this response is acceptable.

By letter dated October 11, 2012 (Reference 67), the NRC staff requested additional information related to the fire modeling. By letter dated October 15, 2012 (Reference 11), the licensee provided a response to these additional RAIs. One of those additional RAIs was relevant to V&V of fire modeling and is described below.

During the technical review process, the NRC staff observed that the software package Pyrosim was used to build the FDS input files. RAI 55 (Reference 67) was issued to ask the licensee to provide technical documentation to demonstrate that Pyrosim has been verified to build the FDS input files correctly. The response to RAI 55 (Reference 11) describes how this software package was verified and where this verification is documented. The NRC staff concludes that this response is acceptable.

3.8.3.2.3 Post-Transition

The licensee also stated that it will revise the appropriate processes and procedures to include NFPA 805 quality requirements for use during the performance of post-transition FPP changes, including those for verification and validation. Revision of the applicable post-transition processes and procedures to include NFPA 805 requirements for V&V is identified as implementation item S-3.10 in Attachment S, Table S-3 of the LAR.

3.8.3.2.4 Conclusion for Section 3.8.3.2

Based on the licensee's description of the CNP process for V&V of calculational models and numerical methods, the NRC staff concludes that the licensee's approach to meeting the requirements of NFPA 805, Section 2.7.3.2, is acceptable.

3.8.3.3 Limitations of Use

The licensee stated in the LAR that engineering methods and numerical models used in support of transition to NFPA 805 requirements were used subject to the limitations of use per NFPA 805, Section 2.7.3.3, and that engineering methods and numerical models used post-transition will be subject to these same use limitations. Revision of post-transition processes and procedures, as necessary, to include NFPA 805 requirements for limitations of use, is identified as implementation item S-3.10 in Attachment S, Table S-3 of the LAR. In LAR Section 4.7.3, the licensee stated that the fire models developed to support the NFPA 805 transition at CNP fall within its V&V limitations.

Based on the licensee's description of the CNP process for placing limitations on the use of engineering methods and numerical models, the NRC staff concludes that the licensee's approach to meeting the requirements of NFPA 805, Section 2.7.3.3, is acceptable.

3.8.3.3.1 General

The NRC staff assessed the acceptability of the application of each empirical correlation or other fire models based on the adequacy of the V&V documentation and the model's applicability within its limits. Specifically, the staff used the following criteria in assessing the acceptability of each fire model:

- V&V has been completed and documented in NUREG-1824, and the model is applied within the limits of its applicability;

- The fire model is widely accepted and used by fire protection engineering professionals, is documented in an authoritative publication of the Society of Fire Protection Engineers (SFPE) (e.g., *The SFPE Handbook of Fire Protection Engineering*) (Reference 88), and is applied within the limits of its applicability.

Based on the fire models meeting one or more of these criteria, the NRC staff concludes that the application of each of the models used in the CNP FPRA to support transition to NFPA 805, is acceptable. SE Table 3.8.3.2-1, Attachment A, summarizes the fire models used, how each was applied in the CNP FPRA, the V&V basis for each, and the staff evaluation of each.

3.8.3.3.2 Discussion of RAIs

By letter dated January 27, 2012 (Reference 66), the NRC staff sought additional information. By letter dated April 27, 2012 (Reference 8), the licensee provided a response to these RAIs related to Limitations of Use for the fire models used. The following paragraphs describe selected RAI responses related to V&V of fire modeling tools.

During the November 2011 site audit, the NRC staff observed that part of the fire modeling performed in support of transition to NFPA 805 is described in a supporting document, which describes FDS and CFAST fire modeling studies of plume/HGL interaction, temperature sensitive equipment ZOI, and HGL effects. These are generic studies that can be applied to specific fire areas to help make the fire modeling analysis more efficient. Since these are generic studies and are not simulations of specific fire areas, there are inherent limitations in how the information is applied to specific fire areas throughout the plant.

In RAI 01(g) (Reference 66), the NRC staff requested the licensee to provide the basis of assurance that the use of the conclusions from these studies in subsequent fire modeling analysis was within the limits of applicability. In its response to RAI 01(g) (Reference 8), the licensee provided the basis of assurance that the use of these fire modeling studies were used within the limits of applicability. For example, the results of these generic studies could only be applied to specific fire areas if specific parameters, such as fire area volume, fire area height, sensitive equipment cabinet dimensions/construction, were in the same range as what was simulated in the generic studies.

Based on its review and the above explanation, the NRC staff concludes that this response is acceptable.

3.8.3.3.3 Post-Transition

The licensee also stated that it will revise the appropriate processes and procedures to include the NFPA 805 quality requirements for use during the performance of post-transition FPP changes, including those for limitations of use. Revision of the applicable post-transition processes and procedures to include NFPA 805 requirements for limitations of use is an implementation item.

3.8.3.3.4 Conclusion for Section 3.8.3.3

Based on the licensee's statements that the fire models used to support development of the FPRA were used within their limitations, and the description of the CNP process for placing limitations on the use of engineering methods and numerical models, the NRC staff concludes that the licensee's approach meets the requirements of NFPA 805, Section 2.7.3.3.

3.8.3.4 Qualification of Users

NFPA 805 requires that personnel performing engineering analyses and applying numerical methods (e.g., fire modeling) 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. The licensee's procedures require that cognizant personnel who use and apply engineering analyses and numerical models be competent in the field of application and experienced in the application of the methods, including those personnel performing analyses in support of compliance with 10 CFR 50.48(c).

The licensee also stated that it will revise the appropriate processes and procedures to include NFPA 805 quality requirements for use during the performance of post-transition FPP changes, including those for qualification of users. Revision of the applicable post-transition processes and procedures to include NFPA 805 requirements for qualification of users is an implementation item.

The NRC staff concludes that appropriately competent and experienced personnel developed the CNP FPRA, including the supporting fire modeling calculations and the additional documentation for models and empirical correlations not identified in previous NRC-approved V&V documents.

In addition, based on the licensee's description of the CNP procedures for ensuring personnel who use and apply engineering analyses and numerical methods are competent and experienced, the NRC staff concludes that the licensee's approach for meeting the requirements of NFPA 805, Section 2.7.3.4, is acceptable.

3.8.3.5 Uncertainty Analysis

NFPA 805 requires that an uncertainty analysis be performed to provide reasonable assurance that the performance criteria have been met. (Note: 10 CFR 50.48(c)(2)(iv) states that an uncertainty analysis performed in accordance with NFPA 805, Section 2.7.3.5, is not required to support calculations used in conjunction with a deterministic approach.) The licensee stated that an uncertainty analysis was performed for the analyses used in support of the transition to NFPA 805, and that an uncertainty analysis will be performed for post-transition analyses.

3.8.3.5.1 General

The industry consensus standard for PRA development (i.e. the ASME/ANS PRA standard) includes requirements to address uncertainty. Accordingly, the licensee addressed uncertainty as a part of the development of the CNP FPRA. Table Y-7, "Sources of Uncertainty," in LAR Attachment Y, "Fire PRA Insights," provides a detailed listing of the sources of uncertainty in the FPRA and the licensee's evaluation of each. The NRC staff's evaluation of the licensee's treatment of these uncertainties is discussed in SE Section 3.4.7.

According to NUREG-1855, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," March 2009 (Reference 55), there are three types of uncertainty associated with fire modeling calculations:

(1) Parameter Uncertainty: Input parameters are often chosen from statistical distributions or estimated from generic reference data. In either case, the uncertainty of these input parameters affects the uncertainty of the results of the fire modeling analysis.

(2) Model Uncertainty: Idealizations of physical phenomena lead to simplifying assumptions in the formulation of the model equations. In addition, the numerical solution of equations that have no analytical solution can lead to inexact results. Model uncertainty is estimated via the processes of V&V. An extensive discussion of quantifying model uncertainty can be found in NUREG-1934, "Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG)," November 2012 (Reference 57).

(3) Completeness Uncertainty: This refers to the fact that a model is not a complete description of the phenomena it is designed to simulate. Some consider this a

form of model uncertainty because most fire models neglect certain physical phenomena that are not considered important for a given application. Completeness uncertainty is addressed by the description of the algorithms found in the model documentation. It is addressed, indirectly, by the same process used to address the Model Uncertainty.

3.8.3.5.2 Discussion of Fire Modeling RAIs

By letter dated January 27, 2012 (Reference 66), the NRC staff sought additional information. By letter dated April 27, 2012 (Reference 8), the licensee provided a response to these RAIs related to Limitations of Use for the fire models used. The following paragraphs describe selected RAI responses related to V&V of fire modeling tools.

Section 4.5.1.2 of the CNP NFPA 805 LAR states that uncertainty analyses were performed as required by Section 2.7.3.5 of NFPA 805 and the results were considered in the context of the application.

NRC RAI 02(a) (Reference 66) requested the licensee to explain in detail the uncertainty analyses for fire modeling that was performed, and describe how the uncertainties of the input parameters (geometry, heat release rate, radiative fraction, etc.) were determined and accounted for and substantiate the statement in Appendix J of the CNP NFPA 805 LAR which states that, "...the predictions .....are deemed to be within the bounds of experimental uncertainty..."

In the response to RAI 02(a) (Reference 8), the licensee provided additional information about how uncertainty associated with fire modeling was accounted for in the analysis. Most of this information was originally included in the supporting documentation from each detailed fire modeling report provided by the licensee's fire modeling contractor. The uncertainty analysis performed with respect to fire modeling was qualitative in nature and focused on the fact that conservative model input parameters were used in the fire modeling calculations and that this, in turn, provides a substantial safety margin. The RAI response provided examples of conservative modeling assumptions that lead to safety margin, as follows:

- Fire scenarios involving electrical cabinets (including the electrical split fraction of pump fires) use the 98th percentile HRR for the severity factor calculated out to the nearest FPRA target. This is considered conservative.
- The fire elevation in most cases is at top of cabinet or pump body. This is considered conservative, since the combustion process will occur where the fuel mixes with oxygen, which is not always at the top of the ignition source.
- The radiant fraction used is 0.4. This represents a 33 percent safety margin over the normally recommended value of 0.3.
- The convective HRR fraction used is 0.7. The normally recommended value is between 0.6 and 0.65 and, thus, the use of 0.7 is conservative.

- For transient fire impacts, a large bounding transient zone assumes all targets within its ZOI are affected by a fire. Time-to-damage is calculated based on the most severe (i.e., closest) target. This is considered conservative, since a transient fire would actually have a much smaller ZOI and varying damage times. This approach is implemented to minimize the multitude of transient scenarios to be analyzed.

- For hot gas layer calculations, no equipment or structural steel is credited as a heat sink, since the closed-form correlations used do not account for heat loss to these items.

- Not all cable trays are filled to capacity. Assuming the trays are full provides conservative estimates of the contribution of cable insulation to the fire and the corresponding time-to-damage.

- As the fire propagates to secondary combustibles, the fire is conservatively modeled as one single fire using the fire modeling closed-form correlations. The resulting plume temperature estimates used in this analysis are therefore also conservative, since in reality, the fire would be distributed over a large surface area, and would be less severe at the target location.

- Target damage is assumed to occur when the exposure environment meets or exceeds the damage threshold. No additional time delay due to thermal response is given.

- The fire elevation for transient fires is 2 feet. This is considered conservative since some transient fires occur at the floor.

- Oil fires are analyzed as both unconfined and confined spills with 20-minute durations. Unconfined spills result in large heat release rates, but usually burn for seconds. The oil fires have been conservatively analyzed for 20 minutes to account for the uncertainty in the oil spill size.

- High energy arcing fault (HEAF) scenarios are conservatively assumed to be at peak fire intensity for 20-minutes from time zero, even though the initial arcing fault is expected to consume the contents of the cabinet and burn for only a few minutes.

- Fire brigade intervention is not credited prior to 85-minutes. Fire brigade drills indicated that typical manual suppression times can be expected to be much less (usually 20 minutes).

In addition to this safety margin discussion, the RAI response also provides justification for the statement, "...the predictions .....are deemed to be within the bounds of experimental uncertainty..." This justification is based on a qualitative analysis of several specific model calculations and their disposition in Table 3-1 of NUREG-1824. The NRC staff concludes that this response is acceptable.

NRC RAI 02(b) (Reference 66) requested the licensee justify why cable tray obstructions could be omitted in the FDS fire modeling analysis for Fire Area AA43. This RAI relates specifically to an example of model and completeness uncertainty. The licensee's response to RAI 02(b) (Reference 8) provided justification for omitting cable obstructions by demonstrating that the cable obstructions would not increase the HRR prescribed in FDS, would not affect how the plume and radiant ZOI was applied, and are not expected to significantly affect hot gas and smoke movement within the fire area. The NRC staff concludes that this response is acceptable.

#### 3.8.3.5.3 Post-Transition

The licensee also stated that it will revise the appropriate processes and procedures to include the NFPA 805 quality requirements for use during the performance of post-transition FPP changes, including those regarding uncertainty analysis. Revision of the applicable post-transition processes and procedures to include NFPA 805 requirements regarding uncertainty analysis is an implementation item.

#### 3.8.3.5.4 Conclusion for Section 3.8.3.5

Based on the licensee's description of the CNP process for performing an uncertainty analysis, the NRC staff concludes that the licensee's approach for meeting the requirements of NFPA 805, Section 2.7.3.5, is acceptable.

Based on the above discussions, the NRC staff concludes that the CNP RI/PB FPP quality assurance process adequately addresses each of the requirements of NFPA 805, Section 2.7.3, which include conducting independent reviews, performing V&V, limiting the application of acceptable methods and models to within prescribed boundaries, ensuring that personnel applying acceptable methods and models are qualified, and performing uncertainty analyses.

### 3.8.4 Fire Protection Quality Assurance Program

Criterion 1 of Appendix A to 10 CFR 50 requires:

> Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

The licensee's Fire Protection Quality Assurance Program was established in accordance with the guidelines of NUREG-0800, Standard Review Plan, Section 9.5-1, "Fire Protection," Branch Technical Position, Chemical Engineering Branch (BTP CMEB) 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," Revision 2, July 1981, Position C.4, "Quality Assurance Program" (Reference 90).

NEI 04-02, Appendix C (Reference 5), provides guidance for the LAR to include a description of how the existing fire protection quality assurance (QA) program will be transitioned to the new NFPA 805 RI/PB FPP. CNP states that the FPP QA program will be transitioned to the new NFPA 805 RI/PB FPP and that the FPP QA program is included within and implemented by the CNP nuclear QA program. Certain aspects of that program are not applicable to the FPP. Further, the licensee will revise the QA program to reflect the applicable requirements of Section 2.7.3 of NFPA 805 through implementation item S-3.10 in Attachment S, Table S-3 of the LAR.

The NRC staff concludes that the licensee's changes to the fire protection QA program are reasonable because they include the expansion of the program to include those fire protection systems that were previously not included within the scope of the fire protection QA program required by NFPA 805, Chapter 4.

3.8.5 Conclusion for Section 3.8

The NRC staff reviewed the licensee's RI/PB FPP and RAI responses for Section 3.8 of this SE. The NRC staff concludes that, upon completion of the implementation item related to the QA program, the licensee's approach for meeting the requirements specified in Section 2.7 of NFPA 805 is acceptable.

## **4.0 FIRE PROTECTION LICENSE CONDITION**

The licensee proposed an FPP license condition regarding transition to an RI/PB FPP under NFPA 805, in accordance with 10 CFR 50.48(c)(3)(i). The new license condition adopts the guidelines of the standard fire protection license condition promulgated in RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, Regulatory Position C.3.1, as issued on December 18, 2009 (74 FR 67253). Plant-specific changes were made to the sample license condition; however, the proposed plant-specific FPP license condition is consistent with the standard fire protection license condition, incorporates all of the relevant features of the transition to NFPA 805 at CNP, and is therefore acceptable.

The following license condition is included in the amended licenses for CNP and will replace Renewed Facility Operating License No. DPR-58, Paragraph 2.C.(4) as follows:

(4) Fire Protection Program

Indiana Michigan Power Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, May 1, 2013, June 21, 2013, and September 16, 2013, and as approved in the Safety Evaluation dated October 24, 2013. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(a) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed Fire PRA (FPRA) model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1x10-7/year (yr) for CDF and less than 1x10-8/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) Other Changes that May Be Made Without Prior NRC Approval

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program and Design Elements

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated

October 24, 2013, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2.C.(4)(c)2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2.C.(4)(b)2. above.

2. The licensee shall implement the modifications to its facility, as described in Enclosure 5, Attachment S, Table S-2, "Plant Modifications Committed," of I&M letter AEP-NRC-2013-75, dated September 16, 2013, to complete the transition to full compliance with 10 CFR 50.48(c) by October 24, 2014. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Enclosure 5, Attachment S, Table S-3, "Implementation Items," of I&M letter AEP-NRC-2013-75, dated September 16, 2013, by October 24, 2014.

4. The licensee shall complete an FPRA focused scope peer review and resolve findings associated with the revised FPRA LERF values, prior to self-approval of changes that result in more than a minimal increase in risk.

5. The licensee shall complete a focused-scope peer review and resolve findings of the PRA upgrade related to reduced mission times for cutsets containing a test and maintenance event combined with a running failure, prior to self-approval of changes that result in more than a minimal increase in risk.

The following license condition is included in the amended licenses for CNP and will replace Renewed Facility Operating License No. DPR-74, Paragraph 2.C.(3)(o):

(o) Fire Protection Program

Indiana Michigan Power Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, May 1, 2013, June 21, 2013, and September 16, 2013, and as approved in the Safety Evaluation dated October 24, 2013. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

I. Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed Fire PRA (FPRA) model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

1. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1x10-7/year (yr) for CDF and less than 1x10-8/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

II. <u>Other Changes that May Be Made Without Prior NRC Approval</u>

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program and Design Elements

   Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

   Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated October 24, 2013, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

III. <u>Transition License Conditions</u>

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2.C.(3)(o)(III)2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change

has been demonstrated to have no more than a minimal risk impact, as described in 2.C.(3)(o)(II)2. above.

2. The licensee shall implement the modifications to its facility, as described in Enclosure 5, Attachment S, Table S-2, "Plant Modifications Committed," of I&M letter AEP-NRC-2013-75, dated September 16, 2013, to complete the transition to full compliance with 10 CFR 50.48(c) by October 24, 2014. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Enclosure 5, Attachment S, Table S-3, "Implementation Items," of I&M letter AEP-NRC-2013-75, dated September 16, 2013, by October 24, 2014.

4. The licensee shall complete an FPRA focused scope peer review and resolve findings associated with the revised FPRA LERF values, prior to self-approval of changes that result in more than a minimal increase in risk.

5. The licensee shall complete a focused-scope peer review and resolve findings of the PRA upgrade related to reduced mission times for cutsets containing a test and maintenance event combined with a running failure, prior to self-approval of changes that result in more than a minimal increase in risk.

## 5.0 SUMMARY

The NRC staff reviewed the licensee's license amendment request, as supplemented, to transition to a risk-informed, performance-based fire protection program in accordance with the requirements established by NFPA 805. The staff concludes that the applicant's approach, methods, and data are acceptable to establish, implement and maintain an RI/PB FPP in accordance with 10 CFR 50.48(c).

Implementation of the RI/PB FPP in accordance with 10 CFR 50.48(c) will include the application of a new fire protection license condition. The new license condition includes a list of modifications that must be implemented in order to support the conclusions made in this SE as well as an established date by which full compliance with 10 CFR 50.48(c) will be achieved. In addition, before the licensee is able to fully implement the transition to a fire protection program based on NFPA 805 and apply the new fire protection license condition, to its full extent, a number of implementation items must be completed within the timeframe specified.

## 6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified on June 11, 2013, of the proposed issuance of the amendment. The State official had no comments.

## 7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on October 4, 2011 (76 FR 61396). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

## 8.0 CONCLUSION

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

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NRR = NRC's Office of Nuclear Reactor Regulation
PNNL = Pacific Northwest National Laboratory
CNWRA = Center for Nuclear Waste Regulatory Analysis

Date: October 24, 2013

Attachments:
A. Table 3.8.3.2-1 – V&V Basis for Fire Modeling Correlations Used at CNP
B. Table 3.8.3.2-2 – V&V Basis for Fire Model Calculations of Other Models Used at CNP
C. Abbreviations and Acronyms

**Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at CNP**
(Note: References for Table 3.8.3.2-1 immediately follow the table)

| Correlation | Application at CNP | V&V Basis | NRC Staff Evaluation of Acceptability |
|---|---|---|---|
| Flame Height (Method of Heskestad) | The Flame Height Correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets. The correlation was used to determine the vertical extension of the flame region as part of the Zone of Influence (ZOI) calculations. | NUREG-1805, Chapter 3, 2004 (Reference 1)<br><br>NUREG-1824, Volume 3, 2007 (Reference 2)<br><br>SFPE Handbook, 4th Edition, Chapter 2-1, Heskestad, 2008 (Reference 3) | • Licensee provided verification of the FMDB and Transient Workbook on basis of comparison with NUREG-1805 (Response to RAI 01(a), Reference 4).<br>• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook.<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(b), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Plume Centerline Temperature<br><br>(Method of Heskestad) | The Plume Centerline Temperature correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets. The correlation was used to determine vertical separation distance, based on temperature, to a target in order to determine the vertical extent of the ZOI. | NUREG-1805, Chapter 9, 2004 (Reference 1)<br><br>NUREG-1824, Volume 3, 2007 (Reference 2)<br><br>SFPE Handbook, 4th Edition, Chapter 2-1, Heskestad, 2008 (Reference 3) | • Licensee provided verification of the FMDB and Transient Workbook on basis of comparison with NUREG-1805 (Response to RAI 01(a), Reference 4).<br>• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook.<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(b), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |

**Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at CNP**

(Note: References for Table 3.8.3.2-1 immediately follow the table)

| Correlation | Application at CNP | V&V Basis | NRC Staff Evaluation of Acceptability |
|---|---|---|---|
| Radiant Heat Flux (Point Source Method) | The Radiant Heat Flux (Point Source Method) correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets. The correlation was used to the horizontal separation distance, based on heat flux, to a target in order to determine the horizontal extent of the ZOI. | NUREG-1805, Chapter 5, 2004 (Reference 1)<br><br>NUREG-1824, Volume 3, 2007 (Reference 2)<br><br>SFPE Handbook, 4th Edition, Chapter 3-10, Beyler, C., 2008 (Reference 5) | • Licensee provided verification of the FMDB and Transient Workbook on basis of comparison with NUREG-1805 (Response to RAI 01(a), Reference 4).<br>• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook.<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(b), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Plume Radius (Method of Heskestad) | The Plume Radius (Method of Heskestad) correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets, but was not used to calculate the ZOI. | SFPE Handbook, 4th Edition, Chapter 2-1, Heskestad, G., 2008 (Reference 3) | Licensee stated that the plume radius was not used as the sole basis for any target failures (Response to RAI 01(d), Reference 3). Hence NRC staff did not evaluate the acceptability of this correlation in the CNP application. |
| Hot Gas Layer (Method of McCaffrey, Quintiere, and Harkleroad) | The Hot Gas Layer (Method of McCaffrey, Quintiere, and Harkleroad) correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets. The correlation was used to calculate the hot gas layer temperature for a room with natural ventilation. | NUREG-1805, Chapter 2, 2004. (Reference 1)<br><br>NUREG-1824, Volume 3, 2007. (Reference 2)<br><br>SFPE Handbook, 4th Edition, Chapter 3-6, Walton W. and Thomas, P., 2008 (Reference 6) | • Licensee provided verification of the FMDB and Transient Workbook on basis of comparison with NUREG-1805 (Response to RAI 01(a), Reference 4).<br>• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook.<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(b), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |

**Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at CNP**
(Note: References for Table 3.8.3.2-1 immediately follow the table)

| Correlation | Application at CNP | V&V Basis | NRC Staff Evaluation of Acceptability |
|---|---|---|---|
| Hot Gas Layer (Method of Beyler) | The Hot Gas Layer (Method of Beyler) correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets. The correlation was used to calculate the hot gas layer temperature for a room with no ventilation. | NUREG-1805, Chapter 2, 2004 (Reference 1)<br><br>NUREG-1824, Volume 3, 2007 (Reference 2)<br><br>SFPE Handbook, 4th Edition, Chapter 3-6, Walton W. and Thomas, P., 2008 (Reference 6) | • Licensee provided verification of the FMDB and Transient Workbook on basis of comparison with NUREG-1805 (Response to RAI 01(a), Reference 4).<br>• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook.<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(b), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Hot Gas Layer (Method of Foote, Pagni, and Alvares [FPA]) | The Hot Gas Layer (Method of Foote, Pagni, and Alvares) correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets. The correlation was used to calculate the hot gas layer temperature for a room with forced ventilation. | NUREG-1805, Chapter 2, 2004 (Reference 1)<br><br>NUREG-1824, Volume 3, 2007 (Reference 2)<br><br>SFPE Handbook, 4th Edition, Chapter 3-6, Walton W. and Thomas, P., 2008 (Reference 6) | • Licensee provided verification of the FMDB and Transient Workbook on basis of comparison with NUREG-1805 (Response to RAI 01(a), Reference 4).<br>• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook.<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(b), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |

**Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at CNP**
(Note: References for Table 3.8.3.2-1 immediately follow the table)

| Correlation | Application at CNP | V&V Basis | NRC Staff Evaluation of Acceptability |
|---|---|---|---|
| Hot Gas Layer (Method of Deal and Beyler) | The Hot Gas Layer (Method of Deal and Beyler) correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets. The correlation was used to calculate the hot gas layer temperature for a room with forced ventilation. | NUREG-1805, Chapter 2, 2004 (Reference 1)<br><br>NUREG-1824, Volume 3, 2007 (Reference 2)<br><br>SFPE Handbook, 4th Edition, Chapter 3-6, Walton W. and Thomas, P., 2008. (Reference 6) | • Licensee provided verification of the FMDB and Transient Workbook on basis of comparison with NUREG-1805 (Response to RAI 01(a), Reference 4).<br>• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook.<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(b), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Ceiling Jet Temperature (Method of Alpert) | The Ceiling Jet Temperature (Method of Alpert) correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets. The correlation was used to calculate horizontal separation distance, based on temperature at the ceiling of a room, to a target in order to determine the horizontal extent of the ZOI. | NUREG-1824, Volume 4, 2007 (Reference 2)<br><br>SFPE Handbook, 4th Edition, Chapter 2-2, Alpert, R., 2008 (Reference 7) | • Licensee provided verification of the FMDB and Transient Workbook on basis of comparison with FIVE-Rev1 (Response to RAI 01(a), Reference 4).<br>• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook.<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(b), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |

**Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at CNP**
(Note: References for Table 3.8.3.2-1 immediately follow the table)

| Correlation | Application at CNP | V&V Basis | NRC Staff Evaluation of Acceptability |
|---|---|---|---|
| Smoke Detection Actuation Correlation (Method of Heskestad and Delichatsios) | Smoke Detection Actuation (Method of Heskestad and Delichatsios) correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets. The Ceiling Jet Temperature (Method of Alpert) correlation was used to determine the ceiling jet temperature that is used as input for smoke detector activation and then Heskestad and Delichatsios method was used to calculate the activation time. The correlation was used to calculate smoke detection timing. | NUREG-1805, Chapter 11, 2004 (Reference 1)<br><br>NUREG-1824, Volume 3, 2007 (Reference 2)<br><br>SFPE Handbook, 4th Edition, Chapter 4-1, Custer R., Meacham B., and Schifiliti, R., 2008. (Reference 8)<br><br>SFPE Handbook, 4th Edition, Chapter 2-2, Alpert, R., 2008. (Reference 7) | • Licensee provided verification of the FMDB and Transient Workbook on basis of comparison with NUREG-1805 (Response to RAI 01(a), Reference 4).<br>• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(b), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Heat Detection Actuation Correlation | None. | NUREG-1805, Chapter 11, 2004. (Reference 1)<br><br>NFPA Handbook, 19th Edition, Chapter 3-9, Budnick, E., Evans, D., and Nelson, H., 2003 (Reference 8) | In response to RAI 01(c) (Reference 4), licensee stated that Heat Detection Actuation Correlation was not used in fire modeling at CNP. Hence, NRC staff did not evaluate the acceptability of this correlation in the CNP application. |

**Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at CNP**
(Note: References for Table 3.8.3.2-1 immediately follow the table)

| Correlation | Application at CNP | V&V Basis | NRC Staff Evaluation of Acceptability |
|---|---|---|---|
| Sprinkler Activation Correlation | Sprinkler Activation Correlation was implemented in the Fire Modeling Database (FMDB) and Transient Worksheets. The correlation was used to estimate sprinkler actuation timing based on ceiling jet temperature, velocity, and thermal response of sprinkler. | NUREG-1805, Chapter 10, 2004 (Reference 1)<br><br>NFPA Handbook, 19$^{th}$ Edition, Chapter 3-9, Budnick, E., Evans, D., and Nelson, H., 2003. (Reference 9) | • Licensee provided verification of the FMDB and Transient Workbook on basis of comparison with NUREG-1805 (Response to RAI 01(c), Reference 4).<br>• The correlation is validated in an authoritative publication of the NFPA Handbook.<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(b), Reference 4).<br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |

**References for Attachment A: Table 3.8.3.2-1**


1. NUREG-1805, "Fire Dynamics Tools (FDT[s]) Quantitative Fire Hazard Analysis Methods for the U. S. Nuclear Regulatory Commission Fire Protection Inspection Program," U.S. Nuclear Regulatory Commission, Washington, DC, December 2004.
2. NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," U.S. Nuclear Regulatory Commission, Washington, DC, May 2007.
3. Heskestad, G., "Fire Plumes, Flame Height, and Air Entrainment," Chapter 2–1 of *The SFPE Handbook of Fire Protection Engineering*, 4th Edition, P. J. DiNenno, Editor-in-Chief, National Fire Protection Association, Quincy, MA, 2008.
4. Letter from M. H. Carlson, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Response to Request for Additional Information Regarding the Application for Amendment to Transition the Fire Protection Program to National Fire Protection Association Standard 805 (TAC Nos. ME6629 AND ME6630)," AEP-NRC-2012-29, dated April 27, 2012, ADAMS Accession No. ML12132A390.
5. Beyler, C. L., "Fire Hazard calculations for Large, Open Hydrocarbon Fires" Chapter 3–10 of *The SFPE Handbook of Fire Protection Engineering*, 4th Edition, P. J. DiNenno, Editor-in-Chief, National Fire Protection Association, Quincy, MA, 2008.
6. Walton W. and Thomas, P., "Estimating Temperatures in Compartment Fires" Chapter 3–6 of *The SFPE Handbook of Fire Protection Engineering*, 4th Edition, P. J. DiNenno, Editor-in-Chief, National Fire Protection Association, Quincy, MA, 2008.
7. Alpert, R. L., "Ceiling Jet Flows," Chapter 2–2 of *The SFPE Handbook of Fire Protection Engineering*, 4th Edition, P. J. DiNenno, Editor-in-Chief, National Fire Protection Association, Quincy, MA, 2008.
8. Custer R.L.P., Meacham B. J., and Schifiliti, R. P, "Design of Detection Systems," Chapter 4–1 of *The SFPE Handbook of Fire Protection Engineering*, 4th Edition, P. J. DiNenno, Editor-in-Chief, National Fire Protection Association, Quincy, MA, 2008.
9. Budnick, E.K., D.D. Evans, and H.E. Nelson, "Simplified Fire Growth Calculations" Section 3, Chapter 9, *NFPA Fire Protection Handbook*, 19 Edition, A.E. Cote, Editor-in-Chief, National Fire Protection Association, Quincy, Massachusetts, 2003.

**Attachment B: Table 3.8.3.2-2, V&V Basis for Fire Model Calculations of Other Models Used at CNP**

(Note: References for Table 3.8.3.2-2 immediately follow the table)

| Calculation | Application at CNP | V&V Basis | NRC Staff Evaluation of Acceptability |
|---|---|---|---|
| Hot Gas Layer Calculations using Fire Dynamics Simulator (Version 5) | Fire Dynamics Simulator (Version 5) was to calculate hot gas layer height and temperatures. | NUREG-1824, Volume 7, 2007 (Reference 1)<br><br>NIST Special Publication 1018-5, Volume 2: Verification (Reference 2)<br><br>NIST Special Publication 1018-5, Volume 3: Validation (Reference 3) | • The modeling technique is validated in NUREG-1824 (Reference 1) and an authoritative publications of NIST (References 2 and 3).<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(e), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Hot Gas Layer Calculations using Consolidated Model of Fire Growth and Smoke Transport (CFAST) (Version 6) | CFAST (Version 6) was used to calculate upper and lower layer temperatures for various compartments, the layer height, and smoke obscuration. It was also used to calculate abandonment time for the CNP main control room. | NUREG-1824, Volume 5, 2007 (Reference 1)<br><br>NIST Special Publication 1086, 2008 (Reference 5) | • The modeling technique is validated in NUREG-1824 (Reference 1) and an authoritative publication of NIST (Reference 5).<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(f), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Temperature Sensitive Equipment Hot Gas Layer Study | CFAST (Version 6) was used to calculate the upper and lower gas layer temperatures for various compartments, and the layer height, for use in assessment of damage to temperature sensitive equipments. | NUREG-1824, Volume 5, 2007 (Reference 1)<br><br>NIST Special Publication 1086, 2008 (Reference 5) | • The modeling technique is validated in NUREG-1824 (Reference 1) and an authoritative publication of NIST (Reference 5).<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(f), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |

**Attachment B: Table 3.8.3.2-2, V&V Basis for Fire Model Calculations of Other Models Used at CNP**
(Note: References for Table 3.8.3.2-2 immediately follow the table)

| Calculation | Application at CNP | V&V Basis | NRC Staff Evaluation of Acceptability |
|---|---|---|---|
| Temperature Sensitive Equipment Zone of Influence Study | FDS (Version 5) was used to calculate the radiant heat flux ZOI at which temperature sensitive equipment will reach damage thresholds. | NUREG-1824, Volume 7, 2007 (Reference 1)<br><br>NIST Special Publication 1018-5, Volume 2: Verification (Reference 2)<br><br>NIST Special Publication 1018-5, Volume 3: Validation (Reference 3) | • The modeling technique is validated in NUREG-1824 (Reference 1) and authoritative publications of NIST (References 2 and 3).<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(e), Reference 4).<br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Plume/Hot Gas Layer Interaction Study | FDS (Version 5) was used to locate the point where hot gas layer and plume interact and establish limits for plume temperature application. | NUREG-1824, Volume 7, 2007 (Reference 1)<br><br>NIST Special Publication 1018-5, Volume 2: Verification (Reference 2)<br><br>NIST Special Publication 1018-5, Volume 3: Validation (Reference 3) | • The modeling technique is validated in NUREG-1824 (Reference 1) and authoritative publications of NIST (References 2 and 3).<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(e), Reference 4).<br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Corner and Wall HRR | The corner and wall HRR was used to adjust the heat release for fires near a wall and corner | SFPE Handbook, 4th Edition, Chapter 2-14, Lattimer, 2008 (Reference 6)<br><br>Zukoski (Reference 7)<br>Sargent (Reference 8)<br><br>Cetegen (Reference 9)<br><br>Williamson (Reference 10). | • The modeling technique is documented in the authoritative publication of SFPE (Reference 6).<br>• Licensee stated that in most cases, the correlation has been applied within the validated range applied within the validated range reported in the studies in references 7-10. Licensee provided justification for cases where the correlation was used outside validated range reported in these authoritative publications (Response to RAI 01(c), Reference 4).<br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |

**Attachment B: Table 3.8.3.2-2, V&V Basis for Fire Model Calculations of Other Models Used at CNP**
(Note: References for Table 3.8.3.2-2 immediately follow the table)

| Calculation | Application at CNP | V&V Basis | NRC Staff Evaluation of Acceptability |
|---|---|---|---|
| Correlation for Heat Release Rates of Cables (Method of Lee) | Method of Lee was used to correlate bench scale data to heat release rates from cable tray fires. | SFPE Handbook, 4th Edition, Chapter 3-1, Babrauskas, 2008 (Reference 11).<br><br>Lee (Reference 12) | • The modeling technique is documented in the authoritative publication of SFPE (Reference 11).<br>• Licensee stated that in most cases, the correlation has been applied in configurations similar to that reported by Lee (Reference 12). Licensee provided justification for cases where the correlation was used outside the configuration reported in the authoritative publication (Response to RAI 01(c), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Correlation for Flame Spread over Horizontal Cable Trays (FLASH-CAT) | The FLASH-CAT method was used to calculate the growth and spread of a fire within a vertical stack of horizontal cable trays | NUREG/CR-7010, Section 9, 2010 (Reference 13) | • The modeling technique is validated in NUREG/CR-7010 (Reference 13).<br>• Licensee stated that in most cases, the correlation has been applied in configurations similar to that reported in NUREG/CR-7010. Licensee provided justification for cases where the correlation was used outside the configuration reported in the authoritative publication (Response to RAI 01(c), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |
| Fire Door Closure Calculation using FDS (Version 5) | FDS (Version 5) was used to evaluate that the door thermal link will activate prior to cable damage. | NUREG-1824, Volume 7, 2007 (Reference 1)<br><br>NIST Special Publication 1018-5, Volume 2: Verification (Reference 2)<br><br>NIST Special Publication 1018-5, Volume 3: Validation (Reference 3) | • The modeling technique is validated in NUREG-1824 (Reference 1) and authoritative publications of NIST (References 2 and 3).<br>• Licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. Licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to RAI 01(e), Reference 4).<br><br>Based on these observations, the NRC staff concludes the use of this correlation in the CNP application is acceptable. |

**References for Attachment B: Table 3.8.3.2-2**


1. NUREG-1824, "Verification & Validation of Selected Fire Models for Nuclear Power Plant Applications," U.S. Nuclear Regulatory Commission, Washington, DC, May 2007.

2. McDermott, R., McGrattan, K., Hostikka, S., Floyd, J. "Fire Dynamics Simulator (Version 5) Technical Reference Guide Volume 2: Verification," NIST Special Publication 1018-5, National Institute of Standards and Technology, Gaithersburg, MD, 2010

3. McGrattan, K., Hostikka, S., Floyd, J, McDermott, R. "Fire Dynamics Simulator (Version 5) Technical Reference Guide Volume 3: Validation," NIST Special Publication 1018-5, National Institute of Standards and Technology, Gaithersburg, MD, 2010

4. Letter from M. H. Carlson, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Response to Request for Additional Information Regarding the Application for Amendment to Transition the Fire Protection Program to National Fire Protection Association Standard 805 (TAC Nos. ME6629 AND ME6630)," AEP-NRC-2012-29, dated April 27, 2012, ADAMS Accession No. ML12132A390.

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## Attachment C: Abbreviations and Acronyms

| | |
|---|---|
| AC | alternating current |
| ADAMS | Agencywide Documents Access and Management System |
| AFW | auxiliary feedwater |
| AHJ | authority having jurisdiction |
| ALARA | as low as reasonably achievable |
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute |
| ASD | alternate shutdown |
| ASME | American Society of Mechanical Engineers |
| BAT | boric acid tank |
| BWR | boiling-water reactor |
| CCW | component cooling water |
| CDF | core damage frequency |
| CFAST | consolidated model of fire and smoke transport |
| CFR | *Code of Federal Regulations* |
| CHRISTIFIRE | Cable Heat Release, Ignition, and Spread in Tray Installations During Fire |
| CNP | Donald C. Cook Nuclear Plant |
| CNWRA | Center for Nuclear Waste Regulatory Analysis |
| CPT | control power transformer |
| CST | condensate storage tank |
| CT | current transfer |
| CVCS | chemical and volume control system |
| DC | direct current |
| DID | defense-in-depth |
| DID-RA | defense-in-depth recovery action |
| ECCS | emergency core cooling system |
| EEEE | existing engineering equivalency evaluation |
| EPRI | Electric Power Research Institute |
| ERFBS | electrical raceway fire barrier system |
| ERO | emergency response organization |
| ESW | emergency service water |
| ESW | essential service water |
| F&O | facts and observations |
| F&S | findings and suggestions |
| FAQ | frequently asked question |
| FDS | fire dynamics simulator |
| FDT | fire dynamics tool |
| FIVE | Fire Induced Vulnerability Evaluation Methodology |
| FLASH-CAT | Flame Spread over Horizontal Cable Trays |
| FMDB | fire modeling database |
| FPP | fire protection program |
| FPRA | fire probabilistic risk assessment |
| FR | *Federal Register* |
| FRE | fire risk evaluation |
| FSA | fire safety analysis |
| FSAR | final safety analysis report |

| | |
|---|---|
| GDC | general design criteria |
| GET | general employee training |
| GL | generic letter |
| GPM | gallons per minute |
| HEAF | high-energy arcing fault |
| HEP | human error probability |
| HGL | hot gas layer |
| HRE | high(er) risk evolution |
| HRR | heat release rate |
| I&M | Indiana Michigan Power Company |
| IN | information notice |
| KSF | key safety function |
| kV | kilovolt |
| kW | kilowatt |
| LAR | license amendment request |
| LERF | large early release frequency |
| LFS | limiting fire scenario |
| MCB | main control board |
| MCC | motor control center |
| MCR | main control room |
| MEFS | maximum expected fire scenario |
| MSO | multiple spurious operation |
| NEI | Nuclear Energy Institute |
| NFPA | National Fire Protection Association |
| NPO | non-power operation |
| NRC | U.S. Nuclear Regulatory Commission |
| NRR | Office of Nuclear Reactor Regulation |
| NSCA | nuclear safety capability assessment |
| NSPC | nuclear safety performance criteria |
| ODCM | offsite dose calculation manual |
| OMA | operator manual action |
| P&IDS | piping and instrumentation drawings |
| PB | performance-based |
| PCS | primary control station |
| PNNL | Pacific Northwest National Laboratory |
| PORV | power-operated relief valve |
| POS | plant operational state |
| PRA | probabilistic risk assessment |
| PSA | probabilistic safety assessment |
| PWR | pressurized-water reactor |
| PWROG | PWR Owners Group |
| QA | quality assurance |
| RA | recovery action |
| RAI | request for additional information |
| RCP | reactor coolant pump |
| RCS | reactor coolant system |
| RG | regulatory guide |
| RHR | residual heat removal |

| | |
|---|---|
| RI | risk-informed |
| RI/PB | risk-informed, performance-based |
| RIS | regulatory issue summary |
| RTI | response time index |
| RWST | refueling water storage tank |
| SE | safety evaluation |
| SER | safety evaluation report |
| SFPE | Society of Fire Protection Engineers |
| SG | steam generator |
| SR | supporting requirement |
| SSC | structures, systems, and components |
| SSCA | safe shutdown capability assessment manual |
| T&M | test and maintenance |
| TDAFW | turbine-driving auxiliary feedwater |
| TI-SGTR | thermally-induced steam generator tube rupture |
| TR | technical/topical report |
| TS | technical specifications |
| UFSAR | updated final safety analysis report |
| V&V | verification and validation |
| VAC | volts alternating current |
| VCT | volume control tank |
| VDC | volts direct current |
| VEWFDS | very early warning fire detection system |
| VFDR | variance from deterministic requirements |
| WOG | Westinghouse Owners Group |
| YD | yard |
| ZOI | zone of influence |

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

**/RA/**

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Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 322 to DPR-58
2. Amendment No. 305 to DPR-74
3. Safety Evaluation

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