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

January 13, 2014

Mr. Adam C. Heflin
Senior Vice President and
Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT, UNIT 1 – ISSUANCE OF AMENDMENT REGARDING TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c) (TAC NO. ME7046)

Dear Mr. Heflin:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 206 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1 (Callaway). The amendment consists of changes to the license and Technical Specifications (TSs) in response to your application dated August 29, 2011, as supplemented by letters dated November 9, 2011, April 17 and July 12, 2012, and February 19, August 5, September 24, and December 19, 2013. Union Electric Company (dba Ameren Missouri, the licensee), submitted a license amendment request (LAR) to revise the fire protection program in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.48(c), for Callaway and change the license and TSs accordingly.

The proposed amendment would transition the Callaway fire protection program to a risk-informed, performance-based program based on National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (NFPA 805), in accordance with 10 CFR 50.48(c). NFPA 805 allows the use of performance-based methods such as fire modeling and risk-informed methods such as fire probabilistic risk assessment to demonstrate compliance with the nuclear safety performance criteria.

The fire protection license condition in Callaway's license is revised to reflect the use of NFPA 805. To assure proper pagination of the license, the NRC is issuing license pages 3 through 11, but the only changes 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,

CF Lyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Amendment No. 206 to NPF-30
2. Safety Evaluation

cc w/encls: Distribution via ListServ

ENCLOSURE 1

AMENDMENT NO. 206

TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 206
License No. NPF-30

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

   A. The application for amendment by Union Electric Company (UE, the licensee), dated August 29, 2011, as supplemented by letters dated November 9, 2011, April 17 and July 12, 2012, and February 19, August 5, September 24, and December 19, 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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-30 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan*

The Technical Specifications contained in Appendix A, as revised through Amendment No. 206 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

In addition, the license is amended as indicated in the attachment to this license amendment, and Paragraph 2.C.(5) of Facility Operating License No. NPF-30 is hereby amended to read as follows:

(5) Fire Protection Program

Union Electric shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated 8/29/2011 (and supplements dated 11/9/2011, 4/17/2012, 7/12/2012, 2/19/2013, 8/5/2013, 9/24/2013, and 12/19/2013) and as approved in the safety evaluation report dated 01/13/2014. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

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

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

(b) Prior NRC review and approval is not required for individual changes that result in a risk increase less than $1x10^{-7}$/year (yr) for core damage frequency (CDF) and less than $1x10^{-8}$/yr for large early release frequency (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.

<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 report dated 01/13/2014 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.

<u>Transition License Conditions</u>

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

(2) The licensee shall implement the items listed in Enclosure 2, Attachment S, Table S-3, "Implementation Items," of Ameren Missouri letter ULNRC-06060, dated December 19, 2013, by 8 months from the date of issuance of the license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented by 8 months from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-30 and Technical Specifications

Date of Issuance: January 13, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 206

TO FACILITY OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Replace the following pages of Facility Operating License NPF-30 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3 through 8	3 through 10

Replace the following page of Appendix A, Technical Specifications, with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
5.0-5	5.0-5

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

(5) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I 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

UE is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan*

The Technical Specifications contained in Appendix A, as revised through Amendment No. 206 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Environmental Qualification (Section 3.11, SSER #3)**

Deleted per Amendment No. 169.

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* Amendments 133, 134, & 135 were effective as of April 30, 2000 however these amendments were implemented on April 1, 2000.

** The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(4) Surveillance of Hafnium Control Rods (Section 4.2.3.1(10), SER and SSER #2)

Deleted per Amendment No. 169.

(5) Fire Protection Program

Union Electric shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated 8/29/2011 (and supplements dated 11/9/2011, 4/17/2012, 7/12/2012, 2/19/2013, 8/5/2013, 9/24/2013, and 12/19/2013) and as approved in the safety evaluation report dated 1/13/2014. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

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

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

(b) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for core damage frequency (CDF) and less than 1×10^{-8}/yr for large early release frequency (LERF). The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program 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 report dated 1/13/2014 to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

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

(2) The licensee shall implement the items listed in Enclosure 2, Attachment S, Table S-3, "Implementation Items," of Ameren Missouri letter ULNRC-06060, dated December 19, 2013, by 8 months from the date of issuance of the license amendment.

(6) Qualification of Personnel (Section 13.1.2, SSER #3, Section 18, SSER #1)

Deleted per Amendment No. 169.

(7) NUREG-0737 Conditions (Section 22, SER)

Deleted per Amendment No. 169.

(8) Post-Fuel-Loading Initial Test Program (Section 14, SER)

Deleted per Amendment No. 169.

(9) Inservice Inspection Program (Sections 5.2.4 and 6.6, SER)

Deleted per Amendment No. 169.

(10) Emergency Planning

Deleted per Amendment No. 169.

(11) Steam Generator Tube Rupture (Section 15.4.4, SSER #3)

Deleted per Amendment No. 169.

(12) Low Temperature Overpressure Protection (Section 15, SSER #3)

Deleted per Amendment No. 169.

(13) LOCA Reanalysis (Section 15, SSER #3)

Deleted per Amendment No. 169.

(14) Generic Letter 83-28

Deleted per Amendment No. 169.

(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. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available, pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

(16) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 190, are hereby incorporated into this license. UE shall operate the facility in accordance with the Additional Conditions.

D. An Exemption from certain requirements of Appendix J to 10 CFR Part 50, are described in the October 9, 1984 staff letter. This exemption is authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, this exemption is hereby granted pursuant to 10 CFR 50.12. With the granting of this exemption the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. UE shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 10 CFR 73.21, are entitled: "Callaway Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 0" submitted by letter dated October 20, 2004, as supplemented by the letter May 11, 2006.

UE shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Callaway Plant Unit 1 CSP was approved by License Amendment No. 203.

F. Deleted per Amendment No. 169.

G. UE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at Midnight on October 18, 2024.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY H. R. DENTON

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Attachment 1 (Deleted per Amendment No. 169)
2. Attachment 2 (Deleted per Amendment No. 169)
3. Appendix A - Technical Specifications (NUREG-1058, Revision 1)
4. Appendix B - Environmental Protection Plan
5. Appendix C - Additional Conditions

Date of Issuance: October 18, 1984

ATTACHMENT 1

Deleted per Amendment No. 169.

ATTACHMENT 2

Deleted per Amendment No. 169.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Not Used; and

e. All programs specified in Specification 5.5.

<u>ENCLOSURE 2</u>

<u>SAFETY EVALUATION BY THE</u>

<u>OFFICE OF NUCLEAR REACTOR REGULATION</u>

<u>TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED</u>

<u>FIRE PROTECTION PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c)</u>

<u>AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE NO. NPF-30</u>

<u>UNION ELECTRIC COMPANY</u>

<u>CALLAWAY PLANT, UNIT 1</u>

<u>DOCKET NO. 50-483</u>

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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

RELATED TO AMENDMENT NO. 206 TO

FACILITY OPERATING LICENSE NO. NPF-30

TRANSITION TO A PERFORMANCE-BASED FIRE PROTECTION

PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c)

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

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, in the *Federal Register* (FR) on November 19, 1980 (45 FR 76602), adding Section 50.48, "Fire Protection" and Appendix R to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." Section 50.48(a)(1) of 10 CFR Part 50 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 Part 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. Section 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. Section 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, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" (Reference 1), which describes a methodology for establishing fundamental fire protection program (FPP) design requirements and elements, determining required fire protection systems and features, applying PB 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 PB or deterministic approaches to be used to meet performance criteria.

Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1 (RG 1.205) (Reference 2), states, in part, that:

> On March 26, 1998, the staff sent to the Commission SECY-98-058, "Development of a Risk-Informed, Performance-Based Regulation for Fire Protection at Nuclear Power Plants" [Reference 83], 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 84], 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 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 PB 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 establish plant fire protection requirements using the methodology in Chapter 2 of NFPA 805, such that the minimum fire protection program elements and design criteria contained in Chapter 3 of NFPA 805 are satisfied. Next, an adoptee identifies fire areas and fire hazards through a plant-wide analysis, and then applies either a PB or a deterministic approach to meet the performance criteria. As part of a PB 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 805 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 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 supplements 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, NEI [the Nuclear Energy Institute] published implementing guidance for the specific provisions of NFPA 805 and 10 CFR 50.48(c) in NEI 04-02, ["Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)."]

RG 1.205 provides the NRC staff's position on NEI 04-02, Revision 2 (Reference 3), 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 a risk-informed, performance-based (RI/PB) FPP.

Accordingly, Union Electric Company (dba Ameren Missouri, the licensee), requested a license amendment to allow the licensee to revise the Callaway Plant, Unit 1 (Callaway) FPP in accordance with 10 CFR 50.48(c) and change the license and technical specifications (TSs) accordingly.

1.2 Requested Licensing Action

By application to the NRC dated August 29, 2011 (Reference 4), as supplemented by letters dated November 9, 2011 (Reference 5), April 17, 2012 (Reference 6), July 12, 2012 (Reference 7), February 19, 2013 (Reference 8), August 5, 2013 (Reference 9), September 24, 2013 (Reference 10), and December 19, 2013 (Reference 88) the licensee submitted an application for a license amendment to transition the Callaway 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 supplemental letters were in response to the NRC staff's requests for additional information (RAIs) dated March 2, 2012 (Reference 11), June 6, 2012 (Reference 12), June 19, 2012 (Reference 13), December 11, 2012 (Reference 14), July 30, 2013 (Reference 15), August 16, 2013 (Reference 16), and November 14, 2013 (Reference 89). The licensee's supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on February 14, 2012 (77 FR 8294).

The licensee requested an amendment to the Callaway operating license and TSs 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 - which was established in accordance with the Standardized Nuclear Unit Power Plant System (SNUPPS) Final Safety Analysis Report (FSAR) for the facility through Revision 15, the Callaway site addendum through Revision 8, and as approved in NUREG-0830, "Safety Evaluation Report Related to the Operation of Callaway Plant, Unit No. 1," dated October 1981 and Supplements 1 through 4 dated April 1982, July 1983, May 1984, and October 1984, respectively (Reference 57) - to a PB 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 Callaway is referred to as RI/PB throughout this SE.

In its license amendment request (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 TS 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 FPP, 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).

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 and Section 2.4.3 discusses TS changes.

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 1). 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 [10 CFR 50.48(b)] for plants licensed to operate before January 1, 1979, or the fire protection license conditions for plants licensed to operate after January 1, 1979. The licensee shall submit a request to comply with NFPA 805 in the form of an application for license amendment under [10 CFR] 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, 33548; June 16, 2004), which states:

> 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 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 (Reference 2). 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 fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage
>
> (3) Providing an adequate level of fire protection for structures, systems and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed

In addition, in accordance with GDC 3, "Fire protection," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, fire detection and fighting systems must be designed such that their rupture or inadvertent operation does not significantly impair the ability of the SSCs important to safety to perform their intended safety functions.

2.1 Applicable Regulations

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.

- 10 CFR 50.48(a)(1) requires that each holder of an operating license have a fire protection plan that satisfies GDC 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, regulatory guides, and standards:

- RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, issued December 2009 (Reference 2), provides guidance for use in complying with the requirements that the NRC has promulgated for RI/PB 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 provides exceptions to 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" (Reference 1), 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, which had not been explicitly addressed by previous requirements and guidelines. NFPA 805 was developed to provide a comprehensive RI/PB 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 and radioactive release. 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)," Revision 2 (Reference 3), provides guidance for implementing the requirements of 10 CFR 50.48(c), and represents methods for implementing in whole or in part an RI/PB 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 RI/PB fire protection goals, objectives, and performance criteria contained in NFPA 805 and the RI/PB 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 17), provides the NRC staff's recommendations for using risk information in support of licensee-initiated licensing basis changes to a nuclear power plant that require such review and approval: The guidance provided does not preclude other approaches for requesting licensing basis 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 licensing basis 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 18), 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 Nuclear Power Plants," Revision 2, issued October 2009 (Reference 19), 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 9.5.1.1, "Fire Protection Program," Revision 0, issued February 2009 (Reference 20), provides the NRC staff with guidance for evaluating LARs related to deterministic FPPs. Previous revisions of this section of NUREG-0800 were issued as Section 9.5.1.

- NUREG-0800, Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection Program," Revision 0, issued December 2009 (Reference 21), 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 License Amendment Requests After Initial Fuel Load," Revision 3, issued September 2012 (Reference 22), 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 23), 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 24, 25, and 26), presents a compendium of methods, data, and tools to perform a fire probabilistic risk assessment (FPRA) 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 a state-of-art FPRA methodology. Both RES and EPRI 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 FPRA. These methods are expected to form a basis for risk-informed analyses related to the plant FPP. Volume 1, the Executive Summary, provides general background and overview information, project insights and conclusions. Volume 2 provides the detailed discussion of the recommended approach, methods, data, and tools for conduct of an FPRA.

- Memorandum from Richard P. Correia, RES, to Joseph G. Giitter, NRR, titled "Interim Technical Guidance on Fire-Induced Circuit Failure Mode Likelihood Analysis," dated June 14, 2013, (Reference 68) notes that, based on new experimental information documented in NUREG/CR-6931 "Cable Response to Live Fire (CAROLFIRE)" issued April 2008 (Reference 86), and NUREG/CR-7100 "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire): Test Results," issued April 2012 (Reference 87), the reduction in hot short probabilities for circuits provided with control power transformers (CPTs) 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 27), provides quantitative methods, known as FDTs, to assist regional fire protection inspectors in performing fire hazard analysis. The FDTs are intended to assist fire protection inspectors in performing RI evaluations of credible fires that may cause critical damage to essential safe shutdown equipment.

- NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," Volumes 1 through 7 (Reference 28), provide technical documentation regarding the predictive capabilities of a specific set of fire models for the analysis of fire hazards in nuclear power plant (NPP) 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) The Fire-Induced Vulnerability Evaluation, -Revision 1 (FIVE) developed by EPRI (Volume 4)
    (3) The zone model, Consolidated Model of Fire and Smoke Transport (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), Phase 1: Horizontal Trays," Volume 1 (Reference 29), describes Phase 1 of the CHRISTIFIRE testing program conducted by NIST. The overall goal of this multiyear 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 30), provides guidance on how to treat uncertainties associated with PRA in RI 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 RI decision-making process itself.

- NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines – Final Report" (Reference 31), 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 (HEPs) for human failure events (HFEs) following the fire-induced initiating events of an FPRA. The report builds on existing HRA methods, and is intended primarily for practitioners conducting a fire HRA to support an FPRA.

- NUREG-1934, "Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG)" (Reference 32), 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 NPP fire hazard analyses. The report also provides information to assist fire model users in applying this technology in the NPP environment.

- Generic Letter (GL) 2006-03. "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations" (Reference 33), 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. Specifically, NRC testing revealed that, for the configurations tested, Hemyc and MT fire barriers failed to provide the protective function intended for compliance with existing regulations.

- NEI 00-01, "Guidance for Post Fire Safe Shutdown Circuit Analysis," Revision 2 (Reference 34), 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 licensing basis) that may be used in conjunction with the deterministic methods for resolving circuit failure issues related to Multiple Spurious Operations (MSOs). The RI 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 35), provides guidance for PRAs used to support RI decisions for commercial light-water reactor NPPs 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 RI decision-making related to design, licensing, procurement, construction, operation, and maintenance.

- Branch Technical Position (BTP) Chemical Engineering Branch (CMEB) 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," Revision 2, July 1981 (Reference 36), provides the NRC staff with guidance for implementing a deterministic FPP in accordance with 10 CFR 50, Section 50.48 and Appendix R.

- NFPA 13, "Standard for the Installation of Sprinkler Systems," 1976 Edition (Reference 37), is the industry benchmark for design and installation of automatic fire sprinkler systems. NFPA 13 addresses sprinkler system design approaches, system installation, acceptance testing, and component options.

- NFPA 14, "Standard for the Installation of Standpipe and Hose Systems" (Reference 38), provides the minimum requirements for the installation of standpipes and hose systems to ensure that systems will work as intended to deliver adequate and reliable water supplies in a fire emergency. NFPA 14 covers all system components and hardware, including piping, fittings, valves, and pressure-regulation devices, as well as system requirements; installation requirements; design; plans and calculations; water supply; and system acceptance.

- Regulatory Issue Summary (RIS) 2004-03, Revision 1, "Risk-Informed Approach for Post-Fire Safe-Shutdown Circuit Inspections," dated December 29, 2004 (Reference 39), informed the industry that the NRC has risk-informed its inspection procedure for post-fire safe shutdown circuit analysis inspections to concentrate inspections on circuit failures that have a relatively high likelihood of occurrence. The RIS describes three categories, or bins, of circuit failure likelihood and the inspection process used to assess circuit configurations in each of the three bins. This RIS also describes the process the NRC will use to implement the Reactor Oversight Process for post-fire safe shutdown circuit inspection findings.

- IN 84-09, Revision 1, "Lessons Learned from NRC Inspections of Fire Protection Safe Shutdown Systems (10 CFR 50, Appendix R)," dated March 7, 1984 (Reference 41), provides the industry with supplemental guidance on meeting the fire protection safe shutdown requirements in 10 CFR 50 Appendix R. IN 84-09 includes supplemental guidance on establishing fire areas, fire barrier testing and configuration, protection of equipment necessary to achieve hot shutdown, performing reassessments for conformance with Appendix R, identification of safe shutdown systems and components, assessing combustibility of electrical cable insulation, detection and automatic suppression, instrumentation and procedures necessary for alternative shutdown, fire protection features for cold shutdown systems, and configuration of reactor coolant pump oil collection systems.

2.3 NFPA 805 Frequently Asked Questions

In the LAR, the licensee proposed to use a number of documents commonly known as NFPA 805 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) in which each FAQ is referenced.

Table 2.3-1: NFPA 805 Frequently Asked Questions

FAQ #	FAQ Title and Summary	Reference No.	SE Section
06-0022	"Electrical Cable Flame Propagation Tests" • This FAQ provides a list of acceptable electrical cable flame propagation tests.	42	3.1.1.6, 3.1.4.2

<table>
<tr><th>FAQ #</th><th>FAQ Title and Summary</th><th>Reference No.</th><th>SE Section</th></tr>
<tr><td>07-0030</td><td>"Establishing Recovery Actions"

• This FAQ provides an acceptable process for determining the recovery actions for NFPA 805 Chapter 4 compliance. The process includes:
▪ Differentiation between recovery actions and activities in the main control room or at primary control station(s).
▪ Determination of which recovery actions are required by the NFPA 805 fire protection program.
▪ Evaluate the additional risk presented by the use of recovery actions.
▪ Evaluate the feasibility of the identified recovery actions.
▪ Evaluate the reliability of the identified recovery actions.</td><td>43</td><td>3.2.5</td></tr>
<tr><td>07-0038</td><td>"Lessons Learned on Multiple Spurious Operations (MSOs)"

• This FAQ reflects an acceptable process for the treatment of MSOs during transition to NFPA 805:
▪ Step 1 – Identify potential MSO combinations of concern.
▪ Step 2 – Expert panel assesses plant specific vulnerabilities and reviews MSOs of concern.
▪ Step 3 – Update the fire PRA and Nuclear Safety Capability Assessment to include MSOs of concern.
▪ Step 4 – Evaluate for NFPA 805 compliance.
▪ Step 5 – Document the results.</td><td>44</td><td>3.2.1.1, 3.2.4</td></tr>
<tr><td>07-0039</td><td>"Incorporation of Pilot Plant Lessons Learned – Table B-2"

• 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 54). In short, the process has the licensees:
▪ Assemble industry and plant-specific documentation;
▪ Determine which sections of the guidance are applicable;
▪ Compare the existing safe shutdown methodology to the applicable guidance; and
▪ Document any discrepancies.</td><td>45</td><td>3.2.1</td></tr>
</table>

FAQ #	FAQ Title and Summary	Reference No.	SE Section
07-0040	"Non-Power Operations (NPO) Clarifications" • This FAQ clarifies an acceptable NFPA 805 NPO program. The process includes: ▪ Selecting NPO equipment and cabling. ▪ Evaluation of NPO Higher Risk Evolutions (HRE). ▪ Analyzing NPO key safety functions (KSF). ▪ Identifying plant areas to protect or "pinch points" during NPO HREs and actions to be taken if KSFs are lost.	46	3.5.3, 3.5.3.1 thru 3.5.3.4
08-0048	"Revised Fire Ignition Frequencies" • This FAQ provides an acceptable method for using updated fire ignition frequencies in the licensee's fire PRA. The method involves the use of sensitivity studies when the updated fire ignition frequencies are used.	47	3.4.2.2, 3.4.7
08-0050	"Manual Non-Suppression Probability" • This FAQ updates the treatment of manual suppression and fire brigade response. The update includes a process to adjust the non-suppression analysis for scenario-specific fire brigade responses.	48	3.4.4
08-0052	"Transient Fires - Growth Rates and Control Room Non-Suppression" • This FAQ clarifies and updates the treatment of transient fires in terms of both manual suppression and time-dependent fire growth modeling.	49	3.4.2.2, 3.4.2.3.2
08-0054	"Demonstrating Compliance with Chapter 4 of NFPA 805" • This FAQ provides an acceptable process to demonstrate Chapter 4 compliance for transition: ▪ Step 1 – Assemble documentation ▪ Step 2 – Document Fulfillment of Nuclear Safety Performance Criteria ▪ Step 3 – Variance From Deterministic Requirements (VFDR) Identification, Characterization, and Resolution Considerations ▪ Step 4 – Performance-Based Evaluations ▪ Step 5 – Final VFDR Evaluation ▪ Step 6 – Document Required Fire Protection Systems and Features	50	3.5.1.4

FAQ #	FAQ Title and Summary	Reference No.	SE Section
09-0056	"Radioactive Release Transition" • This FAQ provides an acceptable level of detail and content for the radioactive release section of the LAR. It includes: ▪ Justification of the compartmentation, if the radioactive release review is not performed on a fire area basis. ▪ Pre-fire plan and fire brigade training review results. ▪ Results from the review of engineering controls for gaseous and liquid effluents.	51	3.6
10-0059	"NFPA 805 Monitoring Program" • This FAQ provides clarification regarding the implementation of an NFPA 805 monitoring program for transition. It includes: ▪ Monitoring program analysis units; ▪ Screening of low safety significant structures, systems, and components; ▪ Action level thresholds; and ▪ The use of existing monitoring programs.	52	3.7
12-0064	"Hot Work/Transient Fire Frequency Influence Factors" • 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.	53	3.4.2.2

2.4 Orders, License Conditions, and Technical Specifications

Paragraph 50.48(c)(3)(i) of 10 CFR states 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 Callaway's LAR (Reference 4), with regard to NRC-issued Orders pertinent to Callaway that are being revised or superseded by the NFPA 805 transition process. The LAR stated that the licensee conducted a review of its 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 Callaway are maintained. The licensee discussed the affected orders and exemptions in Attachment O of the LAR. Callaway was licensed to operate after January 1, 1979, and, as such, 10 CFR Part 50,

Appendix R, is not applicable, and exemptions from the regulation were not necessary. The licensee determined that no orders need to be superseded or revised to implement an FPP at Callaway that complies with 10 CFR 50.48(c).

The review conducted by the licensee included an assessment of docketed correspondence by performing electronic searches of the docketed correspondence files using the Callaway Licensing Research System which contains Callaway licensing documents, correspondence, and regulatory and guidance materials, including documents pertaining to the operating license, the TSs, the FPP, the FSAR and subsequent revisions, correspondence sent to the NRC, and correspondence received from the NRC. The review was performed to ensure that compliance with the physical protection requirements, security orders, and adherence to commitments applicable to Callaway are maintained. The NRC staff concludes that the licensee has met the requirement in 10 CFR 50.48(c)(3) with regard to identifying any orders that must be revised or superseded and accepts the licensee's determination that no exemptions need to be rescinded and that no orders need to be superseded or revised to implement NFPA 805 at Callaway. Section 2.5 of this SE discusses rescission of exemptions.

In addition, the licensee performed a specific review of the license amendment for license condition 2.C.15 issued June 27, 2007 (Reference 40) that incorporated the mitigation strategies required by Section B.5.b of Commission Order EA-02-026 to ensure that any changes being made in order to comply with 10 CFR 50.48(c) do not invalidate existing commitments applicable to Callaway. The NRC staff notes that the requirements in Commission Order EA-02-026 were codified in 10 CFR 50.54(hh)(2) in 2011. The licensee's review of this order and the related license amendment demonstrated that changes to the FPP during transition to NFPA 805 will not affect the mitigation measures required by Commission Order EA-02-026. The NRC staff accepts the licensee's determination in regard to Commission Order EA-02-026.

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 Callaway 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 Callaway fire protection license condition, 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 revised license condition provides 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 revised license condition also defines 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 are completed, the risk analysis is not based on the as-built, as-operated, and maintained plant.

Overall, the licensee's revised license condition allows self-approval for FPP changes that meet the requirements of NFPA 805 with regard to engineering analyses, fire risk evaluations (FREs), and plant change evaluations (PCEs). The NRC staff's evaluation of the self-approval process for FPP changes (post-transition) is contained in Section 2.6 of this SE. The license condition also identifies the implementation items and associated implementation schedules that must be accomplished at Callaway to complete transition to NFPA 805 and achieve full compliance with 10 CFR 50.48(c). These implementation items and implementation schedules are identical to those identified elsewhere in the LAR and supplements, as discussed by the NRC staff in Sections 2.7 and 2.8, and reviewed in Section 3.0, of this SE.

Section 4.0 of this SE provides the NRC staff's review of the Callaway FPP license condition.

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 Callaway TSs that are being revised or superseded during the NFPA 805 transition process. According to the LAR, the licensee conducted a review of the Callaway TSs, and TS Bases including proposed TS changes that have been submitted to the NRC for approval, 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 two changes.

The licensee determined that changes to TS 5.4.1.d and TS Bases 3.3.4 are necessary to transition to NFPA 805. TS 5.4.1.d states that written procedures shall be established, implemented, and maintained for FPP implementation. The licensee proposed deleting this TS because after transition to NFPA 805 is complete, the requirements for establishing, implementing, and maintaining FPP procedures will be contained in 10 CFR 50.48(a) and 10 CFR 50.48(c), in accordance with Section 3.2.3 of NFPA 805. TS Bases 3.3.4 includes a safety analysis and a reference section. The licensee proposed revisions to these sections to include reference to 10 CFR 50.48(c).

Based on the review of LAR Section 5.2.2 and LAR Attachment N, the NRC staff concluded that the TS changes proposed by the licensee are acceptable, because NFPA 805 Section 3.2.3 requires procedures be established for implementation of the FPP, and the licensing basis of Callaway will be 10 CFR 50.48(a) and (c) after the transition to NFPA 805 is completed.

2.4.4 Final Safety Analysis Report (FSAR)

The NRC staff reviewed the LAR and noted that Figure 4-8, "NFPA 805 Planned Post-Transition Documents and Relationships," of the LAR indicates that post-transition NFPA 805 documentation will include the revised license condition and FSAR Standard Plant (SP) Section 9.5.1. The NRC staff noted that implementation item 11-805-073 in LAR Attachment S, Table S-3, provides for revisions to add the monitoring program to the Callaway Operating Quality Assurance Manual (OQAM), Section 18.8.e, to change the fire protection quality assurance audit frequency in the OQAM from 2 to 3 years, and to relocate the audit requirements contained in the OQAM to FSAR SP Section 9.5.1.

Updates to the FSAR are required by 10 CFR 50.71(e), and the licensee states in its FSAR SP that records are and will be maintained in accordance with the requirements of sections (a) through (e) of 10 CFR 50.71.

Since the licensee's process for updating its FSAR is in accordance with 10 CFR 50.71(e), which is consistent with the guidance provided in NEI 04-02 for updating the FSAR, the NRC staff concludes that the licensee's method for updating the FSAR 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.

Since Callaway was licensed to operate after January 1, 1979, any licensing actions associated with 10 CFR Part 50, Appendix R, were not issued as exemptions to the regulation; therefore, no exemptions need to be rescinded.

2.6 Self-Approval Process for FPP 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, in part, that:

> 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 PCE 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, Section 5.3, "Plant Change Process," as well as Appendices B, I, and J, as modified by RG 1.205, Regulatory Positions 2.2.4, 3.1, 3.2, and 4.3.

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

1. defining the change
2. preliminary risk screening
3. risk evaluation
4. acceptability determination

In the LAR, the licensee stated that the PCE process begins by defining the change or altered condition to be examined and the baseline configuration. The licensee further stated that the baseline is defined as that plant condition or configuration that is consistent with the Design Basis and Licensing Basis (NFPA 805 Licensing Basis post-transition) and that 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 be performed to identify and resolve minor changes to the FPP and that the screening will be consistent with fire protection regulatory review processes in place at nuclear plants under traditional licensing bases. The licensee further stated that the screening process is modeled after the NEI 02-03, "Guidance for Performing a Regulatory Review of Proposed Changes to the Approved Fire Protection Program," Revision 0 (Reference 55), process and that it will address most administrative changes (e.g., changes to the combustible control program, organizational changes).

The licensee stated that the screening is followed by engineering evaluations that may include fire modeling (FM) and risk assessment techniques and that the results of the evaluations are 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 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 stated that the acceptance criteria require that the resultant change in core damage frequency (CDF) and LERF be consistent with the license condition and include consideration of DID and safety margin (SM), which would typically be qualitative in nature.

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

The licensee stated that PCEs are assessed for acceptability using the change in CDF (delta-CDF or ΔCDF) and change in LERF (delta-LERF or ΔLERF) criteria from the license condition and the proposed changes are assessed to ensure they are consistent with the DID philosophy and that sufficient SM were maintained.

The licensee stated that the Callaway FPP configuration is defined by the program documentation and that 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 which govern the various Callaway 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 NSCA Treatment, etc., generally require new control procedures and processes to be developed since they are new documents and databases created as a result of the transition to NFPA 805. The licensee also stated that the new procedures will be modeled after the existing processes for similar types of documents and databases. The licensee further stated that development of new control procedures and processes for new documents and databases created as a result of the transition to NFPA 805 will be completed as part of LAR implementation. See LAR Attachment S for implementation items.

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 with the first step of the review being 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 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: Complying with NFPA 805, Chapters 3 and 4.2.3 requirements.
- PB Approach: Utilizing the NFPA 805 change process developed in accordance with NEI 04-02, RG 1.205, and the Callaway 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, which requires the use of qualified individuals, procedures that require calculations be subject to independent review and verification, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when errors are discovered.

Since NFPA 805 always requires the use of a PCE, regardless of what element requires the change, the NRC staff concludes that, in accordance with the requirements of NFPA 805, if FPP impacts are determined to exist as a result of the proposed change, the issue would be resolved by utilizing the NFPA 805 change process developed in accordance with NEI 04 02, RG 1.205, and the Callaway 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.

Based on the information provided by the licensee, the NRC staff concludes that the licensee's PCE process is acceptable, because it meets the guidance in NEI 04-02, as well as RG 1.205, and addresses required attributes for using PCEs in accordance with NFPA 805. Section 2.4.4 requires that PCEs consist of an integrated assessment of risk, DID, and SM. Section 2.4.3.1 requires that the PRA 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 PRA 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.

The licensee's PCE process includes 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 SM, as discussed above.

2.6.2 Requirements for the Self Approval Process Regarding Plant Changes

Risk assessments performed to evaluate PCEs 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, as well as RG 1.205. The NRC staff concludes that the proposed PCE process at Callaway, 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 SM.

However, before achieving full compliance with 10 CFR 50.48(c) by completing the implementation items listed in Section 2.8 of this SE (i.e., during full implementation of the transition to NFPA 805), RI 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, because the risk analysis is not consistent with the as-built, as-operated, and maintained plant since the implementation items have not been completed. In addition, the licensee is required to ensure that fire protection DID and SM is 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, 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 RI, 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 the functional equivalency approach does not fall under NFPA 805, Section 1.7, "Equivalency," which states that NFPA 805 is not intended to prevent the use of systems, methods, or devices of equivalent or superior quality, strength, fire resistance, effectiveness, durability, and safety over those prescribed by NFPA 805. NFPA 805 Section 1.7 requires submission of documentation to the AHJ, which is the NRC, to demonstrate such equivalency; and states that the AHJ shall approve the system, method, or device for the intended purpose. Section 1.7 of NFPA 805 is a standard format used throughout NFPA standards that is intended to allow owner/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, NFPA 805 Section 1.7 requires approval from the AHJ because not all of these state-of-the-art features are in current use or have relevant operating experience. This demonstration of equivalency is different 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 change 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 (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:

1. "Fire Alarm and Detection Systems" (Section 3.8);
2. "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
3. "Gaseous Fire Suppression Systems" (Section 3.10); and,
4. "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. Because (1) the proposed NFPA 805 license condition includes the acceptance criteria and other attributes from the sample license condition contained in RG 1.205, and (2) 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 condition, the 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 Callaway 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 PCEs 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, says that a license condition included in a 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.

The NRC staff noted that the list of modifications and implementation items originally submitted in the LAR have been updated by the licensee with the final version of LAR Attachment S, "Plant Modifications and Items to be Completed during Implementation." The updated LAR Attachment S is provided in the licensee's letter dated December 19, 2013 (Reference 10).

2.7.1 Modifications

The NRC staff reviewed LAR Attachment S, "Plant Modifications and Items to be Completed during Implementation," which describes the Callaway plant modifications necessary to implement the NFPA 805 licensing basis, as proposed. These modifications are identified in the LAR as necessary to bring Callaway into compliance with either the deterministic or PB requirements of NFPA 805. As described below, LAR 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.

LAR Attachment S, Table S-1, provides a listing of the already completed modifications performed at Callaway as part of the NFPA 805 transition. LAR Attachment S, Table S-2, provides a detailed listing of the plant modifications that must be completed in order for Callaway to be in full compliance with NFPA 805. As discussed in the updated LAR Attachment S provided in the licensee's letter dated December 19, 2013, all modifications have been completed and moved from LAR Table S-2 to Table S-1.

The NRC staff's review confirmed that the modifications identified in LAR Attachment S, Tables S-1 and S-2 are the same as those identified in LAR Attachment B, 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 Attachment S, 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.

2.7.2 Schedule

LAR Section 5.4 provides the overall schedule for completing the NFPA 805 transition at Callaway. The licensee stated that it will complete the implementation of the new program, including procedure changes, process updates, and training of affected plant personnel to implement the NFPA 805 FPP within 8 months after NRC approval of the license amendment.

LAR Section 5.4, as revised in the licensee's letter dated December 19, 2013, also states that all of the NFPA 805 modifications have been completed.

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 amendment, 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 disposed appropriately under the licensee's corrective action program.

3.0 TECHNICAL EVALUATION

The following sections evaluate the technical aspects of the LAR (Reference 4) to transition the FPP at Callaway to one based on NFPA 805 (Reference 1) 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 21), 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. Specifically:

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

- Section 3.2 provides the results of the NRC staff 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 review of the FM methods used by the licensee to demonstrate the ability to meet the NSPC using an FM PB approach.

- Section 3.4 provides the results of the NRC staff review of the fire risk assessments used to demonstrate the ability to meet the NSPC using an FRE PB approach.

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

- Section 3.6 provides the results of the NRC staff 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 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 review of the licensee's program documentation, configuration control, and quality assurance.

In addition, Attachments A and B to this SE provide additional detailed information that was evaluated and/or disposed 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 and Design Elements

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 PB methods to demonstrate compliance with the NFPA 805, Chapter 3 requirements, 10 CFR 50.48(c)(2)(vii) specifically states that the FPP elements and minimum design requirements of NFPA 805, Chapter 3, may be subject to the PB 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 NFPA 805 and 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 3), as endorsed by the NRC in RG 1.205, Revision 1 (Reference 2), to assess the proposed Callaway 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 Callaway FPP and provided specific compliance statements for each 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 Callaway, and others are provided with multiple compliance statements to fully document compliance with the element.

The methods used by Callaway for achieving compliance with the NFPA 805 Chapter 3 fundamental FPP elements and minimum design requirements are 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" (LAR Table B-1), as "Complies."

2. The existing FPP element complies through the use of an explanation or clarification: noted in LAR Table B-1 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 LAR Table B-1 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 LAR Table B-1 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 PB method in accordance with 10 CFR 50.48(c)(2)(vii): noted in LAR Table B-1 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 LAR Table B-1 as "Complies with Required Action." These outstanding actions are identified as implementation items in Attachment S of the LAR as discussed in Section 2.8 of this SE.

Compliance approach No. 6, "Complies with Required Action," is a modification from the NEI 04-02 (Reference 3) 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 in LAR Attachment S.

The NRC staff has determined that, taken together, these methods compose an acceptable approach for documenting compliance with the NFPA 805, Chapter 3 requirements, because the licensee has followed the compliance strategies identified in the endorsed NEI 04-02 guidance document. The process defined in the endorsed guidance provides an organized structure to document each attribute in NFPA 805 Chapter 3, allowing the licensee to provide significant detail in how the program meets the requirements. In addition to the basic strategy of "Complies," which itself makes the attribute both auditable and inspectable, additional strategies have been provided allowing for amplification of information, when necessary, regarding how or why the attribute is acceptable.

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 Callaway plant configuration. The licensee further stated in LAR Section 4.1.1, "Overview of Evaluation Process," that EEEEs were used where they demonstrated an equivalent condition to the NFPA 805 Chapter 3 requirement. The licensee determined that no EEEEs 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 (Reference 56)) 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 PB 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 Table B-1 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 information provided by the licensee in the LAR and the information gained from the NFPA 805 site audit conducted January 23-27, 2012 (Reference 85) (the documents reviewed, discussions held with the licensee and the plant tours performed), the NRC staff concludes that the licensee's statements of compliance are acceptable.

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

- 3.9.3
- 3.10.2

NFPA 805 Sections 3.9.3 and 3.10.2 provide requirements for fire suppression system and gaseous suppression system alarm annunciation in the control room. The compliance statement for these attributes noted compliance with clarification; however, the clarification was not provided. The compliance basis stated that water flow alarms and system actuation alarms annunciate on panels that connect to KC008, which is located in the control room. In fire protection engineering (FPE) RAI 9 dated March 2, 2012 (Reference 11), the NRC staff requested that the licensee provide further discussion on the clarifications, including description of the alarm process and how the alarming condition is communicated to operators. In its response dated April 17, 2012 (Reference 6), the licensee stated that the correct compliance statement for Sections 3.9.3 and 3.10.2 is "complies," and no clarification was needed because the fire suppression and gaseous suppression systems annunciate in the control room. Since the suppression systems annunciate in the control room, the NRC staff concludes that the licensee's statement of compliance is 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 concludes that the licensee meets the underlying requirements for the FPP elements as clarified.

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

- 3.4.2.4
- 3.5.1(b)
- 3.5.15
- 3.6.1
- 3.6.2
- 3.10.9
- 3.11.5

NFPA 805, Section 3.4.2.4 provides the requirement for pre-fire plans to address coordination with other plant groups during fire emergencies. The compliance statement for this attribute noted compliance with required action; however, no action was identified. In FPE RAI 4 dated March 2, 2012 (Reference 11), the NRC staff requested identification of the required action. In its response dated April 17, 2012 (Reference 6), the licensee stated that the correct compliance statement for Section 3.4.2.4 is complies with clarification, and that no action is required. The licensee clarified that information on coordination with other plant groups is contained in other plant procedures which are used in conjunction with the pre-fire plans as part of the overall fire response. Based on the use of formal procedures as an alternative approach to including this information in the plant pre-fire plans, the NRC staff concludes that the licensee's statement of compliance is acceptable.

NFPA 805, Section 3.5.1 provides the requirement for a fire protection water supply. Two methods of providing a water supply are described in Sections 3.5.1(a) and 3.5.1(b); compliance with one of the two methods is required. While reviewing Approval Request 1 in LAR Attachment L, the NRC staff identified a discrepancy between two sections of the LAR causing confusion with regard to which method of compliance is used at Callaway. LAR Table B-1 originally documented that the requirements of NFPA 805 Section 3.5.1(a) were met and that the compliance statement for NFPA 805, Section 3.5.1(b) was not applicable. However, Approval Request 1 in LAR Attachment L used compliance with Section 3.5.1(b) as the basis for the request. The licensee stated that limiting the non-fire water flow to 250 gallons per minute (gpm) would prevent any impact on automatic fire suppression performance because the non-fire water flow was less than the 500 gpm hose stream allowance. In FPE RAI 12 (Reference 11) the NRC staff requested the reconciliation of the discrepancy. In its response dated April 17, 2012 (Reference 6), the licensee stated that the plant credits the criteria of Section 3.5.1(b) as the basis for the fire protection water supply, and that LAR Table B-1 Section 3.5.1 was revised to credit the criteria of Section 3.5.1(b). In the revised compliance basis for LAR Table B-1 Section 3.5.1(b), the licensee further stated that the water supply is adequate to meet the water supply requirements of the largest design demand of any water spray or sprinkler system in the power block and manual hose streams for a two-hour duration. Based on the water supply providing the required quantity of water for the two-hour duration, the NRC staff concludes that this clarification is acceptable, because the requirements of NFPA 805 Section 3.5.1 are met.

NFPA 805, Section 3.5.15 provides the requirement for hydrant and hose house spacing, and the equipment to be provided in hose houses. The licensee used the exception to the section which states that a mobile means of providing hose and associated equipment is permitted in lieu of hose houses. The exception further states that the mobile equipment shall be equivalent to the equipment supplied by three hose houses. The compliance basis stated that equipment is provided on two mobile units, but the actual amount of equipment was not specified. In FPE RAI 6 dated March 2, 2012 (Reference 11), the NRC staff requested the actual equipment equivalency for the mobile units be specified. In its response dated April 17, 2012 (Reference 6), the licensee stated that each mobile unit has equipment equivalent to three hose houses. Based on the licensee's description of the amount of equipment provided on the mobile units at Callaway being equivalent to the amount specified in the exception to the section, the NRC staff concludes that the licensee's statement of compliance is acceptable.

NFPA 805, Section 3.6.1 provides the requirement for standpipe and hose systems to be installed in accordance with NFPA 14, "Standard for the Installation of Standpipe and Hose Systems" (Reference 38). NFPA 14 specifies the system pressure for various classes of connections. During the NFPA 805 site audit, the licensee indicated that normal working pressures range from 150-160 pounds per square inch (psi), which exceeds the values identified in NFPA 14. In FPE RAI 15 dated March 2, 2012 (Reference 11), the NRC staff requested a description of the system pressures at the hose connections and a justification for any pressures that exceed the values identified in NFPA 14. In its response dated April 17, 2012 (Reference 6), the licensee stated that standpipe and hose stations at Callaway are designed and installed in accordance with the requirements in the 1976 edition of NFPA 14 for Class II service. The licensee further stated that the pressure reducing devices specified in NFPA 14 Section 4-4.2 were removed from the hose valves since use of the hose valves is restricted to the plant fire brigade who is trained in using high pressure hose. In addition to removing the hose rack pressure reducing devices, the fire brigade training program was revised to require that a minimum of two qualified personnel man the fire hose, brigade members were trained in the use of fire hose stations with pressures up to the maximum pressures found on the plant's standpipes, and caution signs were posted on standpipes that have outlet pressures greater than 100 psi. In its response to FPE RAI 15.01 dated February 19, 2013 (Reference 8), the licensee stated that the service testing pressure for fire hoses is 250 psi and adequately accounts for the higher pressures found at the site. In addition, during plant tours at the NFPA 805 site audit, the NRC staff observed that signage warning of high system pressures were posted as appropriate in the plant toured areas. Based on the restrictions placed on the use of the standpipe and hose system and the training provided to the fire brigade on the use of high pressures in fire-fighting, the NRC concludes that the licensee's statement of compliance is acceptable.

NFPA 805, Section 3.6.2 provides the requirement to ensure adequate water flow rate and nozzle pressure for all hose stations. The compliance statement for this attribute was "complies with clarification," and the compliance basis for this attribute stated that standpipe and hose station water flow rate and pressure comply with the requirements of the section. However, the clarification was not provided. In FPE RAI 7 dated March 2, 2012 (Reference 11), the NRC staff requested that the clarification used to support the compliance statement be identified. In its response dated April 17, 2012 (Reference 6), the licensee stated that compliance with this section is applicable to all hose stations except those protecting the essential service water (ESW) pump house, which are supplied from the ESW system rather than the fire protection system. The clarification was provided to identify that the hose stations protecting the ESW pump house were the subject of a clarification of previous NRC approval and discussed in Sections 3.6.1 and 3.6.2 of LAR Attachment A and in LAR Attachment T as part of "Prior Approval Clarification Request 5." The NRC staff's review of Prior Approval Clarification Request 5 is discussed in Section 3.1.1.4 of this SE. Upon review, the NRC staff concludes that the hose station installation at Callaway is consistent with the guidance in NFPA 14. Based on the information submitted by the licensee, the NRC staff concludes that this clarification is acceptable, because the intent of this NFPA 805 Chapter 3 element, to provide standpipes and hose stations in accordance with NFPA 14 and to provide adequate flow and pressure for all hose stations, is achieved.

NFPA 805, Section 3.10.9 provides the requirement for the consideration of the possibility of secondary thermal shock damage during the design of any gaseous fire suppression system. The licensee supplied clarification in the compliance basis that their gaseous fire suppression agent, Halon 1301, does not present a risk of secondary thermal shock. However adequate detail was not provided to support that secondary thermal shock was considered for the design of gaseous suppression systems at Callaway. In FPE RAI 13 dated March 2, 2012 (Reference 11), the NRC staff requested additional information to justify the conclusion that Halon 1301 does not present a risk of secondary thermal shock. In its response dated July 12, 2012 (Reference 7), the licensee stated that a full system discharge test was performed for all Halon systems as part of the initial acceptance testing. During this test, no thermal impacts were noted as a result of the system discharges. Based on no thermal impacts being observed during a full discharge test completed as part of the initial acceptance of the system at Callaway, the NRC staff concludes that the licensee's statement of compliance acceptable.

NFPA 805 Section 3.11.5 provides the requirement that electrical raceway fire barrier systems (ERFBS) required by NFPA 805 Chapter 4 shall be capable of resisting the fire effects of the hazards. ERFBS shall be tested in accordance with and shall meet the acceptance criteria of GL 86-10, Supplement 1, "Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area" (Reference 58). The licensee supplied clarification in the compliance basis that some ERFBS comply with the requirement fully (are capable of resisting the fire effects for the full duration of the deterministic requirement of either 1 hour or 3 hours) and some are degraded and were analyzed using the PB approach in accordance with NFPA 805 Section 4.2.4. In both cases, the licensee stated that the ERFBS installation meets the requirements of Section 3.11.5 which requires that ERFBS shall be capable of resisting the fire effects of the hazards and be tested in accordance with and meet the acceptance criteria of GL 86-10, Supplement 1. For those ERFBS installations that are capable of resisting the fire effects for the full duration of the deterministic requirements, the NRC staff concludes that the installation is acceptable since the installation meets the NFPA 805 deterministic requirements. For the installations that are degraded from the fully qualified configuration, the licensee has performed PB evaluations and found these to be acceptable. Based on a review of the documentation provided by the licensee, the NRC staff concludes that the degraded ERFBS installations have been evaluated and found to be acceptable using the PB approach in accordance with NFPA 805 Section 4.2.4 and are, therefore, also acceptable.

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 the licensee's statements of compliance in these instances are acceptable.

NFPA 805, Section 3.4.1(c), requires that the fire brigade leader and at least two members have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on NSPC. In LAR Attachment A, for this attribute, the licensee stated that its compliance strategy was "Complies." However, it was unclear to the staff whether the personnel assigned to the fire brigade have sufficient training and knowledge of nuclear safety systems as described in NFPA 805. The NRC staff requested additional information regarding this attribute in FPE RAI 18.01 dated August 16, 2013 (Reference 16). In its response dated September 24, 2013 (Reference 10), the licensee revised the compliance strategy for this element to "Complies with Use of EEEE." Based on the licensee's statement of continued validity for the EEEE, the statement that the evaluation demonstrates that the condition evaluated is equivalent to the NFPA 805 Chapter 3 requirement, as well as the statement on the quality and appropriateness of the evaluation, the NRC staff concludes that the licensee's statement of compliance for this attribute is acceptable.

3.1.1.4 Compliance Strategy – Complies with Previous NRC Approval

Certain NFPA 805 Chapter 3 requirements were supplanted by an alternative that was previously approved by the NRC. The approval was documented in NUREG-0830, "Safety Evaluation Report Related to the Operation of Callaway Plant, Unit No. 1," dated October 1981 and Supplements 1 through 4 dated April 1982, July 1983, May 1984, and October 1984, respectively (Reference 57).

The licensee provided a clarification to identify that the hose stations protecting the ESW pump house had received previous NRC approval, as discussed in Sections 3.6.1 and 3.6.2 of LAR Attachment A and in LAR Attachment T as part of "Prior Approval Clarification Request 5." LAR Attachment T stated that, as a result of NRC staff comments during original plant licensing, hose stations had been installed in the ESW pump house but were supplied from the ESW system rather than the fire protection system. The licensee provided justification for use of the ESW system by stating that the ESW lines are normally pressurized by the plant service water system and under emergency conditions by the ESW pumps. The licensee concluded that use of these hose stations does not impair the ability of the ESW system to perform its intended function. The NRC staff reviewed the information submitted by the licensee and concludes that the hose station installation at Callaway is consistent with the guidance in NFPA 14. Based on the information submitted by the licensee, the NRC staff concludes that this clarification is acceptable, because the intent of this NFPA 805 Chapter 3 element, to provide adequate flow and pressure for all hose stations, is achieved.

The licensee also provided a clarification regarding the ability to meet the fire brigade staffing requirements in accordance with NFPA 805 Section 3.4.1(a). Callaway obtained previous NRC staff approval for a two hour grace period for fire brigade staffing to accommodate unexpected absence provided immediate action is taken to fill the required positions. Previous approval was obtained through NRC staff approval of the Callaway TSs. In later revisions to the Callaway TSs, the requirements related to fire protection were removed and relocated to the FSAR SP. The allowance for a 2 hour grace period is now located in FSAR SP Section 16.12.1. The NRC staff reviewed the information provided by the licensee in the LAR, Clarification of Prior NRC Approvals, Request 6, and concludes that a 2 hour grace period for fire brigade staffing was previously approved and, therefore, is an acceptable alternative to compliance to the fire brigade staffing requirements in NFPA 805 Section 3.4.1(a).

The NRC staff noted that the information provided by the licensee for each previous approval was in the form of quotations from the NRC SER, with no reference to the licensee's original request. However, sufficient information was provided for the NRC staff to conclude that previous approval had been obtained. In each instance, the licensee evaluated the basis for the original NRC approval and determined that in all cases the bases remained 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. Based on the licensee's justification 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.

3.1.1.5 Compliance Strategy – Submit for NRC Approval

For certain NFPA 805 Chapter 3 requirements, the licensee requested approval to use PB methods to demonstrate compliance with fundamental FPP elements. In accordance with 10 CFR 50.48(c)(2)(vii), the licensee requested specific approvals be included in the license amendment approving the transition to NFPA 805 at Callaway. The NFPA 805 sections identified in LAR Table B-1 as complying via this method are as follows:

- 3.2.3(1), which concerns procedures that implement the FPP, including inspection, testing and maintenance procedures for fire protection systems. The licensee requested approval to use PB methods to establish inspection, testing and maintenance frequencies for fire protection systems and features required by NFPA 805. The NRC staff's review and approval of this request is documented in Section 3.1.4.1 of this SE.

- 3.3.5.1, which concerns wiring above suspended ceilings, and the requirement that this wiring be listed for plenum use, routed in armored cable, routed in metallic conduit or routed in cable trays with solid metal top and bottom covers. The licensee requested approval to use PB methods to demonstrate an equivalent level of fire protection for the existence of wiring which does not meet the criteria of NFPA 805 Section 3.3.5.1. The NRC staff's review and approval of this request is documented in Section 3.1.4.2 of this SE.

- 3.5.16, which concerns the dedication of fire protection water supply for fire protection use only. The licensee requested approval for the use of fire protection system water for plant evolutions other than fire protection. The NRC staff's review and approval of this request is documented in Section 3.1.4.3 of this SE.

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

3.1.1.6 Compliance Strategy – Complies with Required Action

For certain NFPA 805 Chapter 3 requirements 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:

- 3.2.2.3 Management Policy Direction and Responsibility – Fire Protection Interfaces

  Procedure APA-ZZ-00700, “Fire Protection Program,” will be revised to clearly define the fire protection interfaces with other organizations using the guidelines of Appendix A of NFPA 805. This action is identified as implementation item 07-805-001.

- 3.2.2.4 Management Policy Direction and Responsibility – Document Identifying the AHJ

  The AHJ will be identified in procedure APA-ZZ-00700 using the guidelines of Appendix A of NFPA 805. This action is identified as implementation item 07-805-002.

- 3.2.3(1) Procedures – Inspection Testing and Maintenance

  Procedures APA-ZZ-00700 and APA-ZZ-00703, "Fire Protection Operability Criteria and Surveillance Requirements," will be revised to include inspection, testing, and maintenance requirements for all fire protection systems and features credited by the FPP. This action is identified as implementation item 11-805-048.

  PB surveillance frequencies will be established as described in EPRI TR-1006756, and evaluated in Callaway Plant Calculation KC-162, "Performance Based Fire Protection Surveillance Frequency Program." This action is identified as implementation item 11-805-069.

- 3.3.1.2(1) Control of Combustible Materials – Wood

  Section 4.1.5.b of procedure APA-ZZ-00741, “Control of Combustible Materials,” will be revised to address that cribbing timbers 6 inches (in.) by 6 in. or larger are not required to be fire-retardant treated. This action is identified as implementation item 11-805-049.

- 3.3.1.2(2) Control of Combustible Materials – Plastic Sheeting

  Procedure APA-ZZ-00741 will be revised to include a requirement for plastic sheeting used in the power block to have passed NFPA 701, “Standard Methods of Fire Tests for Flame Propagation of Textiles and Films” (Reference 59). This action is identified as implementation item 07-805-004.

- 3.3.1.2(3) Control of Combustible Materials – Waste, Debris, Scrap

  Sections 4.1.5.c and 4.1.5.e of procedure APA-ZZ-00741 will be revised to include the removal of all waste, debris, scrap and combustible packing materials from all areas, not only safety-related buildings and adjacent areas. This action is identified as implementation item 07-805-005.

- 3.3.1.2(6) Control of Combustible Materials – Flammable Gases

  Procedures will be revised to ensure that the hydrogen supply system is inspected annually and maintained. This action is identified as implementation item 07-050A-001.

  Dry vegetation and combustible material within 15 feet of the hydrogen supply area will be removed. Additionally, procedures will be revised to ensure that the area within 15 feet of the hydrogen supply area is kept free of dry vegetation and combustible materials. This action is identified as implementation item 07-050A-002.

- 3.3.1.3.4 Portable Electrical Heaters

  Procedure APA-ZZ-00742, "Control of Ignition Sources," will be revised to include requirements for not allowing portable electric or fuel-fired heaters in plant areas containing equipment important to nuclear safety or where there is potential for radiological releases resulting from a fire. This action is identified as implementation item 07-805-006.

- 3.3.5.1 Wiring above Suspended Ceilings

  In response to FPE RAI 17 dated February 19, 2013 (Reference 8), the licensee stated that drawing E-2R8900 and procedure EDP-ZZ-04044, "Fire Protection Reviews," will be revised to require that, where wiring must be installed above a suspended ceiling, it shall comply with NFPA 805 Section 3.3.5.1. This action is identified as implementation item 11-805-050.

- 3.3.7.1 Storage of Flammable Gas

  Procedures will be revised to ensure that the hydrogen supply system is inspected annually and maintained by the licensee. This action is identified as implementation item 07-050A-001.

  Dry vegetation and combustible material within 15 feet of the hydrogen supply area will be removed. Additionally, procedures will be revised to ensure that the area within 15 feet of the hydrogen supply area is kept free of dry vegetation and combustible materials. This action is identified as implementation item 07-050A-002.

In response to FPE RAI 1 dated April 17, 2012 (Reference 6), the licensee stated that the compliance statement for Section 3.3.7.1 in NFPA 805 is "complies with required action."

- 3.3.10 Hot Pipes and Surfaces

  Procedures APA-ZZ-00741 and MDP-ZZ-LM001, "Fluid Leak Management Program," will be revised to include a requirement for the prompt cleanup of combustible liquids discovered on insulation, including high flashpoint lubricating oils (instead of only performing an assessment of the potential for fire and the recording of appropriate recommendation in APA-ZZ-00741), and to keep such fluids from coming in contact with hot pipes and surfaces, including insulated pipes and surfaces. This action is identified as implementation item 07-805-009.

- 3.3.11 Electrical Equipment

  Procedure APA-ZZ-00741 will be revised to include requirements for maintaining adequate clearance, free of combustible material, around energized electrical equipment. This action is identified as implementation item 07-805-017.

- 3.4.1(a)(1) NFPA 600-Standard on Industrial Fire Brigade

  A safety and health policy will be documented for the Callaway Fire Brigade. The policy will satisfy the requirements of NFPA 600, "Standard on Industrial Fire Brigades" (Reference 60), Sections 2-1.4 and 2-2.4. This action is identified as implementation item 07-600-001.

  Fire brigade policy documents and procedures will be updated to include a requirement for a standard system to identify and account for each industrial fire brigade member present at the scene of the emergency, in accordance with NFPA 600, Section 2-2.1.4. The requirement will also meet NFPA 600, Section 2-4.5, and will specify that industrial fire brigade members be issued identification for the following purposes:

  1. Assistance in reaching the incident in an emergency
  2. Identification by security personnel
  3. Establishing authority

  This action is identified as implementation item 07-600-002.

A risk management policy will be written for emergency response. The risk management policy shall be routinely reviewed with industrial fire brigade members and shall be based on the following recognized principles:

1. Some risk to the safety of industrial fire brigade members is acceptable where saving human lives is possible.

2. Minimal risk to the safety of the industrial fire brigade members, and only in a calculated manner, is acceptable where saving endangered property is possible.

3. No risk to the safety of industrial fire brigade members is acceptable where saving lives or property is not possible.

This action is identified as implementation item 07-600-003.

The Callaway Fire Brigade training program will be updated to include a periodic review of NFPA 600. This action is identified as implementation item 07-600-004.

In response to FPE RAI 2 dated April 17, 2012 (Reference 6), the licensee stated that the compliance statement for Section 3.4.1(a)(1) in NFPA 805 is "complies with required action."

- 3.4.1(b) Industrial Fire Brigade

Section 4.1.3(c) of procedure APA-ZZ-00743, "Fire Team Organization and Duties," will be revised to include the requirement that industrial fire brigade members shall have no other assigned normal plant duties that would prevent immediate response to a fire or other emergency as required. This action is identified as implementation item 11-805-051.

- 3.4.2 Pre-Fire Plans

The Fire Pre-Plan Manual will be revised as follows:

— The fire pre-plan attachments will be revised where the radiation release criteria are applicable for gaseous and liquid effluent as described in Table E-1/E-2 to include effluent controls and monitoring.

— New Pre-Fire Plans will be added for C-36 and C-37.

— Two new Attachments will be added, for Temporary Structures Inside the PA and for Temporary Structures Outside the PA, and existing Fire Attack Guidelines will be combined into each attachment.

This action is identified as implementation item 11-805-076.

In response to FPE RAI 3 dated April 17, 2012 (Reference 6), the licensee stated that the compliance statement for Section 3.4.2 in NFPA 805 is "complies with required action."

- 3.4.2.3 Pre-Fire Plans

In response to FPE RAI 4 dated April 17, 2012 (Reference 6), the licensee stated that the compliance statement for Section 3.4.2.3 in NFPA 805 is "complies with required action." A statement will be added to procedure APA-ZZ-00700 to require that controlled copies of the pre-fire plans be maintained in the Control Room and made available to the fire brigade. This action is identified as implementation item 07-805-047.

- 3.4.3(a)(1) Fire Brigade Training

The Callaway Fire Brigade training program will be updated to include a periodic review of NFPA 600. This action is identified as implementation item 07-600-004.

- 3.4.3(b) Plant Personnel Responding with the Fire Brigade

Procedure APA-ZZ-00700 will be revised to identify that plant personnel who respond with the industrial fire brigade are trained as to their responsibilities, potential hazards to be encountered, and interfacing with the industrial fire brigade. This action is identified as implementation item 11-805-052.

- 3.4.3(c)(2) Brigade Drills

Procedure FPP-ZZ-00009, "Fire Protection Training Program," will be revised to include an assessment of the proper use of pre-fire plans and coordination with other groups during fire brigade drills, using the guidelines of Appendix A of NFPA 805. This action is identified as implementation item 07-805-013.

- 3.4.3(c)(3) Brigade Drills

Procedure FPP-ZZ-00009 will be updated to provide requirements for drills to be conducted in various plant areas, especially in those areas identified to be essential to plant operation and to contain significant fire hazards, as required by NFPA 805. This action is identified as implementation item 07-805-014.

- 3.4.4 Fire-Fighting Equipment

In response to FPE RAI 2 dated April 17, 2012 (Reference 6), the licensee stated that a requirement that specifies that fire brigade protective clothing and respiratory protective equipment shall conform to the applicable NFPA standard will be documented in procedure APA-ZZ-00700. This action is identified as implementation item 07-805-015.

In response to FPE RAI 5 dated April 17, 2012 (Reference 6), the licensee stated that the compliance statement for Section 3.4.4 in NFPA 805 is "complies with required action."

- 3.9.1(1) NFPA 13 - Sprinkler Systems

  The missing ceiling tiles in the suspended ceiling in fire compartments C-5 and C-6 will be replaced in order to ensure proper operation of sprinkler system SKC34, which is credited in the FPRA, in accordance with NFPA 13, "Standard for the Installation of Sprinkler Systems," 1976 Edition (Reference 37). Configuration control on the ceiling tiles will be ensured. This action is identified as implementation item 11-805-091.

  During the NFPA 805 site audit, the NRC staff observed that quick response sprinkler heads were installed in multiple cable chases, replacing the original sprinkler nozzles. Due to the piping configuration, the quick response sprinkler heads were installed at an angle relative to the ceiling, as opposed to being parallel to it. In FPE RAI 14 dated March 2, 2012 (Reference 11), the NRC asked the licensee to provide the basis and justification for compliance with the appropriate NFPA standard. In its response dated April 17, 2012 (Reference 6), the licensee stated that the condition had been entered in the site corrective action program. Additionally, a plant modification to modify the sprinkler heads in the affected areas to a configuration in accordance with the requirements of the 1976 edition of NFPA 13, was added as implementation item 11-805-094. In its letter dated September 24, 2013 (Reference 10), the licensee stated that the modification had been completed and the implementation item was removed.

- 3.11.3(1) NFPA 80 Fire Doors and Windows

  The scope of Procedure SDP-KC-00002, "Fire Door Position Verification," will be revised to include all doors credited to meet the requirements of NFPA 805. This action is identified as implementation item 11-080-007.

  The scope of Procedure OSP-KC-00015, "Fire Door Inspections," will be revised to include all doors credited to meet the requirements of NFPA 805. This action is identified as implementation item 11-080-008.

Based on the information provided by the licensee in the LAR and the implementation items that will be completed prior to program implementation, the NRC staff concludes that the licensee's statements of compliance are acceptable.

3.1.1.7 Compliance Strategy – 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 that holistic compliance with the fundamental FPP element is assured.

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 Callaway, 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 Callaway because of the following:
  - The licensee states that Callaway does not have systems of this type installed (e.g., Section 3.6.5, which applies to seismic hose station cross-connected to non-fire protection systems and Section 3.9.1(3), which applies to water mist fire protection systems, Section 3.9.1(4), which applies to foam water sprinkler and foam-water spray systems, and Section 3.10.1(3), which applies to clean agent fire extinguishing systems)
  - The type of system, while installed at Callaway is not required under the RI/PB FPP (e.g., Section 3.9.1(2), which applies to water spray systems; Sections 3.10.1(1), 3.10.6, 3.10.7, and 3.10.8, which apply to carbon dioxide extinguishing systems).
  - The requirements are structured with an applicability statement and that statement does not apply to Callaway (e.g., Section 3.5.1(b), which applies to fire protection water supply; Section 3.10.4, which applies to areas required to be protected by both primary and backup gaseous fire suppression systems).

3.1.1.9 Compliance with Chapter 3 Requirements Conclusion

As discussed above, the NRC staff evaluated the results of the licensee's assessment of the proposed Callaway 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 Callaway's state of compliance with the NFPA 805 requirements, which adequately demonstrated compliance in that the licensee was able to substantiate that it complied:

  — With the requirement directly.

  — With the requirement (or element) with clarification.

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

  — Through the use of an EEEE that the licensee determined did not need NRC approval to support compliance with NFPA 805.

  — 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 Callaway structures identified in LAR Table I-1 "Definition of Power Block" as comprising the "power block." The plant structures listed are established as part of the power block for the purpose of denoting the structures and equipment included in the Callaway RI/PB FPP that have additional requirements in accordance with 10 CFR 50.48(c) and NFPA 805. LAR Section 4.1.3 states that power block includes structures contain equipment required for nuclear plant operations.

In response to FPE RAI 16 dated February 19, 2013 (Reference 8), the licensee stated that all site structures in the owner controlled area were evaluated for meeting the definition of power block structure. This evaluation included consideration of all the example structures listed in NEI 04-02. These structures include the containment, auxiliary building, service building, control building, fuel building, radwaste (radioactive waste) building, water treatment building, turbine building and various intake structures. Callaway does not have a structure, fire area, or fire zone designated as a hot machine shop. The licensee stated that the radwaste building, ESW pump house and ultimate heat sink (UHS) cooling tower are stand-alone structures within the yard area (fire area YD-1) that are included in the definition of power block. The licensee also stated that stand-alone components within the yard area, such as above-ground tanks, transformers, and underground fuel storage tanks, are not considered to meet the definition of a power block structure.

The NRC staff reviewed the information discussed above and concludes that the licensee has appropriately evaluated the structures and equipment at Callaway, and 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

Callaway does not use either the Hemyc™ or MT™ electrical raceway fire barrier systems (ERFBS). Therefore, the generic issue discussed in GL 2006-03 (Reference 33) related to the use of these ERFBS is not applicable to Callaway. GL 2006-03 requested that licensees evaluate their facilities to confirm compliance with existing applicable regulatory requirements in light of the results of NRC testing that determined that both Hemyc™ and MT™ fire barriers failed to provide the protective function intended for compliance with existing regulations, for the configurations tested using the NRC's thermal acceptance criteria.

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 PB methods permitted elsewhere in the standard as a means of demonstrating compliance with the prescriptive NFPA 805, Chapter 3, Fundamental FPP Elements and Minimum Design Requirements. Paragraph 50.48(c)(2)(vii) of 10 CFR requires that an acceptable PB 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 PB methods to demonstrate an equivalent level of fire protection for the requirements of the NFPA 805 Chapter 3 elements identified in Section 3.1.1.5 of this SE. The NRC staff evaluation of these proposed methods is provided below.

3.1.4.1 NFPA 805, Section 3.2.3(1) – Inspection, Testing, and Maintenance Procedures

The licensee requested NRC staff review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805 Section 3.2.3(1) regarding procedures that implement the FPP, including inspection, testing and maintenance procedures for fire protection systems. The licensee requested approval to use PB methods to establish inspection, testing and maintenance frequencies for fire protection systems and features required by NFPA 805. The frequencies will be established in accordance with EPRI Technical Report TR-1006756, "Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features" (Reference 61). EPRI TR-1006756 provides guidance for licensees to follow in order to optimize their fire protection surveillance and testing practices and frequencies for fire protection SSCs based upon performance.

The licensee requested approval on the following basis:

- NFPA 805 Section 2.6 requires that a monitoring program be established to ensure availability and reliability of the fire protection systems and features credited by the FPP. Performance monitoring will be performed in conjunction with the monitoring program required by NFPA 805 Section 2.6 and it will ensure site-specific operating experience is considered in the monitoring process.

- This scope and frequency of the inspection, testing, and maintenance activities for fire protection systems and features required in the FPP have been established based on the previously approved TS, licensee controlled documents, and appropriate NFPA codes. This request does not involve the use of EPRI TR-1006756 to establish the scope of those activities, as that is determined by the required system review identified in Table 4-3.

- Reliability and frequency goals will be established to ensure the assumptions in the NFPA 805 engineering analysis remain valid.

The licensee stated that use of PB test frequencies established in accordance with the methods in EPRI TR-1006756, combined with the NFPA 805 Section 2.6, "Monitoring Program," will ensure that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analysis. Therefore, there is no adverse impact to the NSPC. In addition, the licensee stated that use of PB test frequencies in conjunction with the monitoring program will ensure the availability and reliability of the fire protection systems and features are maintained to the levels credited to meet the radioactive release performance criteria. Therefore, there is no adverse impact to radioactive release performance criteria. The licensee further stated that use of EPRI TR-1006756 does not invalidate the inherent SM contained in the codes used for design and maintenance of fire protection systems and features; and the availability and reliability of fire protection systems and features credited for DID will be maintained to the levels assumed in the NFPA 805 engineering analysis. Therefore, the SM inherent and credited in the analysis and DID have been preserved.

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 PB method is an acceptable alternative to the corresponding NFPA 805 Section 3.2.3(1) requirement, because it satisfies the performance goals, objectives, and criteria specified in NFPA 805 related to nuclear safety and radiological release, maintains sufficient SM, and maintains adequate fire protection DID.

3.1.4.2 NFPA 805, 3.3.5.1 – Electrical Wiring Above Suspended Ceilings

The licensee requested NRC staff review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805 Section 3.3.5.1 regarding wiring above suspended ceilings. Specifically, the licensee has requested approval of a PB method to justify the use of limited quantities of wiring/cabling which do not meet the criteria of NFPA 805 Section 3.3.5.1. The licensee stated that wiring exists above suspended ceilings in the control room and associated areas, and in fire areas C-5 and C-6, for which the fire protection was previously approved and therefore not included in the scope of this approval request. All other areas that contain wiring above a suspended ceiling are included in the scope of this request.

As described in the request, wiring exists above suspended ceilings in the TB-1 fire area and RW-1 fire area. NFPA 805 Section 3.3.5.1 requires the wiring to be either plenum rated or routed in metallic conduit, armored cable, or covered tray. The licensee stated that each area has a limited amount of cabling that does not meet NFPA 805 Section 3.3.5.1 requirements, and that nearly all the exposed cables are communication cables associated with computers, telephones, televisions, or projectors located in the fire area. The licensee requests approval on the following basis:

- Only a limited amount of the cable installed above the suspended ceiling is not rated for plenum use or routed in conduit
- The cable is low voltage, less than 480 Volts alternating current (VAC), and therefore less susceptible to self-ignition and electrical shorts that could result in a fire in the enclosed space
- There are no additional ignition sources in the areas above the suspended ceilings
- For the cables that do not meet the NFPA 805 Section 3.3.5.1 criteria, the majority meet one of the cable qualifications listed within FAQ 06-0022, Revision 3 (Reference 42).
- Plant procedures will be revised to ensure future exposed cables installed above the suspended ceilings meet one of the cable qualifications discussed in FAQ 06-0022, Revision 3.

The licensee stated that the presence of non-rated plenum cables above the suspended ceilings in the identified fire areas does not adversely affect the nuclear safety capability. The quantities of non-rated plenum cable which do not meet NFPA 805 Section 3.3.5.1 required metal conduit, armored cable, or enclosed metal cable trays, are limited. In addition, there are no additional ignition sources above the suspended ceilings. Therefore, there is no adverse impact on the NSPC due to the non-rated plenum cabling in these areas. The licensee also stated that the location of non-rated plenum wiring above suspended ceilings has no impact on the radiological release performance criteria. Of the areas applicable to this request, only the following areas are located in a radiological controlled area: radwaste control room, radwaste lab, hot lab, counting room, and vestibule number 3. The radiological review was performed based on the potential location of radiological concerns and is not dependent on the type of wiring or locations of suspended ceilings.

The licensee further stated that, the quantity of non-rated plenum cables above the identified suspended ceiling locations is not significant, and the SM inherent in the analysis for the fire event is preserved. Fire protection DID will be maintained, because the non-plenum rated cable routed above the suspended ceilings does not impact fire protection DID. Finally, the licensee stated that the cabling does not compromise automatic or manual fire suppression functions, fire suppression for systems and structures, or the NSCA.

However, the licensee did not provide sufficient information for the NRC staff to complete its review. In FPE RAI 17 dated December 11, 2012 (Reference 14), the NRC staff requested that the licensee clarify its request. Specifically, the staff requested that the licensee clarify whether any of the cables that do not meet NFPA 805 Section 3.3.5.1 requirements are in the vicinity of nuclear safety capability systems or equipment, describe the separation between the control, power and lighting cables and the communication cables, clarify whether future installations will meet NFPA 805 Section 3.3.5.1 criteria, clarify how the radiological release goals will be met, and clarify whether any of the assumptions and limitations of the analytical methods used in the FPP are affected. In response to FPE RAI 17 dated February 19, 2013 (Reference 8), the licensee provided the following additional bases for the request:

- The affected fire zones are located in non-safety related power-block structures.

- The affected fire zones are high-traffic locations where quick manual detection of fires is probable.

- Plant design and installation requirements for electrical cable require cables to be separated by voltage level and in compliance with Institute of Electrical and Electronics Engineers (IEEE) Standard IEEE-384 (Reference 62).

- Manual suppression, by means of standpipes, is available in all the affected fire zones.

- Plant procedures will be revised to ensure future exposed cables installed above suspended ceilings meet NFPA 805, Section 3.3.5.1.

The licensee also stated that the PB method does not change the assumptions and limitations of the analytical methods used in the development of the NSCA or the development of the radiological release goals, objectives and performance criteria.

Based on its review of the LAR, as supplemented, and in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method is an acceptable alternative to the corresponding NFPA 805 Section 3.3.5.1 requirement for the TB-1 and RW-1 fire areas, because it satisfies the performance goals, objectives, and criteria specified in NFPA 805 related to nuclear safety and radiological release, maintains sufficient SM, and maintains adequate fire protection DID.

3.1.4.3 NFPA 805, 3.5.16 – Dedicated Use of Fire Protection Water Supply

The licensee requested NRC staff review and approval of a PB 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 requested approval of a PB method to justify the use of the fire protection water supply for non-fire protection plant evolutions.

As described in the request, the Shift Manager or Control Room Supervisor (CRS) may approve the use of fire protection system water for plant evolutions under the following conditions.

- Shift Manager/CRS approval is obtained and documented.
- Controls and communications are in place to ensure the non-fire protection system water demand can be secured immediately if a fire occurs.
- The non-fire protection system water demand must be less than 250 gpm.

The licensee requested approval on the following basis:

- The 250 gpm limitation is less than the hose stream allowance postulated in determining fire suppression water storage capacity requirements (a minimum of 500 gpm); therefore, assuming that non-fire water use is terminated upon notification of a fire, there is no adverse impact on the flow and pressure available to any automatic water-based suppression systems.
- Personnel utilizing the fire protection water are in contact with the Control Room, ensuring the ability to secure the non-fire protection system water demand should a fire occur. Therefore, flow will be available for the manual fire suppression demands when needed.

In FPE RAI 11 dated March 2, 2012 (Reference 11), the NRC staff requested the licensee to describe the administrative or operating procedures used to ensure the minimum required fire protection water supply remains available. In its response dated April 17, 2012 (Reference 6), the licensee stated that the following conditions will also need to be met for plant evolutions that use fire water:

- A fire protection impairment [record] is generated to document the intended non-fire usage, approvals, and any administrative controls put in place for the impairment.
- Both fire water storage tanks are functional and have sufficient margin to remain functional during usage.
- Fire water storage tank water level is monitored to ensure the level remains above 260,000 gallons, which is the procedurally required limit.

The licensee further stated that the fire protection procedure that contains operability criteria and surveillance requirements requires that the two fire water tanks maintain a volume of 260,000 gallons, a water level of 31 feet, to remain functional. The tanks are provided with local level indication, normally maintained at a water level of 34 feet and the tank levels are verified by operations personnel on daily rounds.

The licensee stated that the flow limitation ensures that there is no impact on the ability of the automatic suppression systems to perform their functions. The ability to isolate non-fire protection flows ensures there is no impact on manual fire suppression efforts. Therefore, there is no impact on the NSPC. The licensee also stated that the use of fire protection water for plant evolutions other than fire protection has no impact on the radiological release performance criteria because these criteria are satisfied based on the determination of limiting radioactive release, which is not affected by impacts on the fire protection system due to use of fire protection water for non-fire protection evolutions.

The licensee further stated that the use of the fire water system for non-fire protection uses does not impact fire protection DID. The fire pumps have the excess capacity to supply the demands of the fire protection system in addition to the non-fire protection uses bounded by the conditions identified above. This does not result in compromising automatic or manual fire suppression functions, fire suppression for systems and structures, or the NSCA. The licensee concluded that since both the automatic and manual fire suppression functions are maintained, DID is maintained.

Finally, the licensee stated that the methods, input parameters, and acceptance criteria used in the analysis were reviewed against those used for NFPA 805 Chapter 3 acceptance, and were not altered. Therefore, the SM inherent in the analysis has been preserved.

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 PB method is an acceptable alternative to the corresponding NFPA 805 Section 3.5.16 requirement, because it satisfies the performance goals, objectives, and criteria specified in NFPA 805 related to nuclear safety and radiological release, maintains sufficient SM, and maintains adequate fire protection DID.

3.1.5 Other Requested Approvals

In LAR Attachment X, Approval Request 1, the licensee requested NRC staff approval to remove the requirement currently listed in FSAR Table 9.5.1-2 to enter TS 3.0.3 for an inoperable fire suppression water system coupled with the inability to provide a backup fire suppression water system within 24 hours. The licensee stated that the basis for the request is that

> [A]pplication of [TS] 3.0.3 for plant configurations that do not meet the specific criteria of 10 CFR 50.36(c)(2) for inclusion into the plant TS is inappropriate. [TS] 3.0.3 is intended to be applied when a [TS] LCO is not met and the associated [TS] required actions are not met, an associated [TS] required action is not provided, or if directed by associated [TS] required actions. [TS] LCO 3.0.3 was

> not meant to be applied to "non-technical specification" plant configurations. The existing requirement to enter [TS] LCO 3.0.3 would not be consistent with NFPA 805 which indicates that compensatory actions should be appropriate with the level of risk created by the unavailable equipment.

Further, the licensee stated that NFPA 805 Section 3.2.3 (2) requires that licensees establish procedures to accomplish: "Compensatory actions implemented when fire protection systems and other systems credited by the fire protection program and this standard cannot perform their intended function and limits on duration."

The NRC staff reviewed this request and noted that the licensee intended to use an existing procedure to establish the required compensatory actions and impairment durations following transition to NFPA 805. The NRC staff also noted that there was no implementation item in the LAR to address the need to revise the referenced Callaway plant procedure. In its response to FPE RAI 20 dated December 19, 2013 (Reference 88), the licensee created implementation item 13-805-008 to revise this procedure to establish the required compensatory actions and impairment durations. Based on the requirement in NFPA 805 that the licensee establish procedures to implement compensatory actions and limit durations, and the licensee's implementation item to revise their existing procedure to address the NFPA 805 requirement, the NRC staff concludes that the previous requirement to enter TS 3.0.3 for an inoperable fire suppression water system coupled with the inability to provide a backup fire suppression water system within 24 hours is no longer required and may be deleted from the Callaway FSAR.

3.2 <u>Nuclear Safety Capability Assessment (NSCA) Methods</u>

NFPA 805 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.

NFPA 805, Section 2.4, "Engineering Analyses," states, in part, 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... 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," states that:

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

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

> 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," states that:

> 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 [pressurized-water reactor (PWR)] and shall be capable of maintaining or rapidly restoring reactor water level above top of active fuel for a [boiling-water reactor (BWR)] 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 NSCA 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 steps listed above. SE Section 3.5 addresses the assessment of the fourth step.

RG 1.205, Revision 1, endorses NEI 04-02, Revision 2, and Chapter 3 of NEI 00-01, Revision 2, "Guidance for Post-Fire Safe Shutdown Circuit Analysis" (Reference 34), and promulgates the method outlined in NEI 04-02 for conducting an NSCA. This NRC-endorsed guidance (i.e., NEI 04-02 Table B-2, "NFPA 805 Chapter 2 – Nuclear Safety Transition – Methodology Review," and NEI 00-01, Chapter 3) has been determined to address the related requirements of NFPA 805, Section 2.4.2. The NRC staff reviewed LAR Section 4.2.1, "Nuclear Safety Capability Assessment Methodology," and LAR Attachment B, "NEI 04-02 Table B-2 – Nuclear Safety Capability Assessment – Methodology Review" (LAR Table B-2), against these guidelines.

The licensee developed the Callaway NFPA 805 LAR based on the guidance provided in the three guidance documents cited above. Based on the information provided in the licensee's submittal, as supplemented, the licensee used a systematic process to evaluate the Callaway post-fire safe shutdown analysis (SSA) 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. The method used to perform the NSCA with respect to selection of systems and equipment, selection of cables, and identification of the location of equipment and cables is documented in a site-specific calculation which the NRC staff reviewed during the NFPA 805 site audit. The NRC staff reviewed LAR Table B-2, and concluded that the documented applicability, alignment statement, and alignment basis for each of the applicable NEI 00-01 Section 3 guidance elements was acceptable. Based on the review of the processes described in the LAR and the use of the accepted analysis method as prescribed in the NEI 00-01 guidance document, the NRC staff concludes that Callaway met the NRC-endorsed guidance directly, met the intent of the endorsed guidance with adequate justification as discussed below, or met the endorsed guidance based on prior NRC approval.

The licensee used the guidance in FAQ 07-0039 (Reference 45) for documenting the comparison of the post-fire SSA against the NFPA 805 requirements. This method first maps the existing post-fire SSA to the NEI 00-01, Revision 1 (Reference 54), 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 Callaway post-fire SSA 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 LAR Table B-2 in accordance with NEI 04-02.

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

1. The post-fire SSA directly aligns with the attribute: noted in LAR Table B-2 as "Aligns."

2. The post-fire SSA aligns with the intent of the attribute: noted in LAR Table B-2 as "Aligns with Intent." For each instance where this category was used, the NRC staff has included a discussion below further explaining the intent of the associated guidance and how the licensee's process meets this intent.

3. The post-fire SSA 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 LAR Table B-2 as "Not in Alignment, but Prior NRC Approval."

Finally, some attributes may not be applicable to the SSA (for example, the attribute may be applicable only to BWRs or PWRs). These are noted in LAR Table B-2 as "N/A."

3.2.1.1 Attribute Alignment – Aligns

RG 1.205 states that Chapter 3 of NEI 00-01, Revision 2, when used in conjunction with NFPA 805 and the RG, provides one acceptable approach to circuit analysis for a plant implementing a FPP under 10 CFR 50.48(c). For certain NEI 00-01 Chapter 3 attributes, the licensee determined that the post-fire SSA aligns directly with the attribute. In each of these instances, based on the information provided by the licensee in the LAR and the documents reviewed by the NRC staff and discussions with the licensee during the NFPA 805 site audit the NRC staff concludes that the licensee's statements of alignment are acceptable since the analyses are consistent with regulatory guidance for selecting the systems and equipment and their interrelationships necessary to achieve the NSPC, selection of the cables necessary to achieve the NSPC, and the identification of the location of nuclear safety equipment and cables.

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

- 3.0
- 3.3.2[C]
- 3.5.2.1
- 3.5.2.5

Attribute 3.0 – Deterministic Methodology: A deterministic methodology was used to assess the licensee's conformance with the NSPC from Section 1.5.1 of NFPA 805. The Callaway NFPA 805 NSCA deterministic methodology has been reviewed in detail against the guidance, criteria, and assumptions contained within NEI 00-01, Chapter 3. However, the licensee developed its post-fire safe shutdown circuit analysis methodology using the guidance provided in Chapter 3 of NEI 00-01, Revision 1, not NEI 00-01, Revision 2.

In SSA RAI 01 dated March 2, 2012 (Reference 11), the NRC staff requested that Callaway provide a gap analysis comparing its compliance with NEI 00-01, Revision 1, versus Revision 2 to ensure the additional technical issues identified in Revision 2 are appropriately addressed in the NSCA. In its response dated April 17, 2012 (Reference 6), the licensee clarified that Callaway has performed a gap analysis between NEI 00-01, Revision 1, and NEI 00-01, Revision 2. The review concluded that there were no significant differences, and Callaway met the guidelines of NEI 00-01, Revision 2, where applicable. The two notable differences identified by the licensee are the evaluation of multiple spurious operations (MSOs) and the consideration of fire impacts on stem lubrication for rising stem valves. The licensee stated that the evaluation of multiple spurious operations is consistent with the process outlined in FAQ 07-0038, Revision 3, "Lessons Learned on Multiple Spurious Operations" (Reference 44), which is an appropriate process for use in an RI/PB FPP in accordance with NFPA 805 and 10 CFR 50.48(c). The consideration of the effect of fire on stem lubrication where post-fire hand-wheel operation of a valve is credited in the area affected by the fire has been addressed in the licensee's analysis. The licensee stated that the full gap analysis has been incorporated into the Callaway NSCA. Therefore, Callaway concluded that the NSCA was performed consistent with (i.e., aligns with) the deterministic methodology guidance, criteria, and assumptions from Chapter 3 of NEI 00-01, Revision 2. Since the licensee's gap assessment concluded that the NSCA aligns with the latest endorsed guidance, the NRC staff concludes that the licensee's statement of alignment to NEI 00-01 guidance is acceptable. Attribute 3.3.2 [C] - Associated Circuit Cables - Common Enclosure Cables: Although the licensee identified this attribute as directly aligns, the licensee identified in LAR Attachment X, "Other Approval Requests," that fire damage to a cable could propagate to other safe shutdown cables because the circuit is not properly protected by an isolation device (breaker/fuse), such that the fire-induced fault could result in a secondary fire for one or more main generator current transformers (CTs).

Attribute 3.5.2.1 – Circuit Failure Due to an Open Circuit: Although the licensee identified this attribute as directly aligns, the licensee identified in LAR Attachment X that fire damage could cause an open circuit failure on a high voltage (e.g., 4.16 kV) ammeter current transformer circuit such that the fire-induced fault could result in a secondary fire for one or more main generator CTs.

Attribute 3.5.2.5 – Circuit Failure Due to Common Enclosure Concerns: Although the licensee identified this attribute as directly aligns, the licensee identified in LAR Attachment X that fire damage to a circuit either whose isolation device fails to isolate the cable fault or protect the faulted cable from reaching its ignition temperature could occur such that the fire-induced fault could result in a secondary fire for one or more main generator CTs.

For attributes 3.3.2 [C], 3.5.2.1, and 3.5.2.5, the licensee requested approval of a deviation from the requirements of NFPA 805 Section 2.4.2 for specific CT configurations where a fire-induced open-circuit failure could result in a secondary fire. Specifically, the licensee stated that a fire in plant fire area C-21, lower cable spreading room, or plant fire area C-27, main control room, causes a fire-induced open-circuit fault and associated overcurrent conditions at the main generator CTs, which could result in a secondary fire in plant fire area TB-1, turbine building, due to the overheating of the main generator CTs. The licensee's request for approval of the deviation is documented in Approval Request 2 in LAR Attachment X.

The staff reviewed the licensee's request for approval and determined that the licensee has performed a bounding analysis that assumes that a fire in either the lower cable spreading room (fire area C-21) or the main control room (fire area C-27) causes a secondary fire as a result of an open on the secondary circuit of one or more of the main generator CTs. According to the licensee's analyses, overheating and subsequent ignition at the main generator CTs will result in a main generator trip, de-energization of the CTs and a subsequent plant trip. As such, the licensee expects that the overcurrent conditions will be of short duration. The licensee also determined that the effect of the secondary fire will be of minimal consequence, since there are no redundant safe shutdown systems, cables and components located in the secondary fire area, there is a lack of in-situ or transient combustibles in the immediate area of the CTs, and the turbine building is protected by a full area-wide automatic pre-action suppression system, which would promptly and effectively control or suppress the secondary fire. The NRC staff reviewed the information provided by the licensee and concludes that the licensee has adequately considered, using a bounding approach, that a secondary fire caused by an open circuit in the secondary winding of the main generator CTs will have no adverse impact on the ability to achieve and maintain the nuclear safety and radioactive release performance criteria in accordance with Attribute 3.3.2 [C], 3.5.2.1, and 3.5.2.5, as documented in Approval Request 2 in LAR Attachment X.

3.2.1.2 Attribute Alignment – Aligns with Intent

For certain NEI 00-01 Chapter 3 attributes, the licensee determined that the post-fire SSA 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 as follows:

- 3.1
- 3.1.1.9
- 3.2.1.6
- 3.2.2.1
- 3.4.1.5
- 3.4.2.4

Attribute 3.1 – Safe Shutdown Systems and Path Development: The goal of post-fire safe shutdown is to assure that one train of shutdown SSCs remains free of fire damage for a single fire in any single plant fire area. This goal is accomplished by determining those functions important to achieve and maintain hot shutdown. Safe shutdown systems, components, and cables are selected so that the capability to perform these required functions is a part of each safe shutdown path. The functions important to post fire safe shutdown generally include, but are not limited to the following:

- Reactivity Control

- Pressure Control Systems
- Inventory Control Systems
- Decay Heat Removal Systems
- Process Monitoring
- Support Systems
    - Electrical systems
    - Cooling systems

The entry for this attribute in LAR Table B-2 states that the NSCA meets the intent of the NEI guidance because the analysis uses the success criteria of safe and stable conditions from NFPA 805 rather than the deterministic definitions of hot shutdown and cold shutdown from the deterministic criteria of NEI 00-01. The Nuclear Safety Goal provided in 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 NRC staff concludes that complying with the intent of this section of NEI 00-01 is acceptable since the licensee's analysis is consistent with the requirements of NFPA 805. The NRC staff's evaluation of the licensee's treatment of safe and stable plant conditions is contained in section 3.2.2 of this SE.

Attribute 3.1.1.9 – Criteria/Assumptions (72-hour coping period): The deterministic safe shutdown criteria cited in NEI 00-01 states that the analysis should use a 72-hour coping period to perform damage repairs and achieve cold shutdown conditions. The requirements in NFPA 805 do not include this deterministic requirement. NFPA 805 requires that the fuel be maintained in a safe and stable condition during and following a fire in the plant. The NSCA has demonstrated that Callaway can achieve and maintain safe and stable conditions for at least 10 hours with the minimum shift operating staff before having to take action to recharge the nitrogen accumulators. This initial 10 hours provides sufficient time for the Emergency Response Organization (ERO) to respond and be available to support safe and stable actions to extend hot standby conditions. The NRC staff reviewed the prescribed actions and concluded that Callaway has adequately demonstrated the ability to achieve and maintain "safe and stable" conditions for an extended period of time. Since the licensee's analysis uses success criteria that is consistent with the requirement in NFPA 805 to maintain the fuel in a safe and stable condition, the NRC staff concludes that the licensee's statement of alignment to the intent of the endorsed guidance is acceptable (See Section 3.2.2 of this SE for additional details with respect to the evaluation of the 10-hour coping period for Callaway).

Attribute 3.2.1.6 – Criteria/Assumptions (Identify equipment that could spuriously operate or mal-operate): Unintended spurious or mal-operation of equipment and instrumentation can adversely impact the safe shutdown capability; therefore, all mechanical and electrical components must be evaluated for potential fire-induced spurious or mal-operation. The intent of the NEI guidance for this attribute is to organize the SSA so that spurious actuations/ mal-operations can be analyzed using a deterministic method presented in RIS 2004-03 (Reference 39). The identification of spurious equipment for the Callaway NSCA did not include binning as described in RIS 2004-03. However, the licensee included mechanical and electrical system components such as pumps, air operated valves, motor-operated valves, and solenoid-operated valves, fans, heaters, electrically controlled circuit breakers, instrumentation, dampers, etc., in the NSCA if they maintain a system boundary or if the spurious operation(s) of the components have a potential adverse impact on NSCA capabilities. Since the NSCA information is being used to support an RI/PB analysis using an FPRA, the binning referenced in RIS 2004-03 is not necessary. The FRE process is intended to more fully evaluate fire-induced spurious actuations/mal-operations and their impact on the ability to meet the NSPC. Based on the fact that the NSCA is being used to support the FRE PB approach, which is more rigorous than that referenced in the guidance, the NRC staff concludes that the licensee's statement of alignment to the intent of the endorsed guidance is acceptable.

Attribute 3.2.2.1 – Identify the System Flow Path for Each Shutdown Path: The goal of post-fire safe shutdown is to ensure that one safe shutdown train is free from fire damage. The intent of the NEI guidance for this attribute is to document the credited safe shutdown path and maintain that documentation as a part of the SSA. The licensee stated that it performed iterative reviews of piping and instrumentation diagrams (P&IDs), electrical drawings, instrument loop diagrams, etc., to identify the NSCA systems, and to identify and develop the NSCA system-to-component logic relationships (i.e., Boolean logic/success paths) and the NSCA component-to-component logic success path relationships (i.e., success paths). The licensee stated that marked up and annotated reviewed documents were not maintained as part of the NSCA record; however, the reviewed documentation (i.e., document numbers and revision levels) was recorded for configuration management. The licensee used a systematic process for identifying safe shutdown flow paths and supporting components and cables. Therefore, the NRC staff concludes that the safe shutdown success paths are adequately assessed by the licensee. Although the marked-up and annotated reviewed documents are not maintained as part of the NSCA record, the information that otherwise would be documented in the marked up documents (including all of the NSCA systems, components and cables, system-to-component relationships, and component-to-component logic relationships), is reflected in SAFE-PB, a computer database analytical tool that Callaway used to identify the success paths and the equipment and cables required to demonstrate that the NSPC of NFPA 805 are met for each fire area of the plant. Since post-transition changes affecting the NSCA will be captured and analyzed using SAFE-PB by qualified personnel, the NRC staff concludes that the licensee's statement of alignment to the intent of the endorsed guidance is acceptable.

Attribute 3.4.1.5 – Criteria/Assumptions (Where appropriate achieve and maintain cold shutdown within 72 hours, use repairs to equipment required in support of post-fire shutdown): This attribute ensures the capability to achieve and maintain post-fire cold shutdown within 72 hours if cold shutdown is credited to demonstrate NSPC. The intent of this attribute of the NEI guidance is to assure that the SSA meets the deterministic requirements for cold shutdown (including repairs). The requirements in NFPA 805 do not include these deterministic requirements. NFPA 805 requires that the fuel be maintained in a safe and stable condition during and following a fire in the plant. The licensee defines safe and stable conditions as being able to achieve and maintain the reactor in a hot standby plant operating state. The licensee determined that the NSCA demonstrated that Callaway can achieve and maintain safe and stable conditions for at least 10 hours with the minimum shift operating staff before having to take action to recharge the nitrogen accumulators. This initial 10 hours provides sufficient time for the ERO to respond and be available to support safe and stable actions to extend hot standby conditions. Based on the above, the NRC concludes that the licensee's statement of alignment to the intent of the endorsed guidance is acceptable (See SE Section 3.2.2, Safe and Stable, for additional details).

Attribute 3.4.2.4 – Develop Compliance Strategy or Disposition to Mitigate the Effects Due to Fire Damage to Each Required Component or Cable: All potential fire-induced damages to components or cables required for safe shutdown must be evaluated to ensure safe shutdown capability is not adversely impacted. The intent of this attribute of the NEI guidance is to inform the analyst of the available deterministic strategies available to address post-fire safe shutdown compliance issues. The resolution strategies available in accordance with NFPA 805 are different than described in NEI 00-01 Chapter 3. The analyst can use either the FM PB approach in accordance with NFPA 805 Section 4.2.4.1, or the FRE PB approach in accordance with NFPA 805 Section 4.2.4.2. The licensee has used the FRE PB approach for Callaway. The Callaway NSCA equipment resolutions identify and provide a traceable link for each component failure on a fire area basis that requires further engineering justification to be determined acceptable as-is (i.e., not having any adverse impact to the NSCA), or that requires further engineering review to identify and propose a plant change such as an operator manual action (OMA), or a physical plant modification. Each equipment and cable resolution includes a description to document the engineering review basis. NSCA cable resolutions identify and provide a traceable link for protected cables in the fire area (i.e., raceway protected by ERFBS, raceway embedded in concrete, raceway in buried duct bank through one or more manholes). Circuit analysis may be used to assess and disposition specific circuit failure modes (as documented in the NSCA equipment resolutions). NSCA equipment resolutions that propose recovery actions are identified as VFDRs and included in Attachment G of the LAR. Proposed plant modifications are included in Attachment S of the LAR. Since the licensee has used the FRE PB approach and identified VFDRs to be evaluated assessing risk, DID, and SM; the NRC staff concludes that the licensee's methodology for identifying and evaluating potential component and cable damages adequately address the above attribute and, therefore, the licensee's statement of alignment to the intent of the endorsed guidance is acceptable.

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 SSA 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. The NEI 00-01 Chapter 3 attributes identified in LAR Table B-2 as complying via this method are as follows:

- 3.1.2.5

Attribute 3.1.2.5 – Process Monitoring: The availability of process monitoring instrumentation provides assurance that plant parameters are kept within operating range during transient conditions. The intent of this attribute of the NEI guidance is to assure that the post-fire SSA will require sufficient instrumentation be available to be able to safely monitor and control the safe shutdown of the plant during and following a fire. NFPA 805 contains a similar requirement in Section 1.5.1(e), "Process Monitoring," that requires that process monitoring be capable of providing the necessary indication to assure the NSPC have been achieved and are being maintained. The Callaway NSCA model requires the following instruments be used for process monitoring:

- Reactor coolant temperature (T-hot / T-cold): These instruments are modeled in support of the decay heat removal performance goal.
- Pressurizer pressure and level: These instruments are modeled in support of the Inventory and pressure control performance goal.
- Neutron flux monitoring (source range): These instruments are modeled in support of the reactivity control performance goal.
- Level indication for various tanks: These instruments are included in the system logics for which the tank is required.
- Steam Generator (SG) level and pressure: These instruments are modeled in support of the decay heat removal performance goal.
- Diagnostic instrumentation for safe shutdown systems: Diagnostic instrumentation such as pump suction pressure, flow, and temperature are generally provided by local indicators that require no electrical power.

The information provided in the LAR indicates that several of these attributes do not meet the requirements in NEI 00-01. However, all of these deviations were approved by the NRC staff as part of the existing deterministic fire protection licensing basis. The use of these instruments is consistent with the minimum process monitoring instrumentation expectations identified in IN 84-09, "Lessons Learned from NRC Inspections of Fire Protection Safe Shutdown Systems (10 CFR 50, Appendix R)" (Reference 41), as previously approved by the NRC in the 10 CFR Part 50, Appendix R licensing basis, and is being carried forward with the NFPA 805 LAR.

Where beneficial to reduce operator burden, the licensee has included instruments that read out in the main control room in the model and logically associated them with the component being monitored. In addition, instruments which provide permissive or controlling signals to safe shutdown components are modeled in direct support of the component as part of the cable selection process. The licensee identified no changes to the minimum sets of process monitoring instruments as previously approved by the NRC. Based on the instrumentation provided by the licensee and the previous approval by the NRC staff, the staff concludes that the licensee has identified the process monitoring systems and equipment in the NSCA model.

3.2.1.4 NFPA 805 NSCA 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 Callaway post-fire SSA against the NFPA 805 NSCA requirements using the NRC-endorsed process in Chapter 3 of NEI 00-01, Revision 2 and documenting the results of the review in LAR Table B-2 in accordance with NEI 04-02.

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 concluded that the licensee's method is acceptable because it either:

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

3.2.2 Maintaining Fuel in a Safe and Stable Condition

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, NUREG-0800, Section 9.5.1.1 (Reference 20), and, in part on 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 licensee stated that the NFPA 805 licensing basis is to ensure that the plant can achieve and maintain the reactor fuel in a safe and stable condition at a temperature equal to or less than that required for hot standby, assuming that a fire event occurs during Mode 1 (Power Operation), Mode 2 (Startup), Mode 3 (Hot Standby), or Mode 4 (Hot Shutdown), up to the point at which the motor control center breakers for the residual heat removal loop suction isolation valves, BBPV8702A, BBPV8702B, EJHV8701A, and EJHV8701B, are unlocked and closed. As described in LAR Section 4.2.1.2, the NSCA demonstrates that Callaway can achieve and maintain safe and stable conditions for at least 10 hours with the minimum shift operating staff before having to take additional actions to recharge the nitrogen accumulators. The licensee stated that the initial 10 hours provides sufficient time for the ERO to respond and be available to support safe and stable actions to extend hot standby conditions. LAR Attachment C (Table B-3) identifies the systems and components credited with supporting safe and stable plant conditions by fire area. The licensee stated that the systems, functions, and components required to achieve and maintain safe and stable plant conditions post-fire per the NSPC of NFPA 805 are identified in a Callaway calculation.

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 TSs. TS action statements may require the licensee to bring the unit to conditions other than safe and stable as defined in NFPA 805 (numerous action statements require the unit to be in Cold Shutdown within a defined time frame). Accordingly, TSs must still be met.

The NFPA 805 requirement is to be able to maintain the fuel in a safe and stable condition as necessary until actions can be taken to place the plant in cold shutdown in accordance with the TSs. Since the licensee used a defined coping time in its analysis, the NRC staff needed to know more about the basis of the 10-hour coping time. In SSA RAI 6 dated March 2, 2012 (Reference 11), the NRC requesting the following:

- The physical or design constraints that form the basis of the defined time (what is the defined time based on?).
- What plant impact will occur if the time is exceeded? Describe any additional actions that must be taken to maintain safe and stable conditions beyond the time in sufficient detail to determine whether they are recovery actions or maintenance actions.
- Will the identified physical limitations have an adverse impact on risk?

In its response dated April 17, 2012 (Reference 6), the licensee stated that in the event the nitrogen accumulators cannot be recharged, the atmospheric steam dump function would eventually result in the cycling of the SG code safety valves. In addition, the turbine-driven auxiliary feedwater (TDAFW) flow control valves would fail open due to the loss of nitrogen pressure. However, TDAFW flow can be controlled by locally throttling a manual valve. The licensee also indicated that components and/or cables associated with the actions to recharge the nitrogen accumulators are included in the NSCA equipment list. Furthermore, actions to recharge the nitrogen accumulators and to manually throttle the TDAFW flow control valve are addressed in plant procedures and have been demonstrated to be feasible. The licensee also stated that a qualitative risk assessment had been performed for this scenario, which demonstrated that the risk of not being able to maintain the defined safe and stable conditions is acceptably low beyond the defined coping time limit of 10 hours. The NRC staff reviewed the actions required to maintain safe and stable conditions and concludes that the licensee adequately demonstrated the capability to achieve and maintain safe and stable conditions post-fire. Based on the nature of the required actions (to recharge the nitrogen accumulators), the ability of the ERO to provide additional resources and the possibility of using recovery actions to perform actions to perform the same functions upon loss of nitrogen, the NRC staff concludes that the licensee's statement of alignment to the intent of the endorsed guidance is acceptable.

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 Callaway post-fire SSA against the NFPA 805 NSCA requirements using the NRC-endorsed process in Chapter 3 of NEI 00-01, Revision 2 and documenting the results of the review in LAR Table B-2 in accordance with NEI 04-02. 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.

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 pressurized-water reactors (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 Callaway 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 Callaway 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 included the nuclear safety 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 PB evaluation performed in accordance with NFPA 805, Section 4.2.4, demonstrated that the integrated assessment of risk, DID, and SM for the fire area was acceptable.

Therefore, the NRC staff concludes 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.

3.2.4 Assessment of Multiple Spurious Operations

NFPA 805, Section 2.4.2.2.1, "Circuits Required in Nuclear Safety Functions," states, in part, 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. 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 PRA evaluation shall address the risk contribution associated with all potentially risk-significant fire scenarios. Because the PB approach taken at Callaway used FREs in accordance with NFPA 805 Section 4.2.4.2, "Use of Fire Risk Evaluation," the licensee is required to adequately identify and include potential MSO combinations 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 Callaway.

As part of the NFPA 805 transition project, Callaway reviewed and evaluated the susceptibility to fire-induced MSOs. The licensee stated that the process was conducted in accordance with NEI 04-02 and RG 1.205, as supplemented by FAQ 07-0038 (Reference 44). The review method used insights from the FPRA developed in support of transition to NFPA 805 and consists of the following:

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

This process supports the transition to the new licensing basis. Post-transition changes will use the RI/PB change process. The post-transition change process for the assessment of a specific MSO will be a simplified version of this process, and may not need the level of detail shown in the following section (e.g., an expert panel may not be necessary to identify and assess a new potential MSO. Identification of new potential MSOs may be part of the plant change review process or inspection process).

Callaway used MSO expert panels to address the MSO issue. An initial MSO expert panel was conducted at Callaway in July 2007, and the results were integrated into the NFPA 805 NSCA and provided as input into the FPRA development effort. A second MSO expert panel assessment was conducted at Callaway in January 2008. This panel discussed and disposed open items from the original panel and addressed new generic MSOs that had been identified since the first panel. The results of the second expert panel assessment were documented and used to update to the original report. The MSO expert panel report was then reviewed by the PWR Owner's Group (PWROG) Peer Review Team, and updated to include the generic MSO list.

The expert panels were comprised of industry experts in the areas of PRA, HRA, electrical engineering, Appendix R and FPE, NFPA 805 project management, and nuclear plant operations. In addition, the expert panels had the ability to request supplemental engineering support (e.g., transient analysis, systems engineers, etc.) as needed. The endorsed industry guidance provided in NEI 04-02, as supplemented by FAQ 07-0038 and RG 1.205, states that the expert panel should be made up of a diverse team of experts in operations, engineering, electrical (Appendix R), PRA and others (fire protection). Based on the review performed by the NRC staff during the NFPA 805 site audit on January 23-27, 2012, of the Callaway NFPA 805 Fire PRA MSO Expert Panel Report, which provides a description for each of the experts including the area of expertise, professional background, qualifications, and years of experience, the NRC staff concludes that the make-up of the expert panels covered all necessary subject areas and included qualified members and, therefore, was acceptable.

The licensee stated that the process used to identify MSOs by the expert panels is in accordance with NEI 04-02 and RG 1.205, as supplemented by FAQ 07-0038. This process includes the following 5 steps:

- Step 1 – Identify potential MSOs of concerns. Information sources that were used as inputs are:
    - Piping and instrumentation diagrams (P&IDs)
    - Post-fire SSA (e.g., NEI 00-01, Revision 1, Chapter 3)
    - Generic lists of MSOs (e.g., from Owners Groups and/or NEI 00-01)
    - Self-assessment results (e.g., NEI 04-06 (Reference 63) assessments performed to address RIS 2004-03)
    - PRA insights (e.g., NEI 00-01, Revision 1, Appendix F)
    - Operating experiences (e.g., licensee event reports, NRC inspection findings, etc.)
- Step 2 – Conduct an expert panel to assess plant-specific vulnerabilities.
- Step 3 – Update the FPRA model and NSCA to include the MSOs of concern.
- Step 4 – Evaluate for NFPA 805 compliance.
- Step 5 – Document results.

During Step 2, a system level review of each of the NSPC is performed to identify failure mechanisms that could defeat the nuclear safety pathways (caused either by spurious operation or fire-induced failures) using all available information sources (P&IDs, SSAs, generic MSO lists, self-assessment results, PRA insights, and operating experiences).

The licensee stated that the MSO expert panel's members attended an MSO project instruction training session and followed a set of established ground rules based on NEI 00-01, Revision 1, in order to collectively achieve consensus during the MSO evaluation process. The MSO expert panel's report described 47 functional areas where MSO scenarios were developed and systematically evaluated.

The NRC staff reviewed the licensee's expert panel process for identifying circuits susceptible to MSO 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. The NRC staff concludes that the licensee's approach for assessing the potential for MSO combinations is acceptable for use at Callaway.

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, in part, 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 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. OMAs are actions performed by plant operators to manipulate components and equipment from outside the main control room (MCR) 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.

The licensee stated that it used the guidance in NEI 04-02 and FAQ 07-0030 (Reference 43) to transition pre-transition OMAs and to determine the population of post-transition RAs. This process consists of the following steps:

- Step 1: Clearly define the primary control station (PCS) and determine which pre-transition OMAs are taken at the PCS. Activities that occur in the MCR are not considered pre-transition OMAs. Activities that take place at the PCS or in the MCR are not RAs, by definition.

- Step 2: Determine the population of RAs that are required to resolve VFDRs (to meet the risk acceptance criteria or maintain a sufficient level of DID).

- Step 3: Evaluate the additional risk presented by the use of RAs required to demonstrate the availability of a success path.

- Step 4: Evaluate the feasibility of the RAs.

- Step 5: Evaluate the reliability of the RAs.

The review results are documented in Callaway site-specific fire protection calculations. Attachment G to the LAR includes the summary of the results from the process.

The licensee based its approach for transitioning OMAs into the RI/PB FPP as RAs on NEI 04-02, Section 4.6, "Regulatory Submittal and Transition Documentation," as endorsed with exceptions by RG 1.205. The population of OMAs addressed during the NFPA 805 transition process included the existing OMAs in the deterministic FPP, as well as those being added during the NFPA 805 transition to address MSOs and as a result of development of the FPRA.

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 it subjected all RAs (including DID-RAs) to a feasibility review. The feasibility criteria used were based on FAQ 07-0030, which lists the attributes used to assess RA feasibility as follows:

- Demonstrations – The proposed recovery actions should be verified in the field to ensure the action can be physically performed under the conditions expected during and after the fire event.

- Systems and Indications – Consider availability of systems and indications essential to perform the recovery action.

- Communications – The communications system should be evaluated to determine the availability of communication, where required for coordination of recovery actions.

- Emergency Lighting – The lighting (fixed and/or portable) should be evaluated to ensure sufficient lighting is available to perform the intended action

- Tools-Equipment – Any tools, equipment, or keys required for the action should be available and accessible. This includes consideration of SCBA and personal protective equipment if required. (This includes staged equipment for repairs.)

- Procedures – Written procedures should be provided.

- Staffing – Walk-through of operations guidance (modified, as necessary, based on the analysis) should be conducted to determine if adequate resources are available to perform the potential recovery actions within the time constraints (before an unrecoverable condition is reached), based on the minimum shift staffing. The use of essential personnel to perform actions should not interfere with any collateral industrial fire brigade or control room duties.

- Actions in the Fire Area – When Recovery Actions are necessary in the fire area under consideration or require traversing through the fire area under consideration, the analysis should demonstrate that the area is tenable and that fire or fire suppressant damage will not prevent the recovery action from being performed.

- Time – Sufficient time to travel to each action location and perform the action should exist. The action should be capable of being identified and performed in the time required to support the associated shutdown function(s) such that an unrecoverable condition does not occur. Previous action locations should be considered when sequential actions are required.

- Training – Training should be provided on the post-fire procedures and implementation of the recovery actions.

- Drills – Periodic drills, which simulate the conditions to the extent practical (e.g., communications between the control room and field actions, the use of self-contained breathing apparatus (SCBAs) if credited, appropriate use of operator aids) should be performed.

The NRC staff reviewed the licensee's statements in the LAR regarding the above considerations and concludes that the licensee has followed the endorsed guidance of NEI 04-02, as enhanced by FAQ 07-0030, and RG 1.205 to transition OMAs in the existing deterministic FPP to 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 as described by the licensee are acceptable based on conformance with the endorsed guidance contained in NEI 04-02 as enhanced by FAQ 07-0030. The additional risk of the use of RAs is discussed in SE Section 3.4.4, "Additional Risk Presented by Recovery Actions." RAs are also discussed in SE Sections 3.5.1.6, "Recovery Actions," and 3.5.1.7, "Recovery Actions Credited for Defense-in-Depth."

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. First, the NRC staff concluded that the safe and stable condition, proposed by the licensee, is acceptable. Second, 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 NSCA 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 Callaway is considered comprehensive and thorough. The MSO identification process started with an extensive review of plant systems and drawings to determine potential pathways. The initial review was conducted using the existing models and engineering documentation to identify functional failure paths that would be important to risk. This initial review was then supplemented by generic industry lists. The MSO identification process resulted in a list of potential MSO pathways for consideration by the MSO expert panels. Potential MSO combinations were identified and equipment and cables of concern were included in the NSCA as well as the applicable FREs. The NRC staff considers the licensee's approach for assessing the potential for MSO combinations to be 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, regarding the identification of RAs and other actions required to be taken at a PCS. Therefore, this process meets the regulatory requirements of 10 CFR 50.48(c) and NFPA 805.

3.3 Fire Modeling

NFPA 805 allows both fire modeling (FM) and FRE as PB alternatives to the deterministic approach outlined in the standard. These two PB approaches are described in NFPA 805, Sections 4.2.4.1 and 4.2.4.2, respectively. Although FM and FRE are presented as two different approaches for PB compliance, the FRE approach generally involves some degree of FM to support engineering analyses and fire 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 FM as part of the transition to NFPA 805 at Callaway. In LAR Section 4.5.2, the licensee indicated that, in lieu of the FM 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, "Fire Model Utilization in the Application," the licensee indicated that FM was performed as part of the FPRA development. Therefore, the NRC staff reviewed the technical adequacy of the Callaway FRE, including the supporting FM analyses, as documented in Section 3.4.2 of this SE, to evaluate compliance with the NSPC.

The licensee did not propose any FM methods to support PB evaluations in accordance with NFPA 805, Section 4.2.4.1, as the sole means for demonstrating compliance with the NSPC. Therefore, the scope of the licensee's self-approval capability does not include utilizing the FM PB approach in accordance with NFPA 805, Section 4.2.4.1.

3.4 Fire Risk Assessments

This section addresses the licensee's FRE PB method, which is based on NFPA 805, Section 4.2.4.2. The FM PB method of NFPA 805, Section 4.2.4.1 was not used for this application.

NFPA 805, Section 4.2.4.2, "Use of Fire Risk Evaluation," 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 2.4.4.1. The fire risk shall be calculated using the approach described in 2.4.3.

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 PB 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)

As a supplement to the definition of DID provided in NFPA 805, Section 1.2, the NRC-endorsed guidance in NEI 04-02, Section 5.3.5.2, states the following:

> In general, the defense-in-depth requirement is satisfied if the proposed change does not result in a substantial imbalance in:
>
> - Preventing fires from starting
> - Detecting fires quickly and extinguishing those that do occur, thereby limiting fire damage
> - Providing adequate level of fire protection for structures, systems and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed

The NRC staff reviewed LAR Section 4.8.1, "Required Fire Protection Systems," and LAR Table 4-3, "Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features," 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 Callaway.

When implementing the PB approach, the licensee followed the guidance contained in Section 5.3, "Plant Change Process," of NEI 04-02, which includes a detailed consideration of DID as part of the change process. The license documented the method used to meet the DID requirements of NFPA 805 in LAR Table 4-3 and LAR Attachment C, Table B-3. For each of the major fire protection DID attributes, the licensee provided several examples of how that attribute was addressed, along with a discussion of the considerations used in evaluating that element.

In PRA RAI 18 dated March 2, 2012 (Reference 11) the NRC staff asked the licensee to provide a description of how DID was addressed for VFDRs. In its response dated April 17, 2012 (Reference 6), the licensee stated the following:

> <u>Defense-in-Depth Approach</u>
>
> A review of the impact of the VFDRs on defense-in-depth shall be performed, regardless of the risk evaluation method used. The review...is typically qualitative and should address each of the elements with respect to the proposed change.
>
> 1) Evaluate the fire area for the impact of the VFDRs on fire protection defense-in-depth...

2) In general, the defense-in-depth requirement is satisfied if the proposed change does not result in a substantial imbalance among these [fire protection defense-in-depth] elements...

3) In evaluating defense-in-depth, it may become necessary to consider the potential for risk significant fire scenarios to impact VFDRs.... For purposes of defense-in-depth, "potentially risk significant" fire scenarios could be characterized as follows, for example:

   - A scenario in which the calculated risk is equal to or greater than 1E-6/year for CDF and/or 1E-7/year for LERF...

   - A scenario in which the calculated risk falls between 1E-6/year and 1E-8/year for CDF, or between 1E-7/year and 1E-9/year for LERF, and where DID echelon 1 [prevent fires from starting] and 2 attributes [rapidly detect, control and extinguish promptly those fires that do occur, thereby limiting fire damage] are causing a significant reduction in risk... [Note: echelon 3 is to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed.]

   - A scenario with a high consequence (i.e., CCDP [conditional core damage probability]>E-1)...

4) Fire protection features and systems relied upon to ensure defense-in-depth should be clearly identified in the assessment...

5) Verify that defense-in-depth is maintained by assessing and documenting that the balance is preserved among prevention of core damage, prevention of containment failure, and mitigation of consequences...

6) Each fire area shall be evaluated for the need to incorporate defense-in-depth enhancements to provide assurance that plant performance goals can be achieved and maintained...

7) Provide the results of the defense-in-depth review in a tabular format...

In PRA RAI 18 dated March 2, 2012 (Reference 11), the NRC staff also asked the licensee to describe how its reliance upon multiple, time-critical, or complex recovery actions is evaluated to ensure there is no over reliance upon operator actions as part of its DID strategy. In its response dated April 17, 2012 (Reference 6), the licensee stated the following:

> <u>Evaluation of Multiple Recovery Actions</u>
>
> ... [A]ll recovery actions (RA's) credited to meet the Nuclear Safety Performance Criteria must be demonstrated to be feasible. As part of determining RA

> feasibility, a thermal-hydraulic calculation was developed to identify allowed RA completion times by fire area.
>
> The RA feasibility evaluation was then performed ... by considering the expected plant response for a given fire area in conjunction with the following eleven criteria: (1) Draft Fire Procedures, (2) Systems and Indications, (3) Tools-Equipment, (4) Communications, (5) Emergency Lighting, (6) Demonstrations, (7) Actions in the Fire Area, (8) Time, (9) Staffing, (10) Training, And (11) Drills....
>
> ...Based on the evaluation all RA's were determined to be feasible. Therefore, the feasibility determination included consideration of multiple time critical actions credited for a fire area if they existed....
>
> The Fire Human Reliability Analysis task (FHRA...) evaluated the reliability of each RA given the fire-specific scenario. The FHRA used the results of the feasibility assessment, draft procedure guidance, walkdown information and timeline development as input to performing the Human Error Probability (HEP) calculation.... [T]he FHRA reliability determination also included consideration of multiple time critical actions credited for a fire area if they existed.

LAR Table 4-3 and Attachment C, Table B-3 document the results of the licensee's review of fire suppression and fire detection systems at Callaway. The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAI 18 summarized above, and concludes that the transition process included a detailed review of fire protection DID. The NRC staff concludes that the evaluation of DID is acceptable because the licensee's process and results follow the endorsed guidance in NEI 04-02, and are consistent with the guidance in RG 1.205.

3.4.1.2 Safety Margins

Although not a part of the requirements of NFPA 805, and thus not required under 50.48(c), Section A.2.4.4.3 of Appendix A to NFPA 805, provides the following background related to the meaning of the term "safety margins":

> An example of maintaining sufficient safety margins occurs when the existing calculated margin between the analysis and the performance criteria compensates for the uncertainties associated with the analysis and data. Another way that safety margins are maintained is through the application of codes and standards. Consensus codes and standards are typically designed to ensure such margins exist.

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 (SMs):

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

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

LAR Section 4.5.2, "Performance-Based Approaches," states that SMs were considered as part of the transition process. Section 4.5 states that the licensee reviewed SM for each fire area containing VFDRs and documented the review in the FRE for that fire area.

In PRA RAI 18 dated March 2, 2012 (Reference 11), the NRC staff asked the licensee to provide a description of how SMs were addressed for VFDRs. In its response dated April 17, 2012 (Reference 6), the licensee stated the following:

> <u>Safety Margin Approach</u>
>
> The evaluation addresses whether:
>
> (1) Codes and Standards or their alternatives accepted for use by the NRC are met, and
>
> (2) Safety analysis acceptance criteria in the licensing basis...are met, or provide sufficient margin to account for analysis and data uncertainty.
>
> These evaluations can be grouped into categories. These categories are:
>
> 1. Fire Modeling
> 2. Plant System Performance
> 3. PRA Logic Model
> 4. Other
>
> 1) Fire Modeling
>
> If a performance based approach is used, the margin between the parameters describing the [maximum expected fire scenario (MEFS)] and the [limiting fire scenario (LFS)] and the process of judging the adequacy of that fire modeling margin is required... The level of review...considered here involves the integration of that margin with the potential consequences of the upset, or damage, that may occur given the LFS....
>
> 2) Plant System Performance
>
> The development of the fire risk assessment may involve the re-examination of plant system performance given the specific demands associated with the postulated fire event. The methods, input parameters, and acceptance criteria...needs to be reviewed against that used for the plant design basis events....
>
> 3) PRA Logic Model

This subtask evaluates results of the Fire PRA model to verify that the safety margins have not changed. The CDF and LERF importance measures of components in the cutset results will be evaluated to verify that events with high importance values have reasonable failure probabilities for the scenarios of interest... The results of each risk evaluation will be evaluated against the base case fire results to determine that no single event has undue influence on the results of the change analysis...

4) Other

This category addresses any other analyses not addressed above. The general requirements related to codes and standards, and acceptance criteria, provided above apply.

Based on the statements provided in LAR Section 4.5.2 and confirmed by NRC staff observations during the NFPA 805 site audit, the NRC staff concludes that the licensee either used appropriate codes and standards (or alternatives accepted for use by the NRC), met the safety analyses acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses, etc.), or provided sufficient margin to account for analysis and data uncertainty.

The NRC staff concludes that the evaluation of SMs is acceptable because the licensee's process and results follow the endorsed guidance in NEI 04-02, and are consistent with the guidance in RG 1.205.

3.4.1.3 Defense-in-Depth and Safety Margin Conclusion

The licensee's FRE process included a detailed review of fire protection DID and SM. The individual FREs, LAR Table 4-3 and LAR Attachment C Table B-3 document the results of the DID and SM review. The NRC staff concludes that the licensee's evaluation related to DID and SM is acceptable because the licensee's process and results followed the endorsed guidance in NEI 04-02, and are consistent with the NRC staff guidance in RG 1.205 and RG 1.174 (Reference 17).

3.4.2 Quality of the Fire Probabilistic Risk Assessment

In reviewing an RI LAR, the NRC staff evaluates the plant-specific PRA models and their application as proposed in the LAR. 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 LAR, 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 S, "Plant Modifications and Items to be Completed During

Implementation," Attachment U, "Internal Events PRA Quality," Attachment V, "Fire PRA Quality," and Attachment W, "Fire PRA Insights."

The licensee developed its FPRA model by modifying 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 developed its FPRA model using the guidance of NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" (References 24, 25, and 26). The model addresses both Level 1 (CDF) and partial Level 2 (i.e., LERF only) PRA during at-power conditions.

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 FPRA model, beyond those identified and scheduled to be implemented as part of the transition to an FPP based on NFPA 805. Therefore, the NRC staff concludes that the FPRA model for Callaway represents the as-built, as-operated, and maintained plant as it will be configured after full implementation of 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 FPRA 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.

3.4.2.1 Internal Events PRA Model

In LAR Section 4.5.1.1 the licensee evaluated the technical adequacy of the portions of its internal events PRA model used to support development of the FPRA model using the ASME/ANS RA-Sa-2009, "Standard for Level 1/LERF PRA for Nuclear Power Plant Applications" (ASME/ANS PRA Standard) (Reference 35) and RG 1.200 (Reference 18), as discussed below:

> The Callaway Plant internal events PRA (PRA Update 4) was the starting point for the Fire PRA. In 2006, the Callaway Plant internal events PRA underwent a gap assessment, conducted by Scientech, against the Capability Category II requirements of ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," with ASME RA-Sa-2003 and ASME Addenda RA-Sb-2005...
>
> To move the PRA Update 4 internal events model to Capability Category II of the Standard, a large-effort PRA upgrade project was planned and initiated in 2007...
>
> All of the internal events PRA gap analysis Findings that could affect the Fire PRA have been addressed and closed.
>
> The Callaway Plant NFPA 805 Fire PRA Quality Summary report (17671-015) was developed to support the FPRA peer review... The FPRA peer review was

> conducted October 26 through October 30, 2009 and reviewed this report as part of the PRA Maintenance and Update element. The Peer Review team found the categorization and dispositioning acceptable, and had no findings related to this report.

Supporting requirements are detailed, focused statements of "good PRA practice" which, collectively, comprise what is deemed satisfactory for a technically adequate PRA. For each supporting requirement, there are three degrees of satisfaction with the requirement, which are referred to as the Capability Categories. Three Capability Categories are common (I, II, and III), with I being the minimum, II is considered widely acceptable, and III going beyond the state-of-the-art. For each supporting requirement, a PRA reviewer (in the peer review) assigns one of these Capability Categories.

The original internal events PRA peer review was conducted by the Westinghouse Owners Group from November 5-10, 2000, with the final report issued in January 2003. PRA Update 3 incorporated the resolutions of the findings from this peer review. In PRA RAI 01 dated March 2 and June 6, 2012 (References 11 and 12, respectively), the NRC staff questioned the licensee as to alignment of the internal events PRA gap assessment with RG 1.200, Revision 2. The licensee responded to the RAI by letters dated April 17 and July 12, 2012 (References 6 and 7, respectively). In Reference 6, the licensee stated, in part, that:

> The [internal events PRA] peer review used the ASME-RA-Sb-2005 (Dec. 2005) version of the PRA standard. This version of the standard incorporated NRC comments from RG 1.200, Trial for Use, Attachment A (Jan. 2004). As such, the internal events PRA was peer reviewed against the clarifications and qualifications presented in the latest revision of RG 1.200 available at the time of the review. A self-assessment of the internal events PRA against the RG 1.200, Rev. 2 clarifications and qualifications to determine if any gaps exist is in progress and will be completed, with any resolutions completed before transition to NFPA 805 occurs. This is being tracked by Implementation Item 12-805-001....

The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAI 01 summarized above and concludes that the current PRA models are adequate to support the transition FRE because (1) the licensee confirms that its internal events PRA either meets Capability Category II for the applicable supporting requirements or has provided adequate justification as to why satisfaction at a lower category is sufficient in the context of the NFPA 805 transition risk calculations as discussed below, and (2) the licensee has committed to completing its gap assessment against RG 1.200 and any associated resolutions before transition to NFPA 805 (i.e., prior to the using the PRA to support post-transition PCEs) as tracked by implementation item 12-805-001.

One finding from the internal events PRA peer review, originally assigned to Capability Category I, was disposed as closed at Capability Category II. The upgrade required that components for the Data Analysis element be grouped according to characteristics of their usage. This was done, even though the licensee reasoned that addressing this finding would not appreciably impact the results of PRA applications. The NRC staff concludes that the licensee's response is acceptable because both the internal events and FPRAs employed the upgraded data, as required to meet Capability Category II.

In another finding from the internal events PRA, the peer review remarked on the lack of common-cause failure (CCF) modeling for battery chargers and breakers plus the need to update the quantification of CCF probabilities. In PRA RAI 04a dated March 2, 2012 (Reference 11), the NRC staff requested that the licensee provide the current status of CCF modeling in the internal events PRA, including how this also applies to the FPRA. In its response dated April 17, 2012 (Reference 6), the licensee provided the current status of CCF modeling in both the internal events PRA and FPRA, indicating:

1. Breakers for pumps and diesels are considered within the component boundaries as defined in NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," February 2007 (Reference 64), eliminating the need for a specific CCF term.

2. Bus load breakers for distribution of offsite power are not subject to CCF, except for two which supply alternate power to the 4160V emergency busses and are, therefore, modeled with a CCF event.

3. Battery chargers are modeled for CCF.

4. The latest PRA update for internal events employs the CCF equations from NUREG/CR-5485, "Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment," November 1998 (Reference 65), using factors and values from WCAP-16607-NP, Revision 1, "Common Cause Failure Parameter Estimates," June 2008 (Reference 66).

Given the clarification of how the CCF modeling was performed, the NRC staff concludes that the disposition of this finding is acceptable for the application because this current status demonstrates adequate incorporation of the CCF modeling into the PRA.

The licensee identified the resolution of the findings from the internal events PRA peer review in LAR Attachment U. The NRC staff's review and conclusion for the licensee resolution of each of the facts and observations (F&O) is summarized in the F&O Table (Reference 67). In its response to PRA RAI 01 dated April 17 and July 12, 2012 (References 6 and 7, respectively), the licensee confirmed that internal events PRA peer review was performed consistent with RG 1.200, Revision 2. The NRC staff concludes that the licensee dispositions for all internal events PRA peer review findings are acceptable as summarized in the F&O Table. The RAI response cited with regard to the open finding was deemed adequate as it indicated the appropriate upgrade and update has been performed.

3.4.2.2 Fire PRA Model

The licensee evaluated the technical adequacy of the Callaway FPRA model by conducting a peer review of the FPRA model using Part 4 of the ASME/ANS PRA Standard (Reference 35) and RG 1.200, Revision 2, as discussed in its LAR dated August 29, 2011:

> The FPRA peer review was conducted October 26 through October 30, 2009 and reviewed this report as part of the PRA Maintenance and Update element....
>
> The Callaway Plant Fire PRA (Callaway Plant model of record 3Q09-FPRA) was peer reviewed against the requirements of ASME/ANS RA-Sa-2009, Part 4. The PWR Owner's Group (PWROG) issued a report containing the results of the Callaway Plant Fire PRA Review on March 9, 2010 (LTR-RAM-II-10-019). The identification and resolution of the high level findings from the PWROG Fire PRA Review are summarized in Attachment V.
>
> Each of the findings from the fire PRA peer review has either been addressed with a change in the FPRA model or evaluated to have no impact on the Fire PRA. The FPRA Peer Review findings that were evaluated to have no impact either related to documentation improvements or final resolution of technical issues that are not expected to have a significant impact on the Fire PRA risk metrics and insights.

The NRC staff questioned the licensee, in PRA RAI 02 dated March 2, 2012 (Reference 11), as to alignment of the FPRA peer review with RG 1.200, Revision 2. The licensee responded by letter dated April 17, 2012 (Reference 6), as follows:

> The Fire PRA Peer Review team (October 2009) used the clarifications and qualifications to the PRA standard as presented in Regulatory Guide (RG) 1.200, Revision 2... [T]he database used for the peer review process during the Westinghouse [Owners Group] Peer Reviews includes the most up to date RG 1.200 clarifications and qualifications, which facilitates and emphasizes their inclusion during the review. The Fire PRA Peer Review is therefore consistent with the clarifications and qualifications in RG 1.200, Rev. 2.

Since the licensee verified that the clarifications and qualification of RG 1.200, Revision 2, were considered during the peer review, the NRC staff concludes that the response is acceptable.

The licensee stated that no open findings remained from the FPRA peer review. The NRC staff conclusions regarding the disposition of all the findings are summarized in the F&O Table (Reference 67). However, the licensee chose to report both findings and suggestions in Attachment V of the LAR, and one open F&O, of which findings and suggestions (F&S) are a subset, was the subject of an RAI from the NRC staff, although cited as a suggestion. The FPRA peer reviewers suggested that flow diversion paths that had been screened out in the internal events PRA be revisited. In PRA RAI 06c dated March 2, 2012 (Reference 11), the NRC staff questioned whether the disposition to defer this effort to a future update of the FPRA reflected the current status, or whether the effort had been completed. In its response dated April 17, 2012 (Reference 6), the licensee stated, in part, that:

> The Callaway MSO search for flow diversion paths did not rely on the internal events PRA. The MSO Expert Panel had been supplied with instructions and methods for identifying MSO scenarios... These instructions were not included as part of the MSO Expert Panel report for the fire PRA Peer Review.

The NRC staff concludes that the licensee's response is acceptable because it completed the effort cited as pending in the disposition.

One finding remained at Capability Category I from the FPRA Peer Review. The licensee stated that no outlier behavior exists for the plant-specific fire ignition frequency, such that no Bayesian update was warranted. However, the peer reviewers found four electrical cabinet fires and two events cited as outliers, thereby questioning the claim of no such behavior. Pending resolution, the peer reviewers assigned this supporting requirement to Capability Category I. In its disposition, the licensee indicated the all plant-specific fire events were now addressed and dispositioned for exclusion from a Bayesian update. As part of the audit of supporting material, the NRC staff noted only two potential fire events occurring beyond the time frame of the generic database used to generate fire frequencies. Neither was classified as even "potentially challenging," with the basis provided. The NRC staff reviewed these bases and found them acceptable because they are consistent with the classification of generic fire events as per methods acceptable to the NRC staff.

The NRC staff questioned the licensee's disposition of a finding related to supporting requirements on statistically based parametric uncertainty intervals as meeting Capability Category II, since it was based on claiming that conservatism in methods and data outweigh uncertainty. The NRC staff does not agree that a claim of conservatism can generally replace a parametric uncertainty analysis because the parametric uncertainty analysis in an evaluation provides a well-defined quantitative measure while the claim of conservatism in an evaluation is ill-defined and subjective and therefore one cannot generally be measured against the other. In PRA RAI 22 dated March 2 and June 6, 2012 (References 11 and 12, respectively), the NRC staff requested that the licensee justify the disposition by providing at least a sensitivity analysis as an estimate of the uncertainty interval. A sensitivity analysis (also called sensitivity study) is the common practice in both PRA and other engineering analyses where the value of a parameter that is used for a quantitative evaluation is varied between its extremes (low and high) without any other parameters being varied at the same time. This enables determination of how much the result of the evaluation is affected by the potential variability in the parameter. In its response dated July 12, 2012 (Reference 7), the licensee provided a summary of all sensitivity analyses performed, including a conservative summation of the CDF, LERF, delta-CDF, and delta-LERF. These remained below the numerical acceptance thresholds in RG 1.174, as cited in RG 1.205. The NRC staff considers the provision of the quantitative, combined sensitivity results as acceptable justification for not including a parametric uncertainty evaluation in the transition risk evaluations. The NRC staff also concludes that the performance of the sensitivity study demonstrates that the licensee can address uncertainty in the post-transition self-approval calculations as needed and, therefore, the licensee has adequately dispositioned the finding.

Staff Evaluation of Selected Responses to Requests for Additional Information (RAIs). As part of the NRC staff review, both the LAR and supporting material were reviewed, including an NFPA 805 site audit in January 2012. In some cases, the NRC staff requested supplementary information to assess the adequacy of the FRE. In several cases, the RAI response raised issues that required further clarification and these issues are discussed below. Sensitivity evaluations for several additional issues that did not lead to changes in the FPRA are provided in Section 3.4.7 of this SE.

Presence of High-Density Polyethylene (HDPE) Piping in the Essential Service Water (ESW) System. Recently installed HDPE piping in the ESW system in Fire Area C-1 was identified as susceptible to fire damage subsequent to the LAR submittal. In PRA RAI 04f dated March 2, 2012 (Reference 11), the NRC staff asked the licensee to identify and evaluate VFDRs associated with this piping. The licensee response dated July 12, 2012 (Reference 7), cited an FRE for Fire Area C-1 that addressed the VFDRs, including justification for assuming a reduced HRR due to transient combustible fires. FM indicated no fire damage to the HDPE piping, resulting in associated risk and delta-risk values of zero. In the process of this re-evaluation, the licensee identified two other locations where elastomeric components, similar to HDPE piping, were present. Upon re-examination of the corresponding FREs, the licensee concluded that these components did not meet the criteria of a VFDR because both could be recovered by operator action in the MCR. The NRC staff reviewed the information provided by the licensee in its response to PRA RAI 04f and concludes that the licensee's response is acceptable because it identified any VFDRs associated with the HDPE and provided an evaluation for those cited as VFDRs.

Assumption of Human Error Probability (HEP) of 0.1 to Successfully Operate the Alternate Shutdown Panel. In PRA RAIs 07b and 35 dated March 2 and December 11, 2012 (References 11 and 14, respectively), the NRC staff asked for justification of the assumption that a HEP of 0.1 is an adequate estimate for the "total probability of failing to evacuate and establish local control successfully" (i.e., transfer of control from the MCR to an alternate shutdown panel due to fire). In its responses dated April 17, 2012 and February 19, 2013 (References 6 and 8, respectively), the licensee stated that in calculating the CCDP, failure of equipment was accounted for independently of the HEP. There are limitations with existing HRA methods to quantify the MCR Abandonment HEP and capture the complexities of the plant response. Rather than develop a detailed HRA, the licensee response investigated the CDF and LERF margins for successively higher HEP values via two sensitivity analyses. The first examined the impact of a bounding HEP set to failed (probability of 1.0; i.e., a CCDP of 1.0 was assigned to all MCR evacuation scenarios). Using this HEP/CCDP, the CDF for the MCR increases from $7.8\text{x}10^{-7}$/yr to $1.38\text{x}10^{-6}$/yr. The second sensitivity employed a more-realistic HEP/CCDP of 0.5. In this case, the MCR CDF increased only to $1.04\text{x}10^{-6}$/yr. For each of these sensitivity cases, Callaway would continue to meet the risk acceptance criteria of RG 1.205. The plant's model of record continues to assume an MCR Abandonment HEP of 0.1, and considers 0.5 to be an upper bound (and has quantified with an extreme bound of 1.0). In all cases, Callaway meets the RG 1.205 risk acceptance criteria for transition. Note that the acceptable results of these two sensitivities are due primarily to the low evacuation probabilities for panel fires in the MCR, and the train separation of the cable spreading rooms. In response to PRA RAI 36 dated August 5, 2013 (Reference 9), the licensee indicated that it will revise its FPRA to use the HRA calculator with guidance in NUREG-1921 (Reference 31) via implementation item 13-805-005 (See LAR Attachment S, Table S-3) to evaluate HEPs for actions after control room abandonment. The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAIs 07b and 35 summarized above and concludes that the licensee's response is acceptable because the difference in results between the licensee's method and the acceptable method (HRA calculator and guidance from NUREG-1921) for transition is negligible, and the licensee will adopt the acceptable method before post-transition self-approval PCEs.

Execution Dependencies for Local Fire Human Actions. An F&O on the need for the HRA to include dependencies among the HFEs was disposed as closed, but indicated the resolution was pending the next scheduled FPRA update. In PRA RAI 07d dated March 2, 2012 (Reference 11), the NRC staff asked the licensee to confirm that this resolution had indeed been completed. In its response dated April 17, 2012 (Reference 6), the licensee, first noting that this F&O was a suggestion, described how dependencies among HFEs are addressed in the FPRA, clarifying that the F&O was intended as a reminder to check the cutsets for independence among the combinations of execution HFEs. Upon update of the fire response procedures, the licensee plans to re-visit the dependency analysis as part of the NFPA-805 implementation phase. Currently, the Fire HRA considers dependencies between HFEs in two ways, as follows. First, a separate HFE was added to the FPRA to explicitly capture dependencies among those HFEs that were functionally similar. The complementary portion of the operator actions are, by definition, considered independent and have been included explicitly in the FPRA as separate basic events. Second, even though the execution HFEs are modeled as separate basic events, the HEP development for the execution HFEs uses a timeline that takes into account all preceding operator actions in the Fire Response Procedure (whether these operator actions are in the FPRA model or not). The timing analysis, and resulting HEP for each execution HFE, accounts for the other execution HFEs in the same scenario. The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAI 07d summarized above, and concludes that the response is acceptable because the licensee has described in detail an acceptable method to identify and incorporate dependencies between HEPs.

Fire-Induced Instrumentation Failure in HRA. In response to PRA RAI 27 dated February 19, 2013 (Reference 8), the licensee addressed fire-induced instrumentation failure in the HRA in three ways.

1. When there was no reading due to fire impact, any operator action directly requiring the failed instrumentation for diagnosis was not credited in the FPRA (i.e., the HEP was set to 1.0).

2. Where instrumentation required for an operator action was degraded, the HEP was increased to reflect the additional diagnosis effort that may be required using the guidance of NUREG-1921, "EPRI/NRC-RES Fire HRA Guidelines," July 2012 (Reference 31). Typically, this applies to systems where all redundant channels are not showing the same value, requiring additional interpretation from the operators.

3. For off-scale/incorrect/misleading ("spurious") readings, two approaches were taken.

   - First, where operators need to take an action that relies on instrumentation, and the required instrumentation is failed by the fire, the fire HRA quantifies the HEP as described in Item 1. If the operators need to take an action that relies on instrumentation that is degraded by the fire, then the fire HRA quantifies the HEP as described in Item 2.

   - Second, if operators might take actions that are not required that could aggravate the response, such potential cognitive errors of commission were identified by a systematic review of the operating and alarm procedures within the context of the PRA accident scenarios. After the alarm procedure review, the emergency operating procedures were reviewed to identify any instructions that may lead to inappropriate actions given spurious instrumentation failure. At Callaway, all such actions were systematically screened based on diversity of instrumentation or being inconsequential.

The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAI 27 summarized above and concludes that the licensee's response is acceptable because the licensee describes in detail a technically credible means by which to address fire-induced instrument failure in the HRA for new operator actions.

<u>Focused-scope Peer Review of Updates to Internal Events PRA not Included in Fire PRA</u>. In response to PRA RAI 01b dated April 17, 2012 (Reference 6), the licensee provided the F&Os and their dispositions from an August 2011 focused-scope internal events PRA peer review, confirming that those which were not incorporated into the FPRA would have little or no effect. The NRC staff reviewed the licensee's bases for concluding little or no effect and found them acceptable, as summarized in the F&O Table (Reference 67).

<u>PRA Upgrades Requiring Peer Review Since Last Full-Scope Peer Review</u>. The internal events PRA underwent a full-scope review in 2006. The results of that review are summarized in LAR Attachment U. A major revision was subsequently completed in 2009, with changes that were deemed to constitute a "PRA upgrade." The changes were reviewed during a focused-scope peer review in October 2011. In PRA RAI 29 dated December 11, 2012 (Reference 14), the NRC staff requested the licensee identify any changes to the internal events PRA that were deemed upgrades as a result of the 2011 review. In its response dated February 19, 2013 (Reference 8), the licensee indicated that the following items constituted upgrades in the 2009 model and were peer reviewed in October 2011: support system initiating event fault tree models, interfacing systems loss-of-coolant accident (LOCA) modeling, incorporation of the Westinghouse Owner's Group (WOG) 2000 reactor coolant pump (RCP) seal LOCA model, implementation of an expanded CCF methodology, revised LERF model, and revised internal flooding analysis. Additionally, the upgraded HRA methods were addressed during a May 2011 focused-scope peer review. The licensee also stated that no changes were made to the FPRA after the November 2009 full-scope peer review that constituted an "upgrade." The NRC staff reviewed the licensee's description of upgraded methods and confirmation that these methods

were addressed with focused-scope peer reviews, and concludes that the licensee previously identified the upgrades and that the upgrades identified have been adequately addressed.

Timing for Post-Fire Human Failure Events. In PRA RAIs 07c and 36 dated March 2 and June 6, 2012, and July 30, 2013 (References 11, 12, and 15, respectively), the NRC staff questioned the licensee regarding relatively small HEPs for selected rapid human actions. Specifically, three human actions were identified where the time margin for completion of critical tasks was very short (approximately one minute or less). In its responses dated July 12, 2012, and February 19 and August 5, 2013 (References 7, 8, and 9, respectively), the licensee stated that the method used to develop these HEPs was the same applied to all FPRA HEPs and that there were no HRA related peer review comments that remained open. The licensee also provided a sensitivity evaluation where each HEP was assigned a value of 1.0 (totally unsuccessful). The reported increases in CDF, LERF, delta-CDF, and delta-LERF ranged from approximately 3 percent to approximately 15 percent, remaining below the numerical acceptance thresholds in RG 1.174, as cited in RG 1.205. In its response to PRA RAI 36 (Reference 9) the licensee disposed the request for a description of future analyses for these and other HEPs (from PRA RAI 35) by indicating its intent to use alternate analyses using the EPRI HRA calculator in accordance with NUREG-1921 prior to evaluating any post-transition changes under NFPA 805 as implementation item 13-805-001 (See LAR Attachment S, Table S-3). The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAIs 07c, 35, and 36 summarized above, and concludes that the three human actions in question are required after control room abandonment and so will be included in the new HEP calculations. Since the results in the transition risk estimates from these changes will be bounded by the acceptable results from the sensitivity analysis, and the implementation of accepted methods prior to post-transition self-approval, the NRC staff concludes that the licensee's response is acceptable.

Use of Fractional Influence Factors for Transient Fires. A deviation from NUREG/CR-6850 was cited where fractional (<1.0) influence factors were assumed for certain transient fire scenarios. A special weighting factor of 0.05 was used for maintenance in hot work prohibited zones and a 0.1 was used for storage in transient combustible-free zones. Except for the RCP room in containment, the minimum value for occupancy was 1.0, thus the combined weighting factors were always greater than 1.0. In PRA RAI 08 dated March 2 and June 6, 2012 (References 11 and 12, respectively), the NRC staff asked for sensitivity evaluations where the total influence factor had at least weight of 1.0. In its responses dated April 17 and July 12, 2012 (References 6 and 7, respectively), the licensee indicated that these fractional values were always combined with at least a weight of 1.0 for the occupancy influence factor. Therefore, the analyses as performed already constituted the requested sensitivity evaluation. The RCP room was assigned a value of 0.0 for all three factors. Personnel occupancy and maintenance work does not occur in this area during power operation, due to health and safety concerns. The final transient frequency is 0.0, which is inconsistent with the ASME/ANS PRA Standard (Reference 35), but the impact on risk is negligible. The RCP area has a fixed fire frequency of 2.35×10^{-3}/yr, such that adding a transient frequency of 5.8×10^{-4}/yr. (i.e., normal occupancy, storage, and maintenance) would be expected to negligibly impact risk. The NRC staff subsequently asked the licensee in PRA RAI 37 dated July 30, 2012 (Reference 15), if it would update its analysis to align with the accepted resolution in FAQ 12-0064 (Reference 53). In its response dated August 5, 2013 (Reference 9), the licensee responded that it would update its analysis via implementation item 13-805-002 (See LAR Attachment S, Table S-3) before any post-transition self-approval PCEs. The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAIs 08 and 37 summarized above, and concludes that the licensee's response is acceptable because the difference in results between the licensee's method and the acceptable method (FAQ 12-0064) for transition is negligible, and the licensee will adopt the acceptable method before post-transition self-approval PCEs.

<u>Low Ignition Frequency for Bus Duct Fires</u>. Citing plant-specific presence of "considerably fewer iso-phase bus ducts than a typical plant," the licensee reduced the generic bus duct fire frequency by a factor of 5. In PRA RAI 08b dated March 2 and June 6, 2012 (References 11 and 12, respectively), the NRC staff noted this as a deviation from NUREG/CR-6850 and asked the licensee to perform a sensitivity analysis without this reduction. In its response dated July 12, 2012 (Reference 7), the licensee provided the sensitivity analysis which shows roughly 10 percent increases in CDF and LERF for the ignition frequency bin, but less than 1 percent in total fire CDF and LERF. There was no change in the corresponding delta-CDF or delta-LERF values. In response to PRA RAI 38 dated August 5, 2013 (Reference 9), the licensee created implement item 13-805-003 (See LAR Attachment S, Table S-3) to revise its FPRA to use the recommended ignition frequency in NUREG/CR-6850. The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAIs 08b and 38 summarized above, and concludes that the licensee's response is acceptable because the difference in results between the licensee's method and the acceptable method (via NUREG/CR-6850) for transition is negligible, and the licensee will adopt the acceptable method before post-transition self-approval PCEs.

<u>Credit for Control Power Transformers (CPTs) for AC Circuit Failure Probabilities</u>. Based on recent developments from cable fire tests, consensus between the nuclear industry and NRC is that the current credit for reducing "hot short" probabilities when CPTs are present now appears unverifiable. In PRA RAI 09 dated March 2 and June 6, 2012 (References 11 and 12, respectively), the NRC staff asked that a sensitivity analysis be performed without this credit taken (nominally a reduction by a factor of 2). In its response dated July 12, 2012 (Reference 7), the licensee removed the credit, where appropriate, reporting increases in CDF, LERF, delta-CDF, and delta-LERF ranging from approximately 30 percent to nearly 100 percent. Still, in all cases, the theoretical totals after accounting for the increases remain below the numerical acceptance thresholds in RG 1.174, as cited in RG 1.205. In PRA RAI 39 dated July 30, 2013 (Reference 15), the NRC staff further requested that the licensee clarify whether it intended to retain the sensitivity evaluation as its PRA basis. In its response dated August 5, 2013 (Reference 9), the licensee responded that, given the recent guidance in NRC memorandum dated June 14, 2013, from Richard P. Correia, RES, to Joseph G. Giitter, NRR, titled "Interim Technical Guidance on Fire-Induced Circuit Failure Mode Likelihood Analysis" (Reference 68), it would adopt the updated values in the letter prior to evaluating any post-transition changes under NFPA 805 via implementation item 13-805-004 (See LAR Attachment S, Table S-3). The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAIs 09 and 39 summarized above, and concludes that the licensee's response is acceptable because the difference in results between the licensee's method and the acceptable method (Reference 68) for transition is negligible, and the licensee will adopt the acceptable method before post-transition self-approval PCEs.

Fire Growth Time to Peak Heat Release Rate for Trash Fires. FAQ 08-0052 (Reference 43) in Supplement 1 to NUREG/CR-6850 suggests an 8-minute (min) growth time for common trash fires contained within receptacles. In PRA RAI 10 dated March 2 and June 6, 2012 (References 11 and 12, respectively), the NRC staff noted a deviation from NUREG/CR-6850 by the licensee when assuming a growth time of 10 min, thereby asking for the basis for this deviation and a sensitivity analysis using the suggested time of 8 min. In its response dated July 12, 2012 (Reference 7), the licensee cited a re-evaluation of the data supporting the FAQ as the basis for the 10-min growth time, which the NRC staff did not accept as adequate justification for deviation from the FAQ. However, the requested sensitivity evaluation using an 8-min fire growth time as the basis was also provided, indicating only approximately 0.1 percent increase in the CDF for control room fires. In response to PRA RAI 40 dated August 5, 2013 (Reference 9), the licensee indicated that it will revise its FPRA to use the accepted 8-min growth time from NUREG/CR-6850 via implementation item 13-805-005 (See LAR Attachment S, Table S-3). The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAIs 10 and 40 summarized above, and concludes that the licensee's response is acceptable because the difference in results between the licensee's method and the acceptable method (NUREG/CR-6850) for transition is negligible, and the licensee will adopt the acceptable method before post-transition self-approval PCEs.

Uncertainty Analysis for Ignition Frequencies Beyond FAQ 08-0048. While FAQ 08-0048 (Reference 47) in Supplement 1 to NUREG/CR-6850 recommends limited sensitivity analyses for selected ignition frequency bins, which the licensee performed, the PRA standard requires full uncertainty analyses for key assumptions. In PRA RAI 09b dated March 2 and June 6, 2012 (References 11 and 12, respectively), the NRC staff requested a full uncertainty analysis, or surrogate sensitivity analysis, of all ignition frequency bins. In its response dated July 12, 2012 (Reference 7), the licensee performed the latter by applying a multiplication factor that ratioed the 95th percentile frequency to the mean frequency for each bin in Supplement 1 to NUREG/CR-6850. The increase in CDF, LERF, delta-CDF, and delta-LERF for each bin was reported, along with the cumulative effect from all bins. Both individually and cumulatively, the risk metrics remained below the numerical acceptance thresholds in RG 1.174, as cited in RG 1.205. In response to PRA RAI 41 dated August 5, 2013 (Reference 9), the licensee stated that it will implement the uncertainty methodology used in response to PRA RAI 09b to estimate the change in risk associated with post-transition changes to the FPP, with the understanding that the uncertainty methodology can be refined to utilize parametric data evaluations, via implementation item 13-805-006 (See LAR Attachment S, Table S-3). The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAIs 09b and 41 summarized above, and concludes that the licensee's response is acceptable because the difference in results between the licensee's method and the acceptable method (NUREG/CR-6850) will not cause the change in risk from transition to become unacceptably high, and the licensee will implement the uncertainty methodology used in response to PRA RAI 09b to estimate the change in risk associated with post-transition changes to the FPP, with the understanding that the uncertainty methodology can be refined to utilize parametric data evaluations.

Effect of Internal Events PRA Update of Common-Cause Failures (CCFs) on Fire PRA. The PWROG focused-scope internal events peer review suggested that two potential issues with application of the CCF data be addressed. While the licensee addressed these issues for the internal events PRA, the NRC staff requested in PRA RAI 04d dated March 2, 2012 (Reference 11) that at least the effect from those CCFs with increased probabilities be estimated for the FPRA, specifically CDF, LERF, delta-CDF, and delta-LERF. In its response to RAI 04d dated April 17, 2012 (Reference 6), and related material from the responses to PRA RAIs 01c and 33 dated July 12, 2012, and February 19, 2013 (References 7 and 8), the licensee performed a sensitivity evaluation on CDF (CDF is the limiting risk metric at Callaway, because LERF is always lower than 10 percent of CDF) using updated CCF probabilities from the internal events PRA. The sensitivity analysis was conducted in two parts. The first part evaluated the CDF increase for those basic events already modeled in the FPRA. For all the CCF events that have a direct match between the internal events PRA and FPRA, the net change in both CDF and delta-CDF was negative. Total CDF decreased by 1.38×10^{-6}/yr and delta-CDF by 9.19×10^{-8}/yr. The second part was to evaluate the CDF increase for those basic events that are not modeled in the FPRA. The only set of CCF events in the current internal events PRA which are not included in the FPRA are the CCF combinations of the non-safety auxiliary feedwater (AFW) pump and the safety-related motor-driven AFW pumps. For this sensitivity study, a bounding risk approach employed surrogate events (non-safety auxiliary feed water pump test and maintenance events), assuming the non-safety AFW pump is failed (basic event probability is set to 1.0). The increase of 1.24×10^{-5}/yr for the fire CDF results in a total fire CDF of 3.24×10^{-5}/yr. The increase in the variant fire CDF of 9.22×10^{-6}/yr yields a total variant fire CDF of 1.54×10^{-5}/yr. Note that 1.54×10^{-5}/yr is not delta-risk as defined by RG 1.205, but is "variant risk" (i.e., the total risk for all scenarios which include a VFDR). The delta-risk (as defined by RG 1.205) is the difference between the "variant risk equation with all credited RAs" and the "variant risk equation with VFDR cutsets removed." For Callaway, the variant CDF is 6.15×10^{-6}/yr, while the delta-risk as calculated in the FREs is 2.03×10^{-6}/yr, which is approximately 1/3. Therefore, if the variant risk increases to 1.54×10^{-5}/yr, the delta-risk would increase to 5.09×10^{-6} /yr as an estimate of the bounding risk increase. This remains under the RG 1.205 acceptance value of 1×10^{-5}/yr. Based upon the above-described sensitivity analyses, the new internal events PRA CCF probabilities will not change the conclusions made for transition based on the existing FPRA. In response to PRA RAI 42 dated August 5, 2013 (Reference 9), the licensee indicated that it will revise the FPRA to incorporate these CCF changes prior to evaluating any post-transition changes under NFPA 805 via implementation item 13-805-007 (See LAR Attachment S, Table S-3). The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAIs 01c, 04d, 33 and 42 summarized above, and concludes that the licensee's response is acceptable because the difference in results between the licensee's method and the acceptable method (incorporation of all relevant CCF events) for transition will not cause the change in risk from transition to become unacceptably high, and the licensee will adopt the acceptable method before post-transition self-approval PCEs.

The licensee identified resolution of the findings from the FPRA peer review in LAR Attachment V and the results of the NRC staff's review of the disposition of the findings is summarized in the F&O Table (Reference 67). The licensee confirmed that the FPRA peer review was performed consistent with RG 1.200. As a result of this review and the supplemental information provided, the NRC staff concludes that the Callaway FPRA's 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. Upon completion of the modifications to the FPRA identified in the implementation items in LAR Attachment S, Table S-3, the staff concludes that the Callaway fire PRA's quantitative results, supported by any required qualitative evaluations, can be used to demonstrate the change in risk meets or exceeds the change in risk acceptance guidelines for self-approval of FPP changes.

3.4.2.3 Fire Modeling in Support of the Development of the Fire Risk Evaluations (FREs)

The NRC staff performed detailed reviews of the FM used to support the Callaway FRE in order to gain further assurance that the methods and approaches used for the application to transition to NFPA 805 were technically adequate. NFPA 805 has the following requirements that pertain to FM used in support of the development of an FRE:

NFPA 805, Section 2.4.3.3: On Acceptability

> The [probabilistic safety assessment (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 FM (first requirement). The results of the NRC staff's review 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 FREs

FM 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 Callaway FRE. The following algebraic fire models and correlations were used for this purpose:

- Flame Height, Method of Heskestad (Reference 27, Chapter 3)
- Plume Centerline Temperature, Method of Heskestad (Reference 27, Chapter 9)
- Radiant Heat Flux, Point Source Method (Reference 27, Chapter 5)
- Ceiling Jet Temperature, Method of Alpert (Reference 69)

The first three algebraic models are described in NUREG-1805, "Fire Dynamics Tools (FDTS): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program" (Reference 27). Alpert's ceiling jet temperature correlation is described in "EPRI Fire Induced Vulnerability Evaluation Methodology," Revision 1 (FIVE) (Reference 69), and serves as the basis for FDTS 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 28).

The licensee stated that the algebraic fire models and empirical correlations were implemented in a database and workbook referred to as the Fire Modeling Database (FMDB) and Transient Analysis Worksheets (TAWs). The FMDB and TAWs also calculate the plume radius according to Heskestad's correlation described in FIVE. The plume radius was used as the horizontal ZOI where it exceeded the ZOI based on heat flux.

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 FRE 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 described in NUREG-1805, Chapter 2, and implemented in the FMDB and TAWs.

In LAR Section 4.5.1.2, the licensee identified the use of the following additional empirical correlations that are addressed in a NUREG or FIVE, but for which V&V is not addressed in NUREG-1824, Volumes 3 and 4.

- Sprinkler Activation Correlation (Reference 27 Chapter 10)
- Smoke Detection Actuation Correlation, Method of Heskestad and Delichatsios (Reference 27, Chapter 11)
- Corner and Wall Heat Release Rate (Reference 29)
- Correlation for Heat Release Rates of Cables (Reference 25, 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 29)

The licensee used the ZOI approach 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 Callaway-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 HRR from the NUREG/CR-6850 methodology.

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

- Control room abandonment calculations
- Temperature sensitive equipment HGL Study

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

- Temperature sensitive equipment ZOI study
- Plume/HGL interaction study
- HGL temperature calculations in specific areas
- Suppression activation in specific areas

- Smoke detector activation calculations in specific areas, based on the method by Cleary to estimate detector response time (Reference 70)

- Validation and Verification (V&V) of CFAST and FDS is documented in NUREG-1824, Volume 5 and Volume 7 (Reference 28), respectively.

Detailed FM using CFAST and FDS was performed for selected fire scenarios in fire areas. CFAST was used for the temperature sensitive equipment HGL study and the MCR abandonment calculations. FDS was used for the HGL/plume interaction and the temperature sensitive equipment ZOI studies. FDS was also used in the suppression activation analyses in a number of fire areas and the HDPE pipe analysis. The V&V of all empirical correlations and fire models that were used to support the Callaway FRE 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 Callaway Fire PRA

By letter dated March 2, 2012 (Reference 11), the NRC staff submitted RAIs concerning the FM conducted to support the Callaway FRE. By letter dated April 17, 2012 (Reference 6), the licensee provided a partial response to the first round of FM RAIs, and requested clarification of the RAIs that were not responded to. By e-mails dated June 6, 2012 (Reference 12), and June 19, 2012 (Reference 13), the NRC provided the requested clarification. The licensee responded to the remaining first round RAIs by letter dated July 12, 2012 (Reference 7). By letter dated December 11, 2012 (Reference 14), the NRC sent a second set of FM RAIs to the licensee. By letter dated February 19, 2013 (Reference 8), the licensee provided a response to the second round RAIs. The following paragraphs describe selected RAI responses related to the acceptability of the fire models used. Several FM RAIs are not discussed in this SE section. The RAIs not discussed were issued to obtain more details on specific aspects of the FM, and the responses allowed the NRC staff to gain a complete understanding of the FM that was performed in support of the Callaway FPRA.

- The NRC staff issued FM RAI 03(a) dated March 2, 2012 (Reference 11), to ask 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 its response to FM RAI 03(a) dated July 12, 2012 (Reference 7), the licensee explained that the approach for fires involving multiple combustibles was to calculate the HRR of each individual combustible as a function of time, and then use the combined total HRR as the input to the algebraic models. Conservative HRRs were determined from NUREG/CR-6850, Volume 2, and the rules for propagation to cable trays and fire spread rates all followed the FLASH-CAT model found in NUREG/CR-7010. 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 (e.g., at the top of cabinet ignition sources or two feet above the floor for transient ignition sources).

Based on a review of the explanation and justification provided in response to FM RAI 03(a), the NRC staff concludes that the licensee's approach used to establish the algebraic model inputs for fires that involve multiple combustibles is acceptable.

- The NRC staff issued FM RAI 3(d) dated March 2, 2012 (Reference 11), to ask the licensee to justify the use of the FDS suppression activation analysis of fire area C-31 for fire areas A-11 and C-30, even though these three areas are not identical and different ignition source locations and secondary combustibles may need to be considered. The NRC staff also issued FM RAI 3(o) (Reference 11) to ask the licensee to justify the use of the FDS suppression activation analysis of fire area C-21 for fire area C-22, even though the ceiling height of C-22 is approximately half the ceiling height of C-21.

  In its response to FM RAI 03(d) dated April 17, 2012 (Reference 6), the licensee stated that applying the FDS analysis of a single fire compartment to additional fire compartments is justified when the analysis is developed with conservative parameters that bound the results for the other compartments. To determine if an FDS analysis can be applied to another compartment, the licensee established six criteria that need to be met. These criteria pertain to the volume of the compartment, the distance between the detector and the fire, the smallest fire size that could result in the damage state being analyzed, ventilation conditions, sprinkler/detector properties, and compartment boundary materials. Fire compartments A-11, C-30, and C-31 meet these criteria as discussed in detail in the report that describes the FDS suppression activation analyses.

  In its response to FM RAI 03(o) dated April 17, 2012 (Reference 6), the licensee stated that the ceiling height of fire area C-21 in the report that describes the FDS suppression activation analyses is incorrect. The ceiling height that was used in the FDS suppression activation analysis for fire area C-21 is 15 feet, not 25 feet. Furthermore, the licensee demonstrated, based on the aforementioned criteria, that the analysis for C-21 was developed with conservative parameters that bound the results for C-22.

  Based on a review of the explanation and justification provided in response to RAIs 03(d) and 03(o), the NRC staff concludes that the licensee's approach to determine when an FDS analysis for one compartment can be applied to another compartment is acceptable.

- The NRC staff issued FM RAI 03(h) dated March 2 and June 6, 2012 (References 11 and 12, respectively), to ask the licensee to explain how the HRR profiles used in the FDS analyses for fire areas A-11, C-30, and C-31, were conservative for the purposes of damage assessment and sprinkler activation. The NRC staff asked a similar question in FM RAI 03(l) (Reference 11) for fire areas C-21 and C-22, where a HRR of 69 kW was used for the initiating fire in the ZOI calculations versus 45 kW in the FDS suppression activation analysis.

In its response to FM RAI 03(h) dated July 12, 2012 (Reference 7), the licensee stated that a higher HRR will lead to more severe target damage, but will also result in earlier suppression activation. Consequently, a lower HRR was used for the sprinkler activation analyses than for the ZOI calculations in fire areas A-11, C-30, C-31, C-21, and C-22. In its response to FM RAI 03(l) (Reference 7), the licensee explained that the HRR used to estimate suppression activation in the cable spreading rooms (C-21 and C-22) was determined to be 45 kW, based on the fact that (1) a fire generating less than 45 kW will not cause ignition of cable trays and will, therefore, be non-conservative with respect to damage, and (2) selecting a fire larger than 45 kW will cause suppression to activate earlier. A 69 kW transient fire was used to determine the ZOI, resulting in a conservative target damage set.

Based on a review of the responses to RAIs 03(h) and 03(l), the NRC staff concludes that the licensee's approach to determine the HRR of the initiating fire in the FDS suppression activation analyses is acceptable.

- The NRC staff issued FM RAI 03(k) dated March 2, June 6, and June 19, 2012 (References 11, 12, and 13, respectively), to ask the licensee to justify the sprinkler response time index (RTI) value of 130 $(m \cdot s)^{0.5}$ used in the FDS suppression activation analyses of fire areas C-21 and C-22, even though higher values are reported in the literature for the type of sprinklers installed in these areas.

  In its response to FM RAI 03(k) dated July 12, 2012 (Reference 7), the licensee stated that because the actual RTI of the sprinkler heads in fire areas C-21 and C-22 was not known, a generic value of 130 $(m \cdot s)^{0.5}$ for standard response sprinklers, taken from NUREG-1805, Chapter 10, was used. The FDS analysis with this RTI value indicates that suppression will activate before the fourth cable tray in a stack of six ignites. The licensee performed a sensitivity analysis to determine the impact of the RTI value over the range for standard response sprinklers, 80 to 350 $(m \cdot s)^{0.5}$, on the time to suppression activation in areas C-21 and C-22. The sensitivity analysis indicates that in the worst case, i.e., for an RTI of 350 $(m \cdot s)^{0.5}$, activation would be delayed by almost 2 minutes (i.e., after the fourth tray ignites). However, the licensee stated that based on feedback from Factory Mutual Global, it is unlikely that the RTI for sprinkler heads of the design used in C-21 and C-22 is over 225 $(m \cdot s)^{0.5}$. Furthermore, the licensee provided an extensive list of conservative assumptions that were made to account for the uncertainties of the input parameters.

  Based on a review of the response to FM RAI 03(k) and the sensitivity analysis performed by the licensee, the NRC staff concludes that the general conservatism in the FDS suppression activation analysis is unlikely to be offset by the use of a lower-than-actual RTI for the sprinkler heads in the cable spreading rooms. Therefore, the RTI value of 130 $(m \cdot s)^{0.5}$ used in the analysis is acceptable.

- The NRC staff issued FM RAI 03(p) dated March 2 and June 6, 2012 (References 11 and 12, respectively), to ask the licensee to provide justification for the assumption in the control room evacuation study that fires originating in the equipment cabinet area (ECA) will not propagate into the MCR.

  In its response to FM RAI 03(p) dated July 12, 2012 (Reference 7), the licensee identified three potential mechanisms for a fire in the ECA to spread to the MCR: (1) cabinet to cabinet propagation, (2) horizontal propagation via cable trays, and (3) HGL development. The first mechanism was eliminated because the back of the main control board (MCB) is a solid metal wall and the closest cabinet in the ECA is 3 feet from the MCB. The second mechanism was ruled out because the cables in the ECA are IEEE-383 (Reference 71) qualified and the trays that cross over into the MCR originate at a panel that is 20 feet from the MCB and are 9 feet above the panels they pass over. Finally, a HGL development due to a fire originating in the ECA would result in MCR abandonment before it causes damage to the MCB.

  Based on a review of the response to FM RAI 03(p), the NRC staff concludes that the licensee's justification for not considering fires that propagate from the ECA to the MCR in the MCR evacuation study is acceptable.

- The NRC staff issued FM RAI 03.01(a)(ii) dated December 11, 2012 (Reference 14) to ask the licensee to provide justification for postulating a 69 kW transient fire in the FDS high-density polyethylene (HDPE) pipe damage analysis for fire area C-1 instead of the 98^{th} percentile value of 317 kW. Pertaining to the same FDS analysis, the NRC staff also issued FM RAI 03.01(a)(iv) (Reference 14) to ask the licensee to justify why a time to peak heat release rate (HRR) of 8 minutes was used instead of 2 or 0 minutes, as recommended in FAQ-08-0052 for loose trash fires and oil spill fires, respectively.

  In its response to RAIs 03.01(a)(ii) and 03.01(a)(iv) dated February 19, 2013 (Reference 8), the licensee justified postulating a t^2 transient fire that reaches a peak HRR of 69 kW in the FDS HDPE pipe damage analysis for fire area C-1 primarily on the basis of administrative controls that will be put in place so that only small amounts of trash in temporary containers are expected in this area (see implementation item 12-805-004 in LAR Table S-3). Moreover, the licensee stated that large combustible liquid fires are not expected because there is no oil-containing equipment in area C-1.

  Based on a review of the response to RAIs 03.01(a)(ii) and 03.01(a)(iv), the NRC staff concludes the licensee's justification for postulating a t^2 transient fire that reaches a peak HRR of 69 kW in the FDS HDPE pipe damage analysis for fire area C-1 is acceptable.

- The detailed FM 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 FM 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 stated that FM was performed as part of the FRE development (NFPA 805 Section 4.2.4.2). NRC staff issued FM RAI 03.01(c) dated December 11, 2012 (Reference 14), to ask the licensee to (1) confirm that no FM 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 FM in support of the FRE.

  In its response dated February 19, 2013 (Reference 8), the licensee confirmed that no FM 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 a review of the licensee's response to FM RAI 03.01(c), the NRC staff concludes the application of MEFS and LFS in the FM performed at Callaway to be acceptable.

3.4.2.3.3 Conclusion for Section 3.4.2.3

Based on the licensee's description of the Callaway process for performing FM in support of the FRE and the clarifications provided in response to the RAIs, the NRC staff concludes that the licensee's FM approach for meeting the requirements of NFPA 805, Section 2.4.3.3 is acceptable.

3.4.2.4 Conclusions Regarding Fire PRA Quality

The NRC staff concludes that the technical adequacy and quality of the Callaway PRA is sufficient for the FREs that support the proposed license amendment because (1) the PRA models conform to the applicable industry PRA standards for internal events and fires at an appropriate capability category, considering the acceptable disposition of the review findings; (2) the FM used to support the development of the Callaway FPRA has been confirmed as appropriate and acceptable; and (3) the PRA models 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.

However, the self-approval acceptance guidelines are much narrower than the transition acceptance guidelines and, therefore, the NRC staff concludes that the following methods and estimates should be replaced with acceptable methods and estimates before the PRA results are used to support risk-informed self-approval of changes to the FPP:

- Timing for Post-Fire HFEs (Item 13-805-001)
- Fractional Transient Influence Factors (Item 13-805-002)

- Bus Duct Ignition Frequency (Item 13-805-003)
- Removal of CPT Credit (Item 13-805-004)
- Trash Fire Growth Rate (Item 13-805-005)
- Uncertainties on Bin Ignition Frequencies (Item 13-805-006)
- Update of Fire PRA CCFs values (Item 13-805-007)

Each of these FPRA modifications is associated with an implementation item in LAR Attachment S, Table S-3 and will be implemented as part of the License Condition.

In addition, the licensee's PRA satisfies the guidance in RG 1.174, Sections 2.3 and 5, regarding quality of the PRA analysis and quality assurance; RG 1.205 Section 4.3, regarding FPRA; and NUREG-0800, Section 19.2 (Reference 23), regarding the review of risk information used to support permanent plant-specific changes to the licensing basis, which further supports the NRC staff's conclusion that the Callaway PRA is technically adequate and of sufficient quality to allow transition to NFPA 805.

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 the quality of the Callaway PRA is sufficient to support self-approval of future risk-informed changes to the FPP under the NFPA 805 license condition following the implementation of the PRA-related implementation items identified in LAR Attachment S, Table S-3 that will be implemented as part of the License Condition.

3.4.3 Fire Risk Evaluations

The NRC staff reviewed the following information during its evaluation of Callaway's FREs:

- LAR Section 4.5.1, "Fire PRA Development and Assessment"
- LAR Section 4.5.2, "Performance Based Approaches"
- LAR Attachment U, "Internal Events PRA Quality"
- LAR Attachment V, "Fire PRA Quality"
- LAR Attachment W, "Fire PRA Risk Insights"

For those fire areas for which the licensee used a PB 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. Plant configurations that did not meet the deterministic requirements of NFPA 805, Section 4.2.3.1 were considered VFDRs.

After identifying VFDRs, the licensee is required to provide an estimate of the change in risk (CDF and LERF) associated with retaining the VFDR relative to a deterministically-compliant case. In PRA RAI 19 dated March 2 and June 6, 2012 (References 11 and 12, respectively), the NRC staff asked the licensee to describe the calculational technique for the types of VFDRs. The licensee response dated July 12, 2012 (Reference 7), included a description of the steps to identify a VFDR and a categorization of the types of VFDRs. These include cable/equipment damage caused by separation issues, degraded fire wrap, and high density polyethylene piping. The response further explained the quantification process as first calculating the differences between risk estimates (1) and (3), followed then by calculating the difference between (2) and (3), defined as follows: (1) plant configured with no credit for recovery actions related to the VFDR; (2) plant configured with credit for the same recovery actions at their nominal human error probabilities, including fire effects, post-transition; (3) plant configured without the VFDR, i.e., assuming the vulnerable cables cannot fail. If the difference between (1) and (3) indicates that a VFDR is significant enough to require a recovery action, then the difference between (2) and (3) is calculated to determine if the recovery action provides an acceptable risk mitigation strategy. If this second difference (cumulative for all VFDRs in the plant) does not meet RG 1.205 risk criteria, a plant change is required. The NRC staff reviewed the information provided by the licensee in the LAR and in its response to PRA RAI 19 summarized above, and concludes that this process is consistent with the process described in FAQ 08-0054 (Reference 50) and therefore acceptable.

In addition to the above, the licensee identified separation issues in the Callaway's site specific MSO calculation as discussed in Section 3.2.3 of this SE. However, issues related to many separation issues do not constitute VFDRs since (1) the scenario could be mitigated with control room and/or primary control station actions, and (2) actions required to address these separation issues are not considered recovery actions. Accordingly, the NRC staff concludes that the licensee's disposition of the issues related to MSO is acceptable.

3.4.4 Additional Risk Presented by Recovery Actions

The NRC staff reviewed LAR Attachment C, "NEI 04-02 Table B-3 – Transition," Attachment G, "Recovery Actions Transition," and Attachment K, "Existing Licensing Action Transition," during its evaluation of the additional risk presented by the NFPA 805 RAs at Callaway. Section 3.2.4 of this SE describes the identification and evaluation of RAs.

For those fire areas for which the licensee used a PB 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. Plant configurations that did not meet the separation requirements of NFPA 805, Section 4.2.3.1 were considered VFDRs. The licensee evaluated each VFDR for risk impact by comparing it to a hypothetically compliant plant configuration, and the additional risk was summed for each fire area and compared to the acceptance criteria contained in RG 1.174. The process used is the same as described in the previous section, but now limited only to those VFDRs resolved via RAs.

The licensee addressed those fire areas that used a previously approved alternative shutdown strategy utilizing the guidance in RG 1.205 for addressing RAs. This included consideration of

PCS and the definition of RA, as clarified in RG 1.205. Accordingly, any actions required to transfer control to, or operate equipment from, the PCS, while required as part of the RI/PB FPP, were not considered RAs per the RG 1.205 guidance and in accordance with NFPA 805. Conversely, any OMAs required to be performed outside the control room and not at the PCS were considered RAs.

The licensee addressed the additional risk of the RAs associated with an approved alternate shutdown, which takes place in response to loss of habitability of the MCR due to fire effects in that location. This is a two-step process. First, the licensee calculates the frequency of damaging fires affecting critical targets in each MCB panel, using the technique of Appendix L in NUREG/CR-6850. This yields the maximum CCDP. Next the licensee calculates the frequency of abandoning the MCR due to loss of habitability from the fire effects, including heat, smoke and toxic gas, using the CFAST FM code. Credit for suppression is based on FAQ 08-0050 (Reference 48) in Supplement 1 to NUREG/CR-6850. The maximum CCDP from the first step may be used as a conservative estimate of the additional risk of RA, or the credit from the second step that can be taken for recovery using the alternate shutdown panel is taken to reduce the CCDP from the first step. Either method is consistent with the change in risk estimates in FAQ 08-0054 and therefore acceptable.

The additional risk associated with RAs performed as a result of postulated fire damage in the MCR was determined as the sum of the products of the fire ignition frequency, propagation probability, non-suppression probability, evacuation probability, human failure probability to successfully operate the alternate shutdown panel, and conditional CCDP for each MCB panel fire scenario and any postulated fires from transient combustibles. The resulting CDF and LERF, also assumed to be their corresponding, bounding delta values, are 7.8×10^{-7}/yr and 2.1×10^{-8}/yr.

Section 3.5 of this SE discusses and evaluates each individual RA. In addition, the NRC staff reviewed the results of the licensee's calculations associated with the additional risk of RAs, which total 2.2×10^{-6}/yr (delta-CDF) and 4.2×10^{-8}/yr (delta-LERF). The NRC staff concludes that the approaches applied are acceptable because the approach conservatively estimates the risk increases, which remain within the RG 1.174 risk acceptance guidelines of 1×10^{-5}/yr (delta-CDF) and 1×10^{-6}/yr (delta-LERF) for small changes.

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

The licensee did not use any RI or PB alternatives to compliance with NFPA 805, which falls under the requirements of 10 CFR 50.48(c)(4), at Callaway.

3.4.6 Cumulative Risk and Combined Changes

The licensee identified the planned NFPA 805 transition modifications that decrease risk, including those that do not remove a VFDR, and for which the licensee takes credit during the assessment of the cumulative risk impact of the transition to NFPA 805 at Callaway. Modifications that are implemented to remove a VFDR become part of the baseline plant models and, while lowering total risk, do not contribute to change in risk. As summarized in SE Section 2.7.1, LAR Attachment S states that these modifications have been completed.

The licensee credited the risk reductions that will be afforded by these modifications in its evaluation of the total change in risk associated with transition to NFPA 805. The licensee reported in the LAR, as supplemented, the total CDF and total LERF which were estimated by adding the risk assessment results for internal events, fire and seismic plus other external events. The CDF and LERF results are summarized in Table 3.4.6-1.

Table 3.4.6-1: CDF and LERF for Callaway after Transition to NFPA 805

Hazard Group	CDF (/year)	LERF (/year)
Internal Events	2.6×10^{-5}	4.2×10^{-7}
Fires	2.0×10^{-5}	4.0×10^{-7}
Seismic	2.3×10^{-6}	Negligible
TOTAL	4.9×10^{-5}	8.2×10^{-7}

The total CDF after implementation of NFPA 805 remains below 1×10^{-4}/yr, and the total LERF remains below 1×10^{-5}/yr, and, therefore, increases in CDF up to 1×10^{-5}/yr and increases in LERF up to 1×10^{-6}/yr are generally considered acceptable according to the risk acceptance guidelines of RG 1.174.

The licensee also provided the delta-CDF (ΔCDF) and delta-LERF (ΔLERF) estimated for each fire area at Callaway that is not deterministically compliant, in accordance with NFPA 805, Section 4.2.3, "Deterministic Approach." The risk estimates for these fire areas result from the modifications that were implemented as part of the transition to NFPA 805 at Callaway. The ΔCDF and ΔLERF results by fire area are summarized in Table 3.4.6-2. The ΔCDF and ΔLERF results by fire area in Table 3.4.6-2 below may change after completion of implementation item 12-805-005, which verifies the validity of the change in risk calculations.

Table 3.4.6-2: ΔCDF and ΔLERF for Callaway after Transition to NFPA 805

Fire Area	ΔCDF (/year)	ΔLERF (/year)
A-1 (Auxiliary Building General Area)	1.6×10^{-8}	1.3×10^{-10}
A-6 (Auxiliary Building North Stairwell)	2.8×10^{-9}	2.5×10^{-12}
A-8 (Auxiliary Building General Area)	4.5×10^{-9}	2.7×10^{-12}
A-11 (Auxiliary Building Cable Chase A)	2.5×10^{-9}	2.0×10^{-10}
A-13 (Auxiliary Feedwater Pump Room B)	5.2×10^{-8}	4.6×10^{-11}
A-15 (Turbine Driven Auxiliary Feedwater Pump Room)	1.4×10^{-8}	1.3×10^{-11}
A-16 (Auxiliary Building Elevation 2026' General Area)	1.2×10^{-8}	1.1×10^{-10}
A-17 (Electrical Penetration Room B)	3.2×10^{-10}	8.5×10^{-13}
A-18 (Electrical Penetration Room A)	2.2×10^{-8}	4.4×10^{-10}
A-19 (Auxiliary Building General Areas)	2.2×10^{-10}	5.0×10^{-14}

Fire Area	ΔCDF (/year)	ΔLERF (/year)
A-21 (Control Room AC and Filtration Unit B)	$2.7x10^{-8}$	$2.8x10^{-10}$
A-22 (Control Room AC and Filtration Unit A)	$1.7x10^{-8}$	$2.8x10^{-11}$
A-23 (Main Steam and Feedwater Valve Compartment)	$9.1x10^{-10}$	$6.1x10^{-13}$
A-24 (Containment Mechanical Penetration Room A)	$8.9x10^{-9}$	$1.7x10^{-12}$
A-27 (Reactor Trip Switchgear Room)	$2.0x10^{-8}$	$3.2x10^{-11}$
A-28 (Auxiliary Shutdown Panel Section A)	$8.3x10^{-9}$	$2.0x10^{-10}$
A-29 (Auxiliary Feedwater Valve Compartment, SG A&D)	$1.3x10^{-8}$	$1.1x10^{-11}$
A-30 (Auxiliary Feedwater Valve Compartment, SG B&C)	0	0
A-33 (Auxiliary Shutdown Panel Section B)	$3.9x10^{-9}$	$3.4x10^{-12}$
C-1 (Pipe Space and Tank Area)	$7.0x10^{-12}$	$6.8x10^{-14}$
C-7 (Control Building North Cable Chase)	$2.7x10^{-9}$	$2.2x10^{-11}$
C-9 (Switchgear Room A)	$1.1x10^{-7}$	$2.5x10^{-9}$
C-10 (Switchgear Room b)	$2.9x10^{-7}$	$4.5x10^{-9}$
C-11 (Control Building Cable Chase B)	$3.1x10^{-10}$	$3.9x10^{-12}$
C-12 (Control Building Cable Chase A)	$1.0x10^{-8}$	$3.0x10^{-10}$
C-15 (Battery and Switchboard Room)	$1.8x10^{-10}$	$6.9x10^{-13}$
C-16 (Battery and Switchboard Room)	$2.2x10^{-10}$	$7.0x10^{-13}$
C-17 (Control Building Cable Chase B)	$2.1x10^{-9}$	$4.6x10^{-11}$
C-18 (Control Building Cable Chase A)	$9.6x10^{-8}$	$2.7x10^{-9}$
C-20 (Control Building Cable Chase B at Column C-6)	$6.0x10^{-10}$	$5.3x10^{-13}$
C-21 (Lower Cable Spreading Room)	$9.7x10^{-9}$	$1.4x10^{-9}$
C-22 (Upper Cable Spreading Room)	$2.0x10^{-7}$	$3.8x10^{-9}$
C-23 (Control Building Cable Chase B)	$7.6x10^{-8}$	$2.0x10^{-9}$
C-24 (Control Building Cable Chase A)	$4.9x10^{-8}$	$1.1x10^{-9}$
C-25 (Control Building Cable Chase B at Column C-6)	$6.0x10^{-10}$	$5.3x10^{-13}$
C-26 (Control Building Cable Chase A at Column C-3)	$1.4x10^{-11}$	$4.9x10^{-15}$
C-27 (Main Control Room)*	$7.8x10^{-7}$	$2.1x10^{-8}$
C-30 (Control Building Cable Chase B)	$1.9x10^{-8}$	$6.7x10^{-11}$
C-31 (Control Building Cable Chase A)	$1.3x10^{-10}$	$9.0x10^{-13}$
C-32 (Control Building Cable Chase B at Column C-6)	$6.0x10^{-10}$	$5.3x10^{-13}$
C-33 (Control Building Cable Chase B)	$7.6x10^{-9}$	$7.2x10^{-11}$
C-35 (Control Building Corridor)	Epsilon (ε)	ε
C-36 (Control Building Cable Chase B at Column C-6)	$6.0x10^{-10}$	$5.3x10^{-13}$
FB-1 (Fuel Handling Building)	$4.6x10^{-8}$	$8.0x10^{-12}$
RB-1 (Reactor Building General Area)	$2.4x10^{-7}$	$1.9x10^{-9}$
TB-1 (Turbine Building General Area)	0	0

Fire Area	ΔCDF (/year)	ΔLERF (/year)
YD-1 (Plant Yard Area Elevation 2000')	$1.7x10^{-8}$	$2.9x10^{-12}$
TOTAL	**$2.2x10^{-6}$**	**$4.2x10^{-8}$**

* For conservatism, total risk is reported for all control room abandonment scenarios instead of the change in risk.

Each of the individual fire area changes in risk for CDF and LERF fall into Region III (very small change) of the RG 1.174 acceptance guidelines. The risk associated with control room abandonment for Fire Area C-27 is reported as $7.8x10^{-7}$/yr (CDF) and $2.1x10^{-8}$/yr (LERF), and still falls within Region III (very small change).

The licensee performed a series of individual sensitivities for the following deviations from NUREG/CR-6850:

Timing for Post-Fire HFEs
Fractional Transient Influence Factors
Bus Duct Ignition Frequency
Removal of CPT Credit
Trash Fire Growth Rate
Uncertainties on Bin Ignition Frequencies
Internal Events Update for CCFs
Spurious PORV Opening without Closing Block Valve
Use of CCDP > 0.1 for MCR Abandonment

Of these, the ones most likely to affect basic events within the same cut sets are removal of CPT credit and uncertainties on bin ignition frequencies. The remainder would generally be independent and not have compounding effects within the same cut set. The removal of CPT credit increases risk and delta-risk measures from about 30 percent to nearly 100 percent. The uncertainties on bin ignition frequencies have a collective effect of about 200 percent overall as the maximum on any one risk or delta-risk measure. If these two effects compounded in every cut set, the net maximum increase in the risk or delta-risk measure would be by a factor of about 5 (applying the near 100 percent increase at the upper end of the range for the removal of CPT credit). Even with this conservative estimate, only the delta-CDF would rise above the Region II-to-I transition threshold of $1.0x10^{-5}$ /yr (CDF) from RG 1.174, and then only slightly (from $2.2x10^{-6}$/yr to $2.2x10^{-6}$ + [$2.2x10^{-6}$x 5] = $1.3x10^{-5}$/yr). The delta-LERF would remain below the transition threshold of $1.0x10^{-6}$/yr (at $4.2x10^{-8}$/yr + [$4.2x10^{-8}$/yr x 5] = $2.5x10^{-7}$/yr). The NRC staff performed a sensitivity analysis and concludes that the delta-risk results from an integrated sensitivity analysis would remain within acceptable limits because use of a conservative 100 percent increase yields acceptable results.

Based on the results of the licensee's fire risk assessments, as summarized above, the risk increase for each fire area associated with transition to NFPA 805 at Callaway, as well as the cumulative change in risk for all fire areas subject to a PB approach, is within the RG 1.174 risk acceptance guidelines of $1x10^{-5}$/yr ΔCDF and $1x10^{-6}$/yr ΔLERF for small changes and the total CDF will remain below $1x10^{-4}$/yr and total LERF will remain below $1x10^{-5}$/yr. In addition, the licensee has included implementation item 12-805-005, which states that once all proposed modifications have been completed, the licensee will verify the change in risk results, and if RG 1.205 acceptance guidelines are not met, additional analytic efforts and/or procedural or plant changes will be implemented to assure that the acceptance guidelines are met.

Based on a review of the licensee's risk evaluation and its results and the NRC staff's supplemental sensitivity evaluation, the NRC staff concludes that the change in risk associated with the proposed alternative to compliance with the deterministic criteria of NFPA 805 is acceptable in accordance with NFPA 805, Section 2.4.4.1 because the change in risk has satisfied RG 1.174, Sections 2.4 and 2.5, and NUREG-0800, Section 19.2.

3.4.7 Uncertainty and Sensitivity Analyses

For the most part, the licensee employed accepted methods to perform the risk analyses which support its LAR to transition to NFPA 805, following the guidance in NUREG/CR-6850. Where deviations were employed, the licensee either clarified the assumptions used and/or performed additional sensitivity analyses to confirm minimal effect. These issues are discussed in Section 3.4.2.2 of this SE.

The licensee performed two additional sensitivity analyses in support of the LAR: (1) Sensitivity with respect to use of the FAQ 08-0048 (Reference 47) fire ignition frequencies as directed in Supplement 1 to NUREG/CR-6850; and (2) Sensitivity with respect to unknown cable types for cables in the turbine building. The first sensitivity study was performed in accordance with the NRC accepted guidance in FAQ 08-0048. The second study demonstrated that, even if the majority of cable types in the Turbine Building were of a thermoplastic nature, the risk and delta-risk results would be such as to satisfy the transition requirements per RG 1.174, via RG 1.205. The licensee also provided a qualitative discussion of uncertainty with respect to the various FPRA tasks from NUREG/CR-6850.

After reviewing this material, the NRC staff issued several RAIs requiring additional clarification and/or sensitivity analyses, primarily for "deviations" that, if shown to affect the results, would become "key assumptions" to the analysis. All issues for which an uncertainty evaluation was performed that lead to changes in the FPRA are discussed in Section 3.4.1. Several issues for which an uncertainty evaluation was performed and that do not lead to changes in the FPRA are discussed below.

Longer than Expected Time Available to Isolate Reactor Coolant System (RCS) Injection. The licensee credits approximately 36 min as available to isolate the RCS injection flow to avoid PORV challenge on pressurizer overfill. This differs from the Callaway FSAR, Section 15.5.1.2, that states the pressurizer becomes water solid following a spurious Safety Injection signal within 9 min, even if the operator terminates normal charging pump flow at 6 min. In PRA RAI 12 dated March 2 and June 6, 2012 (References 11 and 12, respectively), the NRC staff asked the basis for the cited plant-specific calculation to justify the 36-min time frame. In its response dated July 12, 2012 (Reference 7), the licensee described the scenario in detail, providing the results of the Modular Accident Analysis Program (MAAP) analysis which yields 36 min. This was further compared to the RETRAN analysis used for the FSAR estimate of approximately

9 min. The licensee noted that the MAAP analysis, appropriate for PRA, involves best estimates, whereas the RETRAN analysis involves the much more conservative design basis. The key driver among the different parameter assumptions yielding the large difference in available time is the nominal flow rate into the RCS. In RETRAN, this is conservatively assumed to be 346 gallons per minute (gpm) for 6 min, followed by 299 gpm afterward. In MAAP, a more realistic 126 gpm flow rate is assumed throughout. In addition, the licensee performed a sensitivity analysis where no credit for an operator RA was taken, indicating increases in CDF, LERF, delta-CDF and delta-LERF of <10%. Subsequently, the NRC staff asked PRA RAI 43 (Reference 15) requesting further clarification from the licensee. In its response dated August 5, 2013 (Reference 9), the licensee clarified that its FSAR analysis employed conservative values for key parameters to obtain a "worst case" scenario, specifying these "worst case" assumptions. In particular, "maximum" pump curves conservatively increase flow rates, resulting in the "worst case" timing of 9 min from the FSAR. The licensee contends that such results are unrealistic and cites the MAAP results as appropriate for the PRA. The NRC staff reviewed the information provided by the licensee in the LAR and its response to PRA RAIs 12 and 43 and concludes that the licensee's continued use of the 36-min credit is acceptable because MAAP may be used for PRA calculations when "worst case" analyses are deemed overly conservative.

<u>Single-fire Spurious Opening of PORV with Failure to Close Block Valve</u>. Certain single-fire scenarios could result in spurious opening of a PORV with loss of power to close its associated block valve. Recovery requires local operator action. In PRA RAI 13 dated March 2 and June 6, 2012 (References 11 and 12, respectively), the NRC staff asked the licensee to describe the scenarios in greater detail, including the effects of failure to perform the operator action. In its response dated July 12, 2012 (Reference 7), the licensee described the scenarios in greater detail, noting where the operator recovery action was credited (six of 16 fire areas). The licensee stated that the frequency of fires which caused these failures is relatively low, walkdowns were used to verify that a fire free path exists from the control room to the battery rooms for all fire areas where the action is credited, and that MAAP analysis was performed to determine the timing available before core damage. In addition, a sensitivity analysis was performed removing the potential for the single-fire common failure of the PORV and its block valve. The results showed a potential decrease in fire CDF of <1 percent which, if treated as the delta-CDF, would result in a decrease in delta-CDF of <10 percent. Results for LERF and delta-LERF were similar. An additional bounding sensitivity analysis where the operator action was assumed to always fail showed increases in CDF, LERF, delta-CDF, and delta-LERF no greater than approximately 50 percent. The NRC staff reviewed the information provided by the licensee in the LAR and its response to PRA RAI 13 and concludes that that the evaluation described is consistent with the modeling supporting requirements of the ASME/ANS PRA standard and that removal of the RAs would not be consistent with the intent that realistic evaluation are performed and, therefore, the evaluation is acceptable.

In addition to the sensitivity studies, the licensee developed and applied a systematic approach to search for other key assumptions and sources of uncertainty that could potentially impact the risk analyses related to NFPA 805. A five-step process was used:

1. Identify uncertainties associated with each fire PRA task
2. Develop strategies for addressing the uncertainties
3. Perform review of uncertainties to make decisions
4. Perform sensitivity and uncertainty analyses
5. Document uncertainty and sensitivity results

The licensee process concluded that the assumptions are conservative or realistic when justified by plant-specific configurations and available data. The NRC staff reviewed the licensee's examination of their key assumptions and concludes that the licensee's risk evaluations are reasonable and conservative, and not significantly impacted by the specific modeling assumptions made by the licensee.

3.4.8 Conclusion for Section 3.4

Based on the information provided by the licensee in the LAR, as supplemented, regarding the fire risk assessment methods, tools, and assumptions used to support transition to NFPA 805 at Callaway, the NRC staff concludes the following:

- The licensee's PRA used to perform the risk assessments in accordance with NFPA 805, Section 2.4.4 (PCEs) and Section 4.2.4.2 (FREs), is of sufficient quality to develop risk results that, supplemented by the sensitivity studies, support the application to transition the Callaway FPP to NFPA 805 as proposed in the LAR. In addition, the analyses, assumptions, and approximations used to map the cause-effect relationship associated with the application are technically adequate. Therefore, the NRC staff concludes the PRA approach, methods, tools and data are acceptable in accordance with NFPA 805 Section 2.4.3.3

- The self-approval acceptance guidelines are much narrower than the transition acceptance guidelines and may be applied to all future propose changes to the FPP and, therefore, the NRC staff concludes that several methods and estimates should be replaced with acceptable methods and estimates before the PRA results are used to support risk-informed self-approval of changes to the FPP. All these estimates and method changes are included as implementation items in LAR Table S-3and must be completed prior to the use of the FPRA for post-transition changes as part of completing the License Condition.

- The transition process included a detailed review of fire protection DID and SM as required by NFPA 805. The NRC staff concludes the licensee's evaluation of DID and SM is acceptable. The licensee's process followed the NRC-endorsed guidance in NEI 04-02, and is consistent with the approved NRC staff guidance in RG 1.205, which provides an acceptable approach for meeting the requirements of 10 CFR 50.48(c).

- The changes in risk (i.e., ΔCDF and ΔLERF) associated with the proposed alternatives to compliance with the deterministic criteria of NFPA-805 (FREs) are acceptable and that the licensee has satisfied the guidance 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. By meeting the guidance contained in these approved regulatory documents, the changes in risk have been found to be acceptable to the NRC staff, and therefore meet the requirements of NFPA 805.

- The risk presented by the use of these RAs was determined and provided in accordance with the guidance in RG 1.205 and NFPA 805, Section 4.2.4.

- The licensee did not use any RI or PB alternatives to compliance with NFPA 805 which fall under the requirements of 10 CFR 50.48(c)(4).

- The licensee's application to transition to NFPA 805 is a combined change, as defined by RG 1.205, which includes risk increases identified in the FREs with risk decreases resulting from modifications that include reductions in risk associated with the internal events PRA. Based on the combination of these risk values, the changes associated with NFPA 805 meet the guidance contained in RG 1.205, Regulatory Position 3.2.5, related to meeting the requirements for cumulative risk and combined plant changes.

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 an 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 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 nuclear safety performance criteria 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 use 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 LAR 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 S, "Plant Modifications and Items to be Completed During Implementation" and Attachment W, "Fire PRA Insights," in its evaluation of the ability of each fire area to meet the NSPC of NFPA 805.

Callaway is divided into 81 fire areas. Based on the information provided by the licensee in the LAR, as supplemented, the licensee performed the NSCA on a fire area basis for each of the 81 fire areas. LAR Attachment C provides the results of these analyses on a fire area basis.

Table 3.5-1 of this SE identifies those fire areas that were analyzed using either the deterministic or PB approach in accordance with NFPA 805 Chapter 4 based on the information provided in LAR Attachment C, Table B-3, "Fire Area Transition."

Table 3.5-1 Fire Area and Compliance Strategy Summary

Fire Area	Area Description	NFPA 805 Compliance Basis
A-1	Auxiliary Building – El. 1974, 1988	Performance-Based
A-2	Auxiliary Building Safety-Related Pump Area	Deterministic
A-3	Boric Acid Tank Rooms	Deterministic
A-4	Auxiliary Building Safety-Related Pump Area	Deterministic
A-5	Auxiliary Building Stairway and Elevator (south)	Deterministic
A-6	Auxiliary Building Stairway (North)	Performance-Based
A-7	Boron Injection Room	Deterministic
A-8	Auxiliary Building – El. 2000, General Area	Performance-Based
A-9	RHR [Residual Heat Removal] Heat Exchanger Room	Deterministic
A-10	RHR Heat Exchanger Room	Deterministic
A-11	Cable Chase, Auxiliary Building – El. 2000	Performance-Based
A-12	Auxiliary Building Cable Chase B, Auxiliary Building – El. 2000	Deterministic
A-13	Auxiliary Feedwater Pump Room B	Performance-Based
A-14	Auxiliary Feedwater Pump Room A	Deterministic
A-15	Turbine Driven Auxiliary Feedwater Pump Room	Performance-Based
A-16	Auxiliary Building El. 2026, General Area	Performance-Based
A-17	Electrical Penetration Room B	Performance-Based
A-18	Electrical Penetration Room A	Performance-Based
A-19	Auxiliary Building El. 2047, General Area	Performance-Based
A-20	Personnel Hatch and CCW Surge Tank Area	Deterministic
A-21	Control Room AC and Filtration Unit B	Performance-Based
A-22	Control Room AC and Filtration Unit A	Performance-Based
A-23	Main Steam and Feedwater Valve Compartment	Performance-Based
A-24	Containment Mechanical Piping Penetration Room A	Performance-Based

Fire Area	Area Description	NFPA 805 Compliance Basis
A-25	Pipe Penetration Room B	Deterministic
A-26	Ops Storage/I&C Hot Shop	Deterministic
A-27	Reactor Trip Switchgear Room	Performance-Based
A-28	Auxiliary Shutdown Panel Section A	Performance-Based
A-29	Auxiliary Feedwater Valve Compartment, SG A&D	Performance-Based
A-30	Auxiliary Feedwater Valve Compartment, SG B&C	Performance-Based
A-33	Auxiliary Shutdown Panel Section B	Performance-Based
AB-1	Auxiliary Boiler Room	Deterministic
C-1	Pipe Space and Tank Area, Control Building, El. 1974	Performance-Based
C-2	Control Building North Cable chase, Control Building, El. 1974	Deterministic
C-3	Control Building Cable Chase B, Control Building, El. 1974	Deterministic
C-5	Control Building Access Control Area, Control Building, El. 1984	Deterministic
C-6	Control Building Access Control Area, Control Building, El. 1984	Deterministic
C-7	Control Building North Cable Chase, Control Building, El. 1984	Performance-Based
C-8	Control Building Cable chase B, Control Building, El. 1984	Deterministic
C-9	ESF Switchgear Room A	Performance-Based
C-10	ESF Switchgear Room B	Performance-Based
C-11	Control Building Cable Chase B, Control Building, El. 2000	Performance-Based
C-12	Control Building Cable Chase A, Control Building, El. 2000	Performance-Based
C-13	Class 1E Train B AC Equipment Room	Deterministic
C-14	Class 1E Train A AC Equipment Room	Deterministic
C-15	Battery and Switchboard Room B, Control Building, El. 2016	Performance-Based
C-16	Battery and Switchboard Room A, Control Building, El. 2016	Performance-Based
C-17	Control Building Cable Chase B, Control Building, El. 2016	Performance-Based
C-18	Control Building Cable Chase A, Control Building, El. 2016	Performance-Based
C-19	Control Building Cable Chase A at Column C-3, Control Building, El. 2016	Deterministic
C-20	Control Building Cable Chase B at Column C-6, Control Building, El. 2016	Performance-Based
C-21	Lower Cable Spreading Room	Performance-Based
C-22	Upper Cable Spreading Room	Performance-Based
C-23	Control Building Cable Chase B, Control Building, El. 2032	Performance-Based
C-24	Control Building Cable Chase A, Control Building, El. 2032	Performance-Based

Fire Area	Area Description	NFPA 805 Compliance Basis
C-25	Control Building Cable Chase B at Column C-6, Control Building, El. 2032	Performance-Based
C-26	Control Building Cable Chase A at Column C-3, Control Building, El. 2032	Performance-Based
C-27	Control Room Area	Performance-Based
C-28	Control Room Service Area	Deterministic
C-29	SAS Room, Control Building, El. 2047	Deterministic
C-30	Control Building Cable Chase B, Control Building, El. 2047	Performance-Based
C-31	Control Building Cable Chase A, Control Building, El. 2047	Performance-Based
C-32	Control Building Cable Chase B at Column C-6, Control Building, El. 2047	Performance-Based
C-33	Control Building Cable Chase B, Control Building, El. 2073-6	Performance-Based
C-34	Control Building Cable Chase B at Column C-6, Control Building, El. 2073-6	Deterministic
C-35	Control Building Corridor, Control Building, El. 2016	Performance-Based
C-36	Control Building Cable Chase B at Column C-6, Control Building, El. 2000	Performance-Based
C-37	Control Building Cable Chase A, Control Building, El. 2000	Deterministic
D-1	Diesel Generator A, Diesel Generator Building, El. 2000'	Deterministic
D-2	Diesel Generator B, Diesel Generator Building, El. 2000'	Deterministic
FB-1	Fuel Handling Building	Performance-Based
LDF-1	Laundry Decontamination Facility	Deterministic
RB-1	Reactor Building	Performance-Based*
RSB-1	RAM Storage Building	Deterministic
RW-1	Radwaste Building	Deterministic
TB-1	Turbine Building	Performance-Based
UNCT	Ultimate Heat Sink North Cooling Tower	Deterministic
UNPH	Essential Service Water Pump Room A	Deterministic
USCT	Ultimate Heat Sink South Cooling Tower	Deterministic
USPH	Essential Service Water Pump Room B	Deterministic
YD-1	Yard Area	Performance-Based

* The FRE for Fire Area RB-1 has applied a bounding risk estimate; assuming delta risk is equivalent to the total risk of the area.

LAR Attachment C provides the results of these analyses on a fire area basis. For each fire area, 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)
- A high level discussion of the SSCs required in order to meet the NSPC
- 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 disposition of each VFDR using either modifications (completed or committed) or the performance of an FRE in accordance with NFPA 805, Section 4.2.4.2.

3.5.1.1 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/systems addressed in these sections 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, 20-foot (ft) separation zones, radiant energy shields, and ERFBS 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 Callaway, and identifies if the fire suppression and detection systems, 20-ft separation zones, radiant energy shields, and ERFBS installed in these areas are required to meet criteria for separation, DID, risk, licensing actions, or EEEEs.

The NRC staff reviewed LAR Attachment C for each fire area to ensure fire detection and suppression met the principles of DID in regard to the planned transition to NFPA 805 at Callaway.

Based on the statements provided in LAR Attachment C, and the NRC staff's review, the NRC staff concludes that the licensee has adequately identified the fire detection and suppression systems required to meet the NSPC of NFPA 805 on a fire area basis.

3.5.1.2 Evaluation of Fire Suppression Effects on NSPC

Each fire area of LAR Attachment C includes a discussion of how the licensee met the requirement to evaluate the fire suppression effects on the ability to meet the NSPC.

The licensee stated in LAR Attachment C that damage to plant areas and equipment from the accumulation of water discharged from manual and automatic fire protection systems is minimized by the provision of floor drainage systems or through open doors. In addition, safety-related electrical cable trays are qualified for water exposure, and safety-related electrical motors are on pedestals and are designed and sealed to be water-resistant. Therefore, fire suppression activities will not adversely affect achievement of the NSPC.

Based on the information provided by the licensee in LAR Attachment C, as supplemented, 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.

3.5.1.3 Licensing Actions

Based on the information provided in LAR Attachment C, the licensee identified deviations from the deterministic requirements for each fire area that were previously approved by the NRC and will be transitioned with the NFPA 805 FPP. Disposition of deviations at Callaway followed two different paths during transition to NFPA 805:

- The deviation was found to be unnecessary due to completion of a modification which removed the deviation.
- The deviation 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 engineering evaluations that form the safety basis for approval of these previously approved deviations are being used as qualitative engineering evaluations to demonstrate compliance with the deterministic requirements of NFPA 805. Each of these deviations is summarized in LAR Attachment C on a fire area basis and described in further detail in LAR Attachment K, "Existing Licensing Action Transition." The licensee proposed several clarifications to the previously approved licensing actions and documented these clarifications in LAR Attachment T, "Clarification of Prior NRC Approvals." The elements of the pre-transition FPP licensing basis for which specific NRC previous approval needed clarification were included in LAR Attachment T. Sufficient details were included to demonstrate how those elements of the pre-transition FPP licensing basis met the requirements in 10 CFR 50.48(c)

(RG 1.205, Regulatory Position 2.2.1) or were evaluated using qualitative engineering methods and previously found to be acceptable to the NRC staff.

The licensee used the process described in NEI 04-02, which requires a determination of the basis of acceptability and a determination that the basis of the acceptability is still valid for the licensing actions that will be transitioned. The licensing actions being transitioned, including the clarifications, are summarized in Table 3.5-2.

Table 3.5-2 Callaway Previously Approved Licensing Actions Being Transitioned

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
Licensing Action Number (LAN) 001 - There is no deviation associated with the requirements of Appendix A of BTP ASB 9.5-1; however the manual suppression system is a modified automatic system, which is in fact a deviation from NFPA 13 requirements. The fixed, manually charged, closed head sprinkler system is provided over the cable trays in Zones RB3 and RB4. To protect the chloride sensitive piping and equipment from fire protection system leakage, the standpipes inside the reactor building are normally dry. Control room operator action is required to charge the standpipes.	RB-1 – Reactor Building	Prior Approval Clarification Request (PACR) 3 - Since approval, several containment fire protection features have been modified as follows: 1. Fire hose is not installed at each hose outlet in the reactor building during power operations. Hoses are staged at the Reactor Building personnel hatch for fire brigade use as needed. 2. Portable fire extinguishers are not installed inside the reactor building during power operations. The extinguishers are installed as soon as practical following entering MODE 5, when descending in power, and removed as late as practical in MODE 5, when ascending in power. 3. Locking open the emergency release valves to fire protection deluge	Based on the previous staff approval of the engineering justification for the use of a manually charged, closed head sprinkler system in Zones RB3 and RB4, 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. In addition, since the reactor building is not normally manned during power operations, the NRC staff concludes that staging the hoses and fire extinguishers outside the entrance to the Reactor Building and locking open the emergency release valves is an acceptable alternative to the original configuration since the hoses and extinguishers are available at the entrance for fire brigade use on the way into containment. Locking open the emergency release valves eliminates

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
		valves KCXV0261 and KCXV0262. As part of this LAR submittal and transition to NFPA 805, it is requested that the NRC formally document as a "prior approval" the manual fire protection system in lieu of an automatic system in the reactor building to meet the deterministic requirements of NFPA 805 Section 4.2.3.4 (c). The minor configuration changes to the reactor building fire protection features were implemented within the guidelines of the fire protection license condition and do not adversely affect the overall level of fire protection in the reactor building. Based on the above, the NRC staff concludes that the deviation, as clarified, being carried forward is acceptable.	the potential for spurious failure of the system due to hot shorts. Since this is a manual pre-action system that is actuated by opening a motor operated containment isolation valve, the NRC staff concludes that locking open the emergency release valves will have no adverse impact on system operations.
LAN 002 - Unrated watertight doors with gasketing materials, credited to maintain the 3-hour fire rating of barriers in which they are installed, deviates from the guidance of Section D.1.j of Appendix A to BTP ASB 9.5-1 and Section C.5 of BTP CMEB 9.5-1.	A-1 - Auxiliary Building - El. 1974, 1988 A-2 – Auxiliary Building Safety-Related Pump Area A-4 – Auxiliary Building Safety-Related Pump Area A-7 – Boron Injection Room A-13 – Auxiliary Feedwater Pump Room B A-14 – Auxiliary Feedwater Pump Room A	None	Based on the previous staff approval of the engineering justification for this deviation 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.

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
	A-15 – Turbine Driven Auxiliary Feedwater Pump Room A-29 – Auxiliary Feedwater Valve Compartment, SG A&D A-30 – Auxiliary Feedwater Valve Compartment, SG B&C C-1 – Pipe Space and Tank Area, Control Building, El. 1974		
LAN 003 - Elevator and dumbwaiter doors rated at 1-1/2 hours, credited to maintain the 3-hour rating of barriers in which they are installed, deviates from the guidance of Section D.1.j of Appendix A to BTP ASB 9.5-1 and Section C.5 of BTP CMEB 9.5-1.	A-1 – Auxiliary Building - El. 1974, 1988 A-5 – Auxiliary Building Stairway and Elevator (south) A-8 – Auxiliary Building - El. 2000, General Area A-16 – Auxiliary Building El. 2026, General Area A-20 – Personnel Hatch and CCW Surge Tank Area	None	Based on the previous staff approval of the engineering justification for this deviation 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.
LAN 004 - Unrated missile doors, credited to maintain the 3-hour rating of barriers in which they are installed, deviates from the guidance of Section D.1.j of Appendix A to BTP ASB 9.5-1 and Section C.5 of BTP CMEB 9.5-1. Six of the doors are single swing and four are double swing doors. Each leaf of the 10 doors is of similar construction which includes a 2 1/2-inch thick steel plate front and vertical and horizontal reinforcing beams which form a boxed-in area near	A-5 – Auxiliary Building Stairway and Elevator (south) A-8 – Auxiliary Building - El. 2000, General Area A-16 – Auxiliary Building El. 2026, General Area A-19 – Auxiliary Building El. 2047, General Area A-27 –Reactor Trip Switchgear Room C-9 – Switchgear Room A C-21 – Lower Cable	None	Based on the previous staff approval of the engineering justification for this deviation 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

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
the perimeter of the door. The multiple point latching mechanisms pass through the reinforcing beams and fix the doors in the opening.	Spreading Room C-22 –Upper Cable Spreading Room C-27 –Control Room Area C-35 –Control Building Corridor, Control Building, El. 2016 LDF-1 – Laundry Decontamination Facility RSB-1 –RAM Storage Building TB-1 – Turbine Building YD-1 – Yard Area		
LAN 006 - The SNUPPS penetration seal design, credited to provide an effective 3-hour fire barrier although the seals do not meet the specific ASTM E-119 temperature rise limitation of 325^{0}F above ambient on the unexposed side, deviates from the guidance of Section C.5 of BTP CMEB 9.5-1.	All	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 007 - The SNUPPS fire detection power supplies design deviates from the guidance of Section C.6.a of BTP CMEB 9.5-1 and from the recommended design of NFPA 72D requiring that the backup DC system consists of 4 hour rated batteries located at the local panels. The remote fire protection panels are powered by a non-Class 1E 125V dc system. The non-Class 1E 125V dc	All	None	Based on the previous staff approval of the engineering justification for this deviation 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

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
system is supplied through batteries and battery chargers. Two physically independent offsite power sources provide the normal and preferred source to this system. A standby power source is provided by the station emergency diesel generators. In the event of a battery charger failure, each battery can carry the dc loads for approximately 6 hours. This assumes that ac sources are still available for other non 1E loads. A continuous fire watch would be established by plant technical specification in the event of loss of power to the remote panels.			
LAN 008 - Partial sprinkler systems are provided for the corridor area around the Component Cooling Water (CCW) pumps; however, there is an area between the pumps that does not have sprinkler protection and that contains intervening combustibles (BOP cable trays), not in accordance with the guidance of Section C.5.b of BTP CMEB 9.5-1. Fire stops installed in these intervening cable trays are credited to prevent a fire from spreading to the redundant CCW system pumps.	A-16 – Auxiliary Building El. 2026, General Area	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 009 - The diesel fuel oil day tank dike is located below the day tank and does not extend above the bottom of the tank in	D-1 – Diesel Generator A, Diesel Generator Building, El. 2000' D-2 – Diesel Generator B,	PACR 4 - Subsequent to NRC approval of deviation request dated 02/01/1984, it was determined that the	The original NRC approval was granted based on the overall design of the fire protection features in the

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
accordance with the guidance of Section C.7.i of BTP CMEB 9.5-1. Therefore, leakage from a tank may escape from the confines of the dike and spread into its respective diesel room. The diked area has a free volume of greater than 110% of the tank volume and is provided with a floor drain which drains to a sump within the room. The sump is provided with a solid cover plate and Class 1E level indication in the control room.	Diesel Generator Building, El. 2000'	actual capacity of the diesel day tank dike was less than 100%. As part of this LAR submittal and transition to NFPA 805, it is requested that the NRC formally document as "prior approval" the current design configuration of the two diesel generator day tanks based on the overall design of the fire protection features in the rooms, as well as the gravity fed system design, the drainage system, and the availability of tank level indications. The NRC staff reviewed LAN 009 and PACR 4 and had a concern regarding the conflicting information on the dike capacity and the basis for the prior NRC approval as provided in the LAR. In the response to Callaway SSA RAI 05 (Reference 6), the licensee clarified that the day tank dike has a capacity of 97% of the day tank volume, and the dike area has a floor drain with a covered 900-gallon floor sump designed for combustible liquids. The fuel oil system is a seismic Category I, gravity-fed system. The unpressurized day tanks are vented to the outdoors via piping equipped with flame	rooms and did not specifically rely on the dike capacity. Therefore, although the day tank dike capacity is less than originally stated, the gravity fed system design; drainage and level indication are still applicable. The reduction in dike capacity is not considered to affect the overall performance of the configuration in the event of a leak. Based on the previous staff approval of the engineering justification for this deviation 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

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
		arrestors; therefore, no pressurized sprays will occur as a result of a leak. The day tanks are also equipped with level indication that alarms in the Control Room if there are more than 3 gallons of leakage. Based on the described fuel oil day tanks design features and the existing fire protection features in the Diesel Generator Buildings, the NRC staff concludes that the deviation, as clarified, being carried forward is acceptable.	
LAN 010 – The RCP oil collection system deviates from the guidance of Section C.7.a of BTP CMEB 9.5-1 as being not specifically designed to maintain its integrity following a safe shutdown earthquake event.	RB-1 – Reactor Building	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 011 – Non-rated equipment hatchways with steel plate covers, credited to provide protection equivalent to a 3-hour rated fire barrier, deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-1 – Auxiliary Building – El. 1974, 1988 A-8 – Auxiliary Building – El. 2000, General Area A-16 – Auxiliary Building – El. 2026, General Area A-19 – Auxiliary Building – El. 2047, General Area A-20 – Personnel Hatch and CCW Surge Tank Area	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 012 – Non-rated electrical penetrations in	A-17 – Electrical Penetration Room B	None	Based on the previous staff approval of the

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
the reactor containment wall, credited to provide protection equivalent to a 3-hour rated fire barrier, deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-18 –Electrical Penetration Room A RB-1 – Reactor Building		engineering justification for this deviation 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
LAN 013 – Non-rated mechanical penetrations (process and sampling lines and containment purge penetrations) in the reactor containment wall, credited to provide protection equivalent to a 3-hour rated fire barrier, deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-19 – Auxiliary Building – El. 2047, General Area A-20 – Personnel Hatch and CCW Surge Tank Area A-23 – Main Steam and Feedwater Valve Compartment A-24 – Containment Mechanical Piping Penetration Room A A-25 – Pipe Penetration Room B RB-1 – Reactor Building	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 014 – Nonrated fuel transfer tube connecting reactor containment and the fuel building, credited to provide protection equivalent to a 3-hour rated fire barrier, deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	FB-1 – Fuel Handling Building RB-1 – Reactor Building	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 015 – Nonrated personnel hatch connecting reactor containment and Fire Area A-20, and the hatchways	A-20 – Personnel Hatch and CCW Surge Tank Area RB-1 – Reactor Building	PACR 2 - The personnel hatch to Fire Area A-20 was previously approved; however, the containment emergency	The NRC staff reviewed LAN 015 and PACR 2 and had a concern regarding ignition sources or combustible

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
to YD-1, credited to provide a level of safety equivalent to 3-hour rated barriers, deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	YD-1 – Yard Area	personnel and equipment hatchways to the yard, Fire Area YD-1, were not specifically identified in the approved deviation request dated 03/14/1984. The emergency personnel hatchway is of identical construction to the personnel hatch to Fire Area A-20. The equipment hatch, while not identical, is equally robustly constructed, consisting of a welded steel assembly with a double gasketed, flanged, and bolted cover and provided with a moveable missile shield on the outside of the Reactor Building. As part of this LAR submittal and approval it is requested that the NRC formally document as "prior approval" the Emergency Personnel Hatch and the Equipment Hatch in the Reactor Building Containment walls.	loading in the vicinity of the subject Emergency Personnel Hatch and Equipment Hatch that may challenge the non-rated penetrations, as well as any configuration changes to the fire area/zone since previous approval. In the response to Callaway SSA RAI 04 (Reference 6), the licensee clarified that there are no significant ignition sources or combustible loading in the vicinity that can challenge the non-rated penetrations. Additionally, there have been no significant changes to the areas surrounding these penetrations since the original NRC approval. Based on the above, the NRC staff concludes that the deviation, as clarified, being carried forward is acceptable.
LAN 016 - Lack of full-area detection coverage (specifically no detection in rooms 1103, 1104, 1105, 1106, 1123, 1124, 1125, 1129, 1202, 1203, 1204, and 1329) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-1 – Auxiliary Building – El. 1974, 1988	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 017 - Lack of full-area	FB-1 – Fuel Handling	None	Based on the previous

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
suppression coverage (specifically partial suppression in room 6101 and no suppression in rooms 6102, 6103, 6106, 6201, 6204, 6205, 6210, 6301, and 6302) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	Building		staff approval of the engineering justification for this deviation 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
LAN 018 - Lack of full-area detection coverage (specifically no detection in rooms 3201, 3206, and 3210) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-6 – Control Building Access Control Area, Control Building, El. 1984	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 019 - lack of full-area detection coverage (specifically no detection in rooms 3213, 3214, 3217, 3221, 3224, and 3236) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-5 – Control Building Access Control Area, Control Building, El. 1984	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 020 - Lack of full-area detection coverage (specifically no detection in room 3227) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	TB-1 – Turbine Building	None	Based on the previous staff approval of the engineering justification for this deviation and the statement by the licensee that the basis remains valid, the NRC staff concludes that the

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
			underlying condition allowed by this licensing action is acceptable as a performance-based qualitative engineering analysis
LAN 021 - Lack of full-area detection coverage (specifically no detection in room 3104) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-1 – Pipe Space and Tank Area, Control Building, EL. 1974	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 022 - Lack of full-area detection coverage (specifically no detection in rooms 1302, 1306, 1307, 1308, 1313, 1318, and 1319) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-8 – Auxiliary Building, El. 2000, General Area	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 023 - Lack of full-area detection coverage (specifically no detection in room 1309) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-9 – RHR Heat Exchanger Room	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 024 - Lack of full-area	A-10 – RHR Heat	None	Based on the previous

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
detection coverage (specifically no detection in room 1310) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	Exchanger Room		staff approval of the engineering justification for this deviation 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
LAN 025 - Lack of full-area detection coverage (specifically no detection in rooms 1324 and 1327) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-29 – Auxiliary Feedwater Valve Compartment, SG A&D	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 026 - Lack of full-area detection coverage (specifically no detection in rooms 1328 and 1330) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-30 – Auxiliary Feedwater Valve Compartment, SG B&C	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 027 - Lack of full-area detection coverage (specifically no detection in rooms 3401 and 3412) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-35 – Control Building Corridor, Control Building, El. 2016	None	Based on the previous staff approval of the engineering justification for this deviation and the statement by the licensee that the basis remains valid, the NRC staff concludes that the

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
			underlying condition allowed by this licensing action is acceptable as a performance-based qualitative engineering analysis
LAN 028 - Lack of full-area detection coverage (specifically no detection in rooms 6101, 6201, and 6210) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	FB-1 – Fuel Handling Building	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 029 - Lack of full-area detection coverage (specifically no detection in rooms 1502, 1503, and 1601) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-20 – Personnel Hatch and CCW Surge Tank Area	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 030 - Lack of full-area detection coverage (specifically no detection in room 3604) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-27 – Control Room Area	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 031 – Lack of full-	C-28 – Control Room	None	Based on the previous

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
area detection coverage, (specifically no detection in room 3607) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	Service Area		staff approval of the engineering justification for this deviation 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
LAN 032 – Lack of full-area detection coverage (specifically no detection) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-36 – Control Building Cable Chase B at column C-6, Control Building, El. 2000	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 033 – Lack of full-area detection coverage (specifically no detection) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-37 – Control Building Cable Chase A, Control Building, El. 2000	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 034 – Lack of full-area suppression coverage (specifically partial suppression in room 1101 and no suppression in 1102, 1103, 1104, 1105, 1106, 1115, 1120, 1121,	A-1 – Auxiliary Building – El. 1974, 1988	None	Based on the previous staff approval of the engineering justification for this deviation and the statement by the licensee that the basis remains valid, the NRC staff concludes that the

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
1122, 1123, 1124, 1125, 1128, 1129, 1130, 1201, 1202, 1203, 1204, 1205, and 1329) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.			underlying condition allowed by this licensing action is acceptable as a performance-based qualitative engineering analysis
LAN 035 – Lack of full-area suppression coverage (specifically partial suppression in rooms 1301, 1312, 1316 and 1317 and no suppression in 1302, 1306, 1307, 1308, 1311, 1313, 1314, 1315, 1318, 1319, 1320 and 1321) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-8 – Auxiliary Building – El. 2000, General Area	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 036 – Lack of full-area suppression coverage (specifically partial suppression in room 1331) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-15 – Turbine Driven Auxiliary Feedwater Pump Room	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 037 – Lack of full-area suppression coverage (specifically partial suppression in room 1401 and no suppression in 1402, 1406, and 1408) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	A-16 – Auxiliary Building El. 2026, General Area	None	Based on the previous staff approval of the engineering justification for this deviation 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

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
LAN 038 – Lack of full-area suppression coverage (specifically partial suppression in room 3104) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-1 – Pipe Space and Tank Area, Control Building, El. 1974	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 039 – Lack of full-area suppression coverage (specifically partial suppression in rooms 3213, 3217, 3219, 3220, and 3224) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-5 – Control Building Access Control Area, Control Building, El. 1984	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 040 – Lack of full-area suppression coverage (specifically partial suppression in rooms 3201, 3202, 3205, 3206, 3210 and 3234) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-6 – Control Building Access Control Area, Control Building, El. 1984	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 041 – Lack of full-area suppression coverage (specifically partial suppression in room 3501) deviates from the guidance of Section C.5.b of BTP	C-21 – Lower Cable Spreading Room	None	Based on the previous staff approval of the engineering justification for this deviation and the statement by the licensee that the basis remains valid, the NRC staff

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
CMEB 9.5-1.			concludes that the underlying condition allowed by this licensing action is acceptable as a performance-based qualitative engineering analysis
LAN 042 – Lack of full-area suppression coverage (specifically partial suppression in room 3801) deviates from the guidance of Section C.5.b of BTP CMEB 9.5-1.	C-22 – Upper Cable Spreading Room	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 043 – Lack of fireproofing on Fuel Building roof deviates from the guidance of Section D.1.j of Appendix A to BTP ASB 9.5-1.	FB-1– Fuel Handling Building	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 044 - Non-rated heavy steel cover plate on the trench connecting the fuel building and radwaste tunnel deviates from the guidance of Section D.1.j of Appendix A to BTP ASB 9.5-1.	FB-1 – Fuel Handling Building RW-1 – Radwaste Building	None	Based on the previous staff approval of the engineering justification for this deviation 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

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
LAN 045 - Vent opening in the fire barrier separating the two compartments of Fire Area A-23 deviates from the guidance of Section D.1.j of Appendix A to BTP ASB 9.5-1.	A-23 – Main Steam and Feedwater Valve Compartment	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 046 - Insufficient separation between the Load Shed Emergency Load Sequencer (LSELS) panels, where the redundant panels are located in the same area of the Control Room and their output relays are mounted back-to-back in a common panel, deviates from the guidance of Section C.5.b of BTP CMEB 9 5-1.	C-27 – Control Room Area	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 047 - Lack of low level detectors in the Control Room deviates from Section C.7.a of BTP CMEB 9.5-1.	C-27 – Control Room Area	None	Based on the previous staff approval of the engineering justification for this deviation 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
LAN 048 - Lack of smoke detectors in all Control Room cabinets and consoles containing redundant equipment deviates from Section C.7.a of BTP	C-27 – Control Room Area	None	Based on the previous staff approval of the engineering justification for this deviation and the statement by the licensee that the basis remains valid, the NRC staff

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
CMEB 9.5-1.			concludes that the underlying condition allowed by this licensing action is acceptable as a performance-based qualitative engineering analysis
LAN 049 – Design of the Auxiliary Shutdown Panel and the procedures to achieve and maintain post-fire safe shutdown following Control Room evacuation in accordance with Section C.5.c of BTP CMEB 9 5-1.	C-27 – Control Room Area	PACR 1 - As part of this LAR submittal and transition to NFPA 805, the licensee requested that the NRC formally document as "prior approval" the original design of the ASP and its physical capabilities only. The NSCA has been performed under the transition to NFPA 805 and is submitted for NRC approval. The NRC staff reviewed LAN 049 and PACR 1 and had a concern regarding the current licensing basis which allowed for "cutting a control power cable at the equipment to ensure that a fault in the Control Room does not prevent certain equipment operation." In the response to Callaway SSA RAI 03 (Reference 6), the licensee clarified that the above operations are no longer required, and there are no NFPA 805 recovery actions that require cutting of control power cable. Plant modifications to provide for the capability to isolate and transfer control of fire affected	Modifications performed as part of the NFPA 805 transition address the circuit issues that previously required cutting of control power cables to support establishing control at the ASP. The remainder of the design and operation of the ASP was previously reviewed and approved by the NRC staff. Based on the previous staff approval of the engineering justification for this deviation, the modification to address circuit issues described above, 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

Licensing Action Description	Applicable Fire Areas	Clarification [as applicable]	NRC Staff Evaluation
		components to the local control station, with redundant fusing, are included in the Attachment S of the LAR.	

The NRC staff reviewed the deviations from the pre-NFPA 805 licensing basis identified in Table 3.5-2, including the description of the previously approved deviations from the deterministic requirements, the basis for and continuing validity of the deviation, and the NRC staff's original evaluation or basis for approval of the deviation. The licensee stated in LAR Section 4.2.3, that 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, except as identified in LAR Attachment T and further described in SE Section 3.5.2.

Based on the NRC staff's review of the Callaway licensing actions identified and described in LAR Attachment C, Attachment K, and the clarifications in Attachment T, 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 to carry forward the engineering evaluations, which requires a determination of the basis of acceptability and a determination that the basis is still valid. Based on the previous NRC staff approval of the deviations 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, as clarified, being carried forward supporting the NFPA 805 transition, as identified in Table 3.5-2, are acceptable.

3.5.1.4 Existing Engineering Equivalency Evaluations (EEEEs)

The EEEEs that support compliance with NFPA 805 Chapter 4 were reviewed by the licensee using the methodology contained in NEI 04-02. The methodology for performing the EEEE review included the following determinations:

- The EEEE is not based solely on quantitative risk evaluations,
- The EEEE is an appropriate use of an engineering equivalency evaluation,
- The EEEE is of appropriate quality,
- The standard license condition is met,
- The EEEE is technically adequate,
- The EEEE reflects the plant as-built condition, and
- The basis for acceptability of the EEEE remains valid.

In accordance with the guidance in RG 1.205, Regulatory Position 2.3.2, as clarified by

FAQ 08-0054 (Reference 50), EEEEs that demonstrate that 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 LAR 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, the guidance of RG 1.205, and FAQ 08-0054, and, therefore, is acceptable.

3.5.1.5 Variances from Deterministic Requirements

For those fire areas where deterministic criteria were not met, VFDRs were identified and evaluated using PB methods. VFDR identification, characterization, and resolutions were identified and summarized in LAR Attachment C for each fire area. Documented variances were all represented as separation issues. The following strategies were used by the licensee in resolving the VFDRs:

- An FRE determined that applicable risk, DID, and SM criteria were satisfied without further action.

- An FRE determined that applicable risk, DID, and SM criteria were satisfied with a credited RA

- An FRE determined that applicable risk, DID, and SM criteria were satisfied with a DID action

- An FRE determined that applicable risk, DID, and SM criteria were satisfied with a plant modification(s), as identified in LAR Attachment C, as well as Attachment S, Table S-1 "Completed Modifications," and Table S-2 "Committed Modifications."

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. 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 adequately addressed all separation issues identified in the LAR.

3.5.1.6 Recovery Actions

LAR Attachment G lists the RAs identified in the resolution of VFDRs in LAR Attachment C for each fire area. The RAs identified by the licensee include actions considered necessary to meet risk acceptance criteria and actions relied upon as DID (see SE Section 3.5.1.7 below).

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 details of the NRC staff's review for establishing RAs are described in SE Section 3.2.5 "Establishing Recovery Actions." The NRC staff's evaluation of the additional risk of RAs credited to meet the risk acceptance guidelines is provided in SE Section 3.4.4, "Additional Risk Presented by Recovery Actions."

3.5.1.7 Recovery Actions Credited for Defense in Depth

The licensee specified RAs to enhance DID in Fire Area C-27, Control Room Area. These DID RAs are not credited in the risk determination for the fire area but are credited in the FREs. The nuclear safety and radioactive release performance goals, objectives, and criteria of NFPA 805 were met without these actions. These RAs are required for DID and therefore are part of the RI/PB FPP, which necessitates that these actions be subject to a PCE if subsequently modified or removed.

3.5.1.8 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 associated elements used to form fire area boundaries and barriers separating success path 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.

Fire area boundaries are established for those areas described in LAR Attachment C, as modified by applicable EEEEs that determine the barriers are adequate for the hazard or otherwise disposition differences in barrier design and performance from applicable criteria. The acceptability of fire barriers and separations is also evaluated as part of the NRC staff's review of LAR Attachment A, Table B-1 process and as such are addressed in SE Section 3.1.

3.5.1.9 Electrical Raceway Fire Barrier Systems

The purpose of NFPA 805 Chapter 4 is to determine those fire protection systems and features that are required to meet the NSPC. LAR Table 4-3 and LAR Attachment C identify the fire areas that credit ERFBS as a fire protection feature required to meet the NSPC. The licensee analyzed these fire areas using the PB approach in accordance with NFPA 805 Section 4.2.4. The licensee used ERFBS installations that fully meet the deterministic fire exposure duration requirements as simplifying assumptions in accordance with NFPA 805 Section 4.2.2. Each fire area utilizing degraded ERFBS, as identified in LAR Attachment C, included a discussion of any

VFDR analysis used to evaluate the acceptability of this feature. The NRC staff reviewed the information provided in the LAR and concludes that the licensee's application properly documented each instance of ERFBS use required to meet the NSPC and the acceptability of the installation using the PB approach in accordance with NFPA 805 Section 4.2.4.

ERFBS installations required by NFPA 805 Chapter 4 must meet the fundamental fire protection requirements of NFPA 805 Chapter 3. The NRC staff's review of ERFBS installations against the requirements of NFPA 805 Chapter 3 is provided in section 3.1.1.2 of this SE.

3.5.1.10 Issue Resolution

In reviewing the NSCA, the NRC staff asked SSA RAI 2 dated March 2, 2012 (Reference 11), regarding the assumption that instrument air will be lost during post-fire conditions, particularly how the loss of instrument air is incorporated into the initial and resulting position of components for circuit analysis. In addition, the RAI questioned whether instrument air may be required for operation of post-fire safe shutdown equipment but is not captured in the NSCA. In its response dated April 17, 2012 (Reference 6), the licensee clarified that the instrument air system is not credited or analyzed in the NSCA and NPO analyses. These analyses, however, conservatively assumed that instrument air exists if it can result in an adverse consequence and is lost if it can provide a beneficial effect. Based on the conservative assumptions used in the analysis, the NRC staff concludes that the licensee adequately assessed the potential impact of the loss of instrument air on nuclear safety and, therefore, the clarification is acceptable.

Several SSA RAIs are not discussed in this SE section. These RAIs were issued to obtain more details on specific aspects of the NSCA, and the responses allowed the NRC staff to gain a complete understanding of the NSCA that was performed in support of the Callaway FPRA.

3.5.1.11 Conclusion for Section 3.5.1

As documented in LAR Attachment C for those fire areas that used a deterministic approach in accordance with NFPA 805, Section 4.2.3, the NRC staff concludes that each of the fire areas analyzed using the deterministic approach meets the associated criteria of NFPA 805, Section 4.2.3.2. This conclusion is based on: (1) the licensee's documented compliance with NFPA 805, Section 4.2.3.2; (2) the licensee's assertion that the success path will be free of fire damage without reliance on RAs; (3) an assessment that the suppression systems in the fire area will have no impact on the ability to meet the NSPC; and (4) the licensee's appropriate determination of the automatic 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, the NRC staff concludes that, based on the licensee's LAR, as supplemented, each fire area has been properly analyzed, and compliance with the NFPA 805 requirements demonstrated as follows:

- Deviations from the pre-NFPA 805 fire protection licensing basis that were transitioned to the NFPA 805 licensing basis were reviewed for applicability, as well as continued validity, and found acceptable.

- VFDRs were evaluated and either found to be acceptable based on an integrated assessment of risk, DID, and SM, or modifications or RAs were identified and actions planned or implemented to address the issue.

- 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. The licensee's analysis appropriately identified the fire protection SSCs required to meet the NSPC, including fire suppression and detection systems, 20-ft separation zones, radiant energy shields, and ERFBS.

- ERFBS credited were documented on a fire area basis, verified to be installed consistent with tested configurations and rated accordingly, and evaluated using an FRE that demonstrated the ability to meet the applicable acceptance criteria for risk, DID, and SM.

Accordingly, each fire area utilizing the PB approach was able to achieve and maintain the NSPC, and the associated FREs meet the applicable NFPA 805 requirements for risk, DID, and SM.

3.5.2 Fire Protection during Non-Power Operational Modes (NPO) 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 NPOs. The licensee stated that it used the process provided in NEI 04-02, as supplemented by FAQ 07-0040 (Reference 46), for demonstrating that the NSPC are met for higher risk evolutions (HREs) during NPO modes. FAQ 07-0040 clarified the guidance from NEI 04-02 with regard to providing "reasonable assurance that a fire during non-power operations will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition." Specifically, FAQ 07-0040 clarifies the following:

- The process for selecting equipment and cabling to evaluate for NPO modes.

- Evaluation of HREs during NPO modes.

- The process for analyzing key safety functions (KSFs) in different plant operational states (POSs).

- The actions taken beyond the normal fire protection program DID actions when a specific KSF could be lost as a direct result of fire damage

3.5.2.1 NPO Strategy and Plant Operational States (POSs)

In LAR Section 4.3, the licensee stated that the process used to demonstrate that the NSPC are met during NPO modes is consistent with the processes outlined in NEI 04-02 and FAQ 07-0040. The licensee's strategy for control and protection of equipment during NPO modes includes:

- Review the existing Outage Management Processes.

- Identify Equipment/Cables

    - Review plant systems to determine success paths that support each of the DID KSFs

    - Identify cables required for the selected components and determine their routing.

- Perform Fire Area Assessments (identify pinch points - plant locations where a single fire may damage all success paths of a KSF).

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

3.5.2.2 NPO Analysis Process

The licensee stated that its goal is to ensure that contingency plans are established when the plant is in an HRE and it is possible to lose a KSF due to fire. The controls and measures that are evaluated during NPOs includes time to boil, RCS and fuel pool inventory, decay heat removal capability, and activities that may impact KSFs. LAR Section 4.3 discusses these additional controls and measures. However, during low risk periods, normal risk management controls, as well as fire prevention and protection processes and procedures, will be used at Callaway.

The outage management procedure identifies special requirements for reduced inventory and mid-loop conditions. These conditions are also consistent with FAQ 07-0040, guidance which considers them to be generally the period of highest risk. The identification of systems and components to be included in the NPO review begins with the identification of the POSs that need to be considered. The licensee identified three POSs, consistent with FAQ 07-0040. The following KSFs are then evaluated against the above POSs for inclusion into the NPO transition review:

- Reactivity
- Core Decay Heat Removal
- Containment
- Inventory
- Support Systems / Functions
    - Component Cooling Water
    - Essential Service Water
    - Power Availability (with offsite power)
    - Power Availability (with onsite power)
- Spent Fuel Pool Decay Heat Removal

Equipment and cables are then selected based on the systems identified for meeting each of the above KSFs and the applicable plant operating procedures associated with the NPO modes. The various modes of operation for each system used to satisfy each KSF is reviewed, and a comprehensive list of equipment cables is developed. Where applicable, the NPO selected equipment's functional requirement is reviewed against the functions previously analyzed for the at-power analysis; and cable selection performed as necessary. The equipment and cables were logically tied and related to the applicable KSF success paths. Power supplies and other supporting components such as interlocks were also identified, listed, and tied with their component and KSF success paths. The selected components were flagged as NPO to allow for "pinch point" analysis by fire area. Pinch points refer to a particular location in an area where the damage from a single fire scenario could result in failure of multiple components or trains of a system such that the maximum detriment on that system's performance would be realized from the single fire scenario. Typically, this involves close vertical proximity of cables which support redundant components or trains of a system such that all such cables can be damaged by just one fire scenario.

3.5.2.3 NPO Key Safety Functions and SSCs Used to Achieve Performance

The licensee stated in LAR Attachment D, "Non-Power Operational Modes Transition," that its risk management procedure defines the KSFs, the success paths to achieve the KSFs, and the components required for the success paths. The NPO analysis is contained within Callaway's NSCA, which includes the NPO equipment and cable selection process and results, the NPO cable failure analysis, the NPO KSF pinch point analysis process and results by fire area, the NPO risk reduction actions to be completed, and the evaluation and definition of the POSs that are considered HREs.

Based on its review of the information provided in the LAR, the NRC staff concludes that the licensee used methods consistent with the guidance provided in NEI 04-02, as supplemented by FAQ 07-0040, and RG 1.205 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. These implementation tasks are reflected in LAR Attachment D and Attachment S.

3.5.2.4 NPO Pinch Point Resolutions and Program Implementation

In LAR Attachment D, "Non-Power Operational Modes Transition," the licensee stated that it identified the components needed to support the NPO KSF that were not included in the nuclear safety equipment list and required additional circuit analysis. For those components, which had not been previously analyzed in support of the at-power analysis, or whose functional requirements may have been different for the NPO analysis, the licensee performed cable selection in accordance with approved project procedures. Cables necessary to support the selected function of a component were selected and analyzed for fire impact. In accordance with FAQ 07-0040, any area experiencing fire damage which eliminates all success paths for a KSF (without RAs outside the MCR) is considered a pinch point. The licensee did not use FM to eliminate any fire area from being a pinch point.

During those NPO evolutions where risk is relatively low, the licensee credited the normal FPP DID actions including control of ignition sources, control of combustibles, and compensatory actions for fire protection system impairments, for addressing the risk impact of those fires that potentially impact one or more trains of equipment that provide a KSF required during NPO. During those NPO evolutions that are defined as HREs, additional fire protection DID measures are implemented to manage risks in fire areas that contain known pinch points or where pinch points may arise because of equipment taken out of service. Depending on the significance of the potential damage, the licensee stated that any one or combination of the follow options to reduce the NPO fire risk can be applied:

- Prohibition or limitation of hot work in fire areas during periods of increased vulnerability.

- Verification of operable detection and/or suppression in the vulnerable areas.

- Prohibition or limitation of combustible materials in fire areas during periods of increased vulnerability.

- Plant configuration changes (e.g., removing power from equipment once it is placed in its desired position).

- Provision of additional fire patrols at periodic intervals or other appropriate compensatory measures (such as surveillance cameras) during increased vulnerability.

- Use of RAs to mitigate potential losses of KSFs.

- Identification and monitoring in-situ ignition sources for "fire precursors" (e.g., equipment temperatures).

- Reschedule the work to a period with lower risk or higher DID.

The NPO analysis does not credit the use of RAs to mitigate potential losses of KSFs.

NFPA 805 requires that the NSPC be met during any operational mode or condition, including NPO. As described above, the licensee performed the following engineering analyses to demonstrate that it meets this requirement:

- Identified the KSFs required to support the NSPC during NPO.

- Identified the POSs where further analysis is necessary during NPO.

- 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 were 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.

The NRC staff concludes that the licensee has provided reasonable assurance that the NSPC are met during NPO modes and HREs, based on the engineering analyses described in the LAR, as supplemented, which:

- Identified the POSs to be analyzed.

- Identified plant evolutions considered to be HREs.

- Identified the KSFs to be met during NPO modes.

- Identified the SSCs required to meet the KSFs.

- Identified the location of the SSCs and their associated cables and power supplies.

- Performed an analysis to determine the "pinch points" where a fire could cause the loss of one or more KSFs.

- Identified mitigating actions that could be taken to reduce fire risk during HREs.

3.5.3 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 Callaway.

For those fire areas that used a deterministic approach, the NRC staff verified the following:

- Deviations from the existing Callaway 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.

- The required automatic fire suppression and automatic fire detection systems were appropriately documented for each fire area.

Accordingly, the NRC staff concludes 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 verified the following:

- Deviations from the existing 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 VFDRs were evaluated using the FRE PB method (in accordance with NFPA 805, Section 4.2.4.2) to address risk impact, DID, and SM, and 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 found to be acceptable in accordance with NFPA 805, Section 4.2.4.

- The required automatic fire suppression and automatic fire detection systems were appropriately documented for each fire area.

Accordingly, 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, the associated FREs meet the requirements for risk, DID, and SM.

The NRC staff's review of the licensee's analysis and outage management process during NPO modes found that the licensee provided reasonable assurance that the NSPC will be met during NPO modes and HREs.

3.6 Radioactive Release Performance Criteria

3.6.1 Method of Review

NFPA 805, Chapter 1 defines the radioactive release goals, objectives, and performance criteria that must be met by the fire protection program in the event of a fire at a nuclear power plant in any plant operational mode. NFPA 805, Section 1.3.2, "Radioactive Release Goal," states that:

> 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.

NFPA 805, Section 1.4.2, "Radioactive Release Objective," states that:

> Either of the following objectives shall be met during all operational modes and plant configurations.
>
> (1) Containment integrity is capable of being maintained [such that fire-fighting products are monitored and released within the plant's normal effluents program)].
>
> (2) The source term is capable of being limited [such that any unmonitored releases would not exceed the performance criteria].

NFPA 805, Section 1.5.2, "Radioactive Release Performance Criteria," states that:

> Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, Limits.

The licensee assessed its current FPP using the methodology contained in NEI 04-02 and FAQ 09-0056 (Reference 51). The NRC reviewed whether the LAR provides an acceptable transition of the Callaway FPP licensing basis to meet the radioactive release performance criteria requirements of an RI/PB FPP, in accordance with 10 CFR 50.48(a) and (c) using the guidance in RG 1.205 and NUREG-0800, Section 9.5.1.2 (Reference 21). The NRC staff performed an audit of the licensee's evaluation to determine whether the Callaway FPP is capable of meeting the NFPA radioactive release goals, objectives, and performance criteria. The results of the NRC staff audit and evaluation are provided below.

3.6.2 Scope of Review

The licensee's evaluation of the capability of the Callaway FPP to meet the goals, objectives, and performance criteria of NFPA 805 was performed for all plant operating modes (including power and NPO) and for all plant areas. The licensee's review found that the fire suppression activities, as defined in the pre-fire plans and fire brigade firefighting instruction operating guidelines, were written and valid for any plant operating mode. The scope of the licensee's assessment was determined by the NRC staff to be adequate because the review included all modes of plant operation and all plant areas.

3.6.3 Identification of Plant Areas Containing Radioactive Materials and Providing Containment during Fire Fighting Operations

The licensee performed a screening of plant fire areas to determine where there was a potential for generating radioactive effluents during firefighting operations. The fire areas where there was no possibility of radioactive materials being present were identified (e.g., those outside of the radiological controlled area). These fire areas were eliminated from further review.

For all other fire areas where radioactive materials were present, the availability of engineering controls provided by the existing plant design features for containment of effluent were identified. The licensee's review identified that most plant areas where radioactive materials were present (auxiliary building, reactor building, control building, and most of the turbine building) have adequate engineered containment of liquid and gaseous effluent within the plant's ventilation and liquid collection systems. Engineering controls were determined to be adequate when they provided sufficient capacity to contain the gaseous and liquid firefighting effluents.

The licensee also identified other plant areas where radioactive materials were present and where there were challenges to the adequacy of the engineering controls or impacts on effluent controls during fire suppression activities. In addition, the licensee identified other plant areas where radioactive materials were present where there were no engineered controls for containment of effluents (e.g., Yard Area 1 and portions of the turbine building).

LAR Attachment E, Table E-1 "Radioactive Release Compartment Review" (LAR Table E-1) provides the results of the compartment review to evaluate each fire area and identifies each fire area that is within scope based on the potential for any radiation release in the event of a fire within the fire area. LAR Attachment E, Table E-2, "Radioactive Release Transition - Engineered Controls Review" (LAR Table E-2), provides a detailed summary of the plant's engineering controls in each fire area to contain radioactive effluent. The licensee's assessment of potentially affected areas was determined by the NRC staff to be an adequate assessment because the review identified potentially affected areas and availability of engineering controls in accordance with the guidance in NEI 04-02, as endorsed by RG 1.205.

3.6.4 Fire Pre-plans

The licensee reviewed the existing fire pre-plans to determine whether the Callaway FPP is adequate to ensure that gaseous and liquid radioactive effluents generated as a direct result of fire suppression activities would be contained and monitored before release to unrestricted areas. The results of the licensee's review are documented in the LAR Table E-2. This review included the following steps:

- Identification of applicable documentation, including fire pre-plan, procedures, and support drawings.
- Review of current documentation to identify whether the current documents discuss the containment and monitoring of potential contamination involving fire suppression activities.
- Review of engineering controls for gaseous effluents to ensure that gaseous effluents (for example contaminated smoke and related particulates) are contained within the station boundaries by demonstrating that such effluents would be contained within the fire area's ventilation envelope, leading to a monitored, filtered, and elevated release. Fire areas outside the permanent radiological controlled area credit the bounding analysis that demonstrates that the limitations of 10 CFR Part 20 limits are not exceeded for gaseous radioactive effluents.
- Review of engineering controls for liquid effluents to ensure that liquid effluents (for example automatic or manual fire-fighting water) are contained within the station boundaries by demonstrating that such effluents would be contained within the area's floor drain system, which leads to a monitored storage tank system that is sized for the expected volume of runoff. For outside areas, bounding analysis is credited that demonstrates that the limitations of 10 CFR Part 20 limits are not exceeded for liquid radioactive effluents.
- Revision of documents, as necessary, to require that, where a potential for contamination exists, fire suppression agents and products of combustion be contained and monitored and fire suppression activities be monitored.

The NRC staff concludes that the licensee's review was adequate because the fire pre-plans were reviewed in accordance with the guidance in NEI 04-02, Appendix G, as endorsed by RG 1.205.

3.6.5 Gaseous Effluent Controls

In areas where engineering controls exist for containment, filtering, and monitoring of gaseous effluent, the engineering controls were determined to provide adequate containment of gaseous effluents because the effluent is contained, filtered, and monitored.

For plant areas where the effectiveness of the installed engineering controls could be challenged or impacted by fire suppression activities, the licensee will modify the FPP such that the Site Incident Commander and fire brigade will be instructed to request support from Radiation Protection personnel to establish containment of radioactive effluent and monitoring. Where possible, the fire-fighting activities will route the radioactive gaseous effluent back into the plant ventilation system for filtering and monitoring of the effluent prior to discharge. For these plant areas, the NRC staff concludes that NFPA 805 radioactive release goals, objectives, and performance criteria will be met because the radioactive release will be adequately contained by a combination of the engineered controls and compensatory actions.

In other plant areas where engineered controls do not exist for containment of radioactive effluents, the licensee performed a bounding quantitative analysis of the potential impacts of radioactive gaseous effluents during a fire. The bounding case was the yard area where radioactive materials are stored (e.g., in Intermodal containers, B-25 boxes and 55 gallon drums). The licensee's assessment was based on the type of radionuclides that are stored, and calculated the maximum amount of radioactive material that could be safely stored and released during a fire without exceeding the effluent controls specified in the Callaway FSAR. The licensee will use administrative controls to limit the amount of radioactivity stored in these containers.

During the NRC audit of the licensee's LAR, the NRC reviewed the licensee's calculation methods used to perform the bounding analysis. The analysis methods were consistent with methods used in the licensee's Offsite Dose Calculational Manual (ODCM) to calculate off-site doses to members of the public. The analysis was based on the maximum container inventory (amount of radioactivity) that could be safely stored and released during a fire without exceeding the radiological effluent controls specified in the FSAR.

The NRC staff concludes that the licensee adequately quantified the amount of radioactive material that can be safely stored and released as a gaseous effluent during a fire without exceeding the radiological release performance criteria of NFPA 805 and the public dose limits of 10 CFR Part 20. The assessment was adequate because models and assumptions used were consistent with the licensee's ODCM. The ODCM is a document required by the plant's technical specifications and is prepared in accordance with NRC regulatory guidance. The NRC concluded that the assessment was performed adequately because the methods used were based on an adequate bounding analysis that identified the maximum amount of radioactivity that could be stored without exceeding the effluent controls for a member of the public during a fire.

3.6.6 Liquid Effluent Controls

In areas where engineering controls exist for containment of liquid effluent (e.g., floor drains routed to sumps and tanks), the licensee determined that the engineering controls provide adequate containment because the effluent is collected, stored, processed and monitored in the radwaste building prior to discharge.

In areas where there are not adequate engineering controls to contain liquid effluent, the licensee will revise FPP procedures and training programs to have the fire brigade and Radiation Protection staff provide containment and monitoring of fire suppression agents and products of combustion in potentially contaminated areas .

For potential liquid effluents released in the yard area, the licensee's assessment determined that liquid effluent would either be discharged into storm drains or seep into the ground. Any liquid effluent entering the yard storm drains is adequately contained because the storm drains discharge to on-site retention ponds and prevent release to the off-site areas.

The licensee stated that Radiation Protection personnel would be involved with the fire response and perform follow up radiological surveys as necessary to identify contamination spread due to firewater run-off, therefore the NRC staff concludes that there is reasonable assurance sampling and containment of contaminated effluent will occur. The licensee also stated that site procedures will provide for additional containment, monitoring, dose evaluations, and reporting actions beyond those taken during the immediate fire event as called out in the pre-fire plan.

For liquid effluent that may seep into the ground, the licensee assessed the ground hydrogeological conditions and run-off paths. A bounding assessment was performed based on an assumption that any water, foam or other effluent used in the firefighting effort would not be contained and would thus reach the surrounding soil. The assessment concluded that the radiological impact would not result in any public dose, and therefore would not exceed the radiological release performance criteria of NFPA 805 and the public dose limits of 10 CFR Part 20.

The NRC staff reviewed the calculation methods and concludes that the licensee adequately assessed the potential impact of uncontained liquid effluent because a bounding assessment was performed based on conservative assumptions and adequate analytical methods, and the estimated radiological impacts did not exceed the radiological release performance criteria of NFPA 805 and the public dose limits of 10 CFR Part 20.

3.6.7 Fire Brigade Training Materials

The licensee reviewed the fire brigade training materials to ensure they were consistent with the pre-fire plans in terms of containment and monitoring of potentially contaminated smoke and fire suppression water. The review is documented in LAR Attachment E, Table E-3, "NEI 04-02 Table E-3, Radioactive Release - Training/Lesson Plan Review." Each training module and lesson plan was evaluated, and those training materials needing improvements were identified and documented. The NRC staff reviewed the licensee's evaluation of training materials and concludes that, upon completion of the identified revisions, the training materials will be adequate to instruct the Callaway staff to implement the FPP because staff will able to take actions to limit the radioactive effluents to within the radiological release performance criteria of NFPA 805.

3.6.8 Actions to Be Taken

LAR Attachment S provides actions already completed (LAR Table S-1), and actions yet to be taken (LAR Table S-2) to enable the Callaway FPP to meet the radioactive release goals, objectives, and performance criteria. LAR Table S-3 includes implementation items, such as procedure revisions, that will be completed prior to the implementation of the revised FPP. LAR Table S-3 includes Implementation Item 11-805-079 to revise procedure FPP-ZZ-00009, "Retraining Courses and Activities," to include the containment and monitoring of fire suppression agents and products of combustion in potentially contaminated areas.

3.6.9 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, (4) the assessment of the impact of radioactive effluents when containment is not provided, and (5) the development and implementation of newly revised fire brigade response procedures and 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 Callaway are as low as reasonably achievable and are not expected to exceed the radiological dose limits in 10 CFR Part 20. The NRC staff therefore concludes that the licensee's FPP complies with the requirements specified in NFPA 805, Sections 1.3.2, 1.4.2, and 1.5.2 and that this approach is acceptable.

3.7 NFPA 805 Monitoring Program

For this section of the SE, the following requirements from NFPA 805, Section 2.6, 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 fire protection program in meeting the performance criteria. Monitoring shall ensure that the assumptions in the engineering analysis remain valid.

NFPA 805, Section 2.6.1, "Availability, Reliability, and Performance Levels":

> 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." The NRC staff reviewed the monitoring program that the licensee developed to monitor availability, reliability, and performance of Callaway's FPP systems and features after the transition to NFPA 805. The focus of the NRC staff's 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 is acceptable (see SE Sections 2.7 and 2.8).

The licensee stated that it will develop an NFPA monitoring program consistent with FAQ 10-0059 (Reference 52). The licensee further stated that development of the monitoring program will include a review of existing surveillance, inspection, testing, compensatory measures, and oversight processes for adequacy and that the review will examine adequacy of the scope of SSCs and components within the existing plant programs, performance criteria for availability and reliability of SSCs, and the adequacy of the plant corrective action program. The licensee also stated that the monitoring program will incorporate phases for scoping, screening using risk criteria, risk target value determination, and monitoring implementation. The licensee stated that the scope of the program will include fire protection systems and features, NSCA equipment, SSCs relied upon to meet radioactive release criteria, and fire protection programmatic elements.

Based on the information provided in the LAR, as supplemented, the NRC staff concludes that the licensee's NFPA 805 monitoring program, and development and implementation process is acceptable and will provide a reasonable assurance that Callaway will implement an effective program for monitoring risk significant fire SSCs because it:

- Establishes the appropriate performance monitoring groups to be monitored.
- Uses an acceptable screening process for determining the SSCs to be included in the performance monitoring groups.
- Establishes availability, reliability, and performance criteria for the SSCs being monitored.

- Requires corrective actions when SSC availability, reliability, and performance criteria targets are exceeded in order 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 monitoring program as of the date of this SE, completion of the Callaway NFPA 805 Monitoring Program is an implementation item, as noted in LAR Attachment S, Table S-3, as implementation item 11-805-089.

Completion 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 closure of implementation item 11-805-089, 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 following 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 Callaway:

NFPA 805, Section 2.7.1, "Content."

NFPA 805, Section 2.7.1.1, "General":

> The analyses performed to demonstrate compliance with this standard shall be documented for each nuclear power plant (NPP). The intent of the documentation is that the assumptions be clearly defined and that the results be easily understood, that results be clearly and consistently described, and that sufficient detail be provided to allow future review of the entire analyses. Documentation shall be maintained for the life of the plant and be organized carefully so that it can be checked for adequacy and accuracy either by an independent reviewer or by the AHJ.

NFPA 805, Section 2.7.1.2, "Fire Protection Program Design Basis Document":

> A fire protection program design basis document shall be established based on those documents, analyses, engineering evaluations, calculations, and so forth that define the fire protection design basis for the plant. As a minimum, this document shall include fire hazards identification and nuclear safety capability

assessment, on a fire area basis, for all fire areas that could affect the nuclear safety or radioactive release performance criteria defined in Chapter 1.

NFPA 805, Section 2.7.1.3, "Supporting Documentation":

Detailed information used to develop and support the principal document shall be referenced as separate documents if not included in the principal document.

NFPA 805, Section 2.7.2, "Configuration Control."

NFPA 805, Section 2.7.2.1, "Design Basis Document":

The design basis document shall be maintained up-to-date as a controlled document. Changes affecting the design, operation, or maintenance of the plant shall be reviewed to determine if these changes impact the fire protection program documentation.

NFPA 805, Section 2.7.2.2, "Supporting Documentation":

Detailed supporting information shall be retrievable records. Records shall be revised as needed to maintain the principal documentation up-to-date.

NFPA 805, Section 2.7.3, "Quality."

NFPA 805, Section 2.7.3.1, "Review":

Each analysis, calculation, or evaluation performed shall be independently reviewed.

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.

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," to evaluate the appropriateness of the content of the Callaway FPP design basis document and supporting documentation.

The Callaway FPP design basis is a compilation of multiple documents (i.e., fire safety analyses, calculations, engineering evaluations, NSCAs, etc.), databases, and drawings which are identified in LAR Figure 4-8, "NFPA 805 Transition – Planned Post-Transition Documentation and Relationships for Callaway." The licensee stated that the analyses conducted to support the NFPA 805 transition were performed in accordance Callaway processes which meet or exceed the requirements for documentation outlined in NFPA 805, Section 2.7.1.

Specifically, the licensee stated that the design analysis and calculation procedures provide the methods and requirements to ensure that design inputs and assumptions are clearly defined, results are easily understood by being clearly and consistently described, and that sufficient detail is provided to allow future review of the entire analysis. The process includes provisions for appropriate design and engineering review and approval. In addition, the approved analyses are considered controlled documents, and are accessible via the Callaway document control system. Being analyses, they are also subject to review and revision consistent with the other plant calculations and analyses, as required by the plant design change process.

Callaway's NSCA used a computer database tool, SAFE-PB, to identify the success paths and the equipment and cables required to demonstrate that the NSPC of NFPA 805 are met for each fire area of the plant. NSCA equipment resolutions that proposed OMAs are identified as VFDRs. In addition, circuit analysis may be used to assess and disposition specific circuit failure modes.

In SSA RAI 7 dated December 11, 2012 (Reference 14), the NRC staff requested the licensee provide:

(a) A description of how the post-transition PCE process will ensure that the potential interfaces between integration databases and other databases [SAFE-PB] and analyses (e.g., the cable and raceway database, the NSCA, the FPRA, and fire modeling [FM]) are evaluated and updated, as appropriate.

(b) A description of the process that will be used to ensure that [SAFE-PB] is maintained in accordance with documentation and design configuration control processes and procedures.

(c) A description of the process and procedures that will be used to ensure that [SAFE-PB] analyses are conducted and/or updated by persons properly trained and experienced in its use.

(d) A description of the processes and procedures that will be used to ensure that [SAFE-PB] analyses comply with NFPA 805 FM, content, and quality control requirements.

In its response dated February 19, 2013 (Reference 8), the licensee stated, in part, that:

(a) Changes to the plant will be screened for potential NFPA-805 impact using the existing Engineering Screen/Hazards Review [forms]. These forms contain low level screening questions that will have the Engineer contact the Subject Matter Experts (Fire Protection Engineer, SAFE Engineer, or PRA Engineer) as needed for further evaluation. The Engineer qualified to modify the SAFE software will decide if the proposed change requires an update to the Fire Safety Analyses, SAFE software or NSCA analysis, including CARTS (Cable and Raceway Tracking System) information changes. The Fire Protection Engineer will evaluate proposed changes for any impact on the Fire Safety Analyses, fire modeling calculations, or credited Recovery Actions that are part of the NSCA analysis or Fire PRA and initiate an update. The Fire PRA Engineer will evaluate proposed changes for potential impact on the Fire Risk Evaluations, Fire Safety Analyses or the Fire PRA and make any needed updates.

(b) The post transition change process summarized in the response to question (a) above will ensure that integration databases and software are maintained with up to date plant configuration data. Maintenance and updates to the PRA models are governed by [an existing Callaway procedure], which requires that the models are maintained with up to date plant configuration data.

(c) ...Engineers who perform calculations and detailed FM must be qualified to [existing Callaway qualification standards,] which ensures that detailed [FM] is performed by persons who have the proper training and experience. Implementation Item 11-805-072 in Attachment S of the LAR will develop a Fire PRA qualification standard to ensure that Fire PRA activities are performed by properly trained and experienced personnel. A qualification standard will also be developed for Engineers who will update the SAFE-PB database.

(d) ...The integration databases and software are controlled [by existing plant procedure,] which includes verification and validation requirements. Detailed [FM] is performed in accordance with [existing Engineering Design Guide.] PRA calculations are governed [by existing procedure,] which specifies independent review and quality requirements.

The NRC staff reviewed the information provided by the licensee in the LAR and in its response to SSA RAI 7 and concludes that the integration databases and software used to perform the NFPA 805 analyses will be maintained as required by NFPA 805, Section 2.7 based on the following:

- The process planned for screening plant changes as described in subpart (a) in response to the SSA RAI 7 is acceptable since the licensee plans on using appropriately qualified engineers to review changes for necessary impacts on the FPP, the NFPA 805 analyses performed, and the electronic tools used to perform these analyses.

- The process described in subpart (b) in response to SSA RAI 7 is acceptable since it assures that the integration databases and software will be maintained with up to date configuration data; the PRA models will also be maintained up to date in accordance with existing plant procedures.

- The engineers who perform calculations and detailed FM will be qualified in accordance with existing qualification standards. Upon completion of the implementation item, PRA engineers will be qualified as will engineers maintaining the SAFE-PB database.

- The integration databases and software will be maintained by existing configuration control procedures, including V&V for detailed FM and PRA calculations.

The licensee stated in the LAR that the documentation associated with the Callaway RI/PB FPP will be maintained for the life of the plant and organized in such a way to facilitate review for accuracy and adequacy by independent reviewers, including the NRC staff.

Based on the description provided in the LAR, as supplemented, of the content of the Callaway NFPA 805 FPP design basis and supporting documentation, and taking into account 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, regarding adequate development and maintenance of the FPP design basis documentation.

3.8.2 Configuration Control

The NRC staff reviewed LAR Section 4.7.2, "Compliance with Configuration Control Requirements in Section 2.7.2 of NFPA 805," in order to evaluate the configuration control process at Callaway.

To support the many other technical, engineering, and licensing programs at Callaway, the licensee has existing configuration control processes and procedures for establishing, revising, or utilizing program documentation. Accordingly, the licensee is integrating the RI/PB FPP design basis and supporting documentation into these existing configuration control processes and procedures. These processes and procedures require that all plant changes be reviewed for potential impact on the various Callaway 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 Callaway document control system. The licensee also stated that analyses based on the PRA program, which includes the FRE, are issued as formal analyses subject to these same configuration control processes, and are additionally subjected to the PRA peer review process specified in the ASME/ANS PRA standard (Reference 35).

Configuration control of the FPP during the transition period is maintained by the Callaway PCE process, as defined in existing Callaway configuration management and configuration control procedures. The licensee will revise these existing procedures as necessary for application to the NFPA 805 FPP. The NRC staff reviewed the licensee's process for updating and maintaining the Callaway FRE, in order to reflect plant changes made after completion of the transition to NFPA 805 in Section 3.4.1 of this SE.

Based on the description of the Callaway configuration control process, which indicates that the Callaway 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 the licensee has a configuration control process which meets the requirements of NFPA 805, Sections 2.7.2.1 and 2.7.2.2, for revising FPP design basis documents, supporting documents, and applicable FPP documentation to reflect changes made to the RI/PB FPP after the NFPA 805 FPP has been implemented.

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," to evaluate the quality of the engineering analyses used to support transition to NFPA 805 at Callaway based on the requirements outlined above.

3.8.3.1 Review

NFPA 805, Section 2.7.3.1 requires that each analysis, calculation, or evaluation performed be independently reviewed. 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). The LAR also states that the analyses, calculations, and evaluations performed in support of the transition to NFPA 805 were independently reviewed, and that analyses, calculations, and evaluations to be performed post-transition will be independently reviewed as required by the existing procedures.

Based on the licensee's description of the Callaway process for performing independent reviews of analyses, calculations, and evaluations, the NRC staff concludes that the licensee's approach for meeting the requirements of NFPA 805, Section 2.7.3.1, is acceptable.

3.8.3.2 Verification and Validation

NFPA 805, Section 2.7.3.2 requires that each calculation model or numerical method used be verified and validated through comparison to test results or other acceptable models. The licensee stated in LAR Section 4.7.3 that the calculation models and numerical methods used in support of the transition to NFPA 805 were verified and validated, and that the calculation models and numerical methods used post-transition will be similarly verified and validated. As an example, the licensee provided extensive information related to the V&V of fire models used to support the development of the Callaway FRE, which the NRC staff concluded were acceptable. The NRC staff's evaluation of this information is discussed below.

3.8.3.2.1 General

NUREG-1824 documents the V&V of five selected fire models commonly used to support applications of RI/PB fire protection at NPPs. 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 NPP 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 FM elements performed by the licensee using the V&V applications contained in NUREG-1824 to support the transition to NFPA 805 at Callaway, the NRC approves the use of these models, provided that the application is used within the appropriate limitations, as identified in NUREG-1824.

In LAR Section 4.5.1.2, the licensee also identified the use of several empirical correlations that are not addressed in NUREG-1824. The NRC staff reviewed these empirical correlations, 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.

The NRC staff concluded that the theoretical bases of the models and empirical correlations used in the FM calculations that were not addressed in NUREG-1824 were identified and described in authoritative publications, such as *The Society of Fire Protection Engineers (SFPE) Handbook of Fire Protection Engineering* (Reference 72).

As reflected in Tables 3.8.3.2-1 and 3.8.3.2-2 of Attachment A to this SE, the FM employed by the licensee in the development of the FRE 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 FM analysis.

Based on the above, the NRC staff concludes that this approach provides reasonable assurance that the FM used in the development of the fire scenarios for the Callaway FRE is appropriate, and thus 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 March 2, 2012 (Reference 11), the NRC staff issued RAIs concerning the FM conducted to support the Callaway FRE. By letter dated April 17, 2012 (Reference 6), the licensee provided a partial response to the first round of FM RAIs, and requested clarification of the RAIs that were not responded to. By emails dated June 6 and June 19, 2012 (References 12 and 13, respectively), the NRC provided the requested clarification. The licensee responded to the remaining first round RAIs by letter dated July 12, 2012 (Reference 7). By letter dated December 11, 2012 (Reference 14), the NRC sent a second set of FM RAIs to the licensee. By letter dated February 19, 2013 (Reference 8), the licensee provided a response to the second round RAIs. The following paragraphs describe selected RAI responses related to the V&V of the fire models used. Several FM RAIs are not discussed in this SE section. The RAIs not discussed were issued to obtain more details on specific aspects of the V&V, and the responses allowed the NRC staff to gain a complete understanding of the V&V that was performed in support of the Callaway FPRA.

- The NRC staff noted that a plant-specific Fire Modeling Database (FMDB) and Transient Analysis Worksheets (TAWs) were developed to automate the ZOI and HGL calculations based on Fire Dynamics Tools (FDTs) in NUREG-1805 and the evaluation methodology in FIVE. FM RAI 01(a) dated March 2, 2012 (Reference 11), was issued to ask the licensee to explain how the FMDB was verified; that is, how did the licensee ensure that the equations in the FMDB were coded correctly and that the FMDB solutions are identical to those that would be obtained with the FDTs or FIVE.

  In its response to FM RAI 01(a) (Reference 6), the licensee stated that the FMDB and the TAWs were verified by comparing the results from the FMDB and TAWs for a number of cases to those produced by the NUREG-1805 FDTs and FIVE with identical inputs. The results of this verification are documented in a Callaway report.

  The NRC staff reviewed the Callaway report and concludes that the approach used by the licensee to verify the FMDB and TAWs is acceptable.

- NRC staff issued FM RAIs 01(b), 01(c), 01(e), and 01(f) dated March 2, June 6, and June 19, 2012 (References 11, 12, and 13, respectively), to request that the licensee provide technical documentation to demonstrate that the fire models and correlations used in the NFPA 805 transition have been applied with input parameters that are within the validated range, or to justify the application of the models and correlations outside the validated range reported in the V&V basis document(s).

In its response to RAIs 01(b), 01(c), 01(e), and 01(f) dated July 12, 2012 (Reference 7), the licensee demonstrated that fire models and correlations were generally applied within the validated range, and provided detailed documentation to justify the application in the cases where a model or correlation was used outside the validated range.

Based on a review of the documentation provided, the NRC staff concludes that there is reasonable assurance that fire models and correlations were either used within the validated range of input parameters, or that their application outside the validated range was justified.

- Table J-1 in Attachment J of the LAR describes the V&V basis for the fire models and correlations that were used in the plume/HGL interaction study, the temperature sensitive equipment ZOI study, and the temperature sensitive equipment HGL study. The NRC staff issued FM RAI 01(g) dated March 2, June 6, and June 19, 2012 (References 11, 12, and 13, respectively), to ask the licensee to demonstrate that the results of these studies were used within their range of applicability.

  In its response to FM RAI 01(g) dated July 12, 2012 (Reference 7), the licensee first described how the results of each study were applied and then provided examples to demonstrate that the results were used within the range of applicability of the study.

  Based on its review of the information provided, the NRC staff concludes that the licensee's response to FM RAI 01(g) provides reasonable assurance that the results of the three studies were used within their range of applicability.

- The NRC staff issued FM RAI 01(i) dated March 2, June 6, and June 19, 2012 (References 11, 12, and 13, respectively), to ask the licensee to provide the V&V basis for the smoke detector response model (method of Cleary) implemented in the FDS suppression activation analyses in fire areas C-21 and C-22, and to demonstrate that, in these analyses, FDS was either used within the range of its validity or that the use of FDS outside the verification and validation range is justified.

  In its response to FM RAI 01(i) dated July 12, 2012 (Reference 7), the licensee referred to two authoritative publications (References 70 and 73) as the V&V basis for Cleary's smoke detector response correlation. Discussion of the use of FDS within the validated range was provided as part of the licensee's response to previous FM RAI 01(e) dated July 12, 2012 (Reference 7).

  Based on its review, the NRC staff concludes that the V&V basis for Cleary's smoke detector response correlation provided by the licensee is acceptable.

- The NRC staff issued FM RAI 01(j) dated March 2 and June 6, 2012 (References 11 and 12, respectively), to ask the licensee to demonstrate that the software package PyroSim used to build the FDS input files has been verified.

  In its response to FM RAI 01(j) dated July 12, 2012 (Reference 7), the licensee described how PyroSim was verified and where this verification is documented.

  Based on its review of the information provided, the NRC staff concludes that the licensee's response to FM RAI 01(j) is acceptable.

- The NRC staff issued FM RAI 03(c) dated March 2 and June 6, 2012 (References 11 and 12, respectively) to ask the licensee to explain how it was assured that the mesh size used in the FDS analyses was within the validated range.

  In its response to FM RAI 03(c) dated July 12, 2012 (Reference 7), the licensee stated that the D*(characteristic diameter of fire) and D*/δx (where δx is size of grid cell) values were calculated for each of the FDS analyses, and that in the cases where D*/δx was outside the range specified in NUREG-1824, Volume 7 or NUREG-1934 "Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG)" (Reference 32), a sensitivity study was performed to confirm that the results and conclusions of the FDS analysis were valid.

  Based on its review and explanation provided, the NRC staff concludes that the licensee's response to FM RAI 03(c) provides reasonable assurance that the mesh size used in the FDS analyses is acceptable.

- The NRC staff issued FM RAI 03(e) dated March 2 and June 6, 2012 (References 11 and 12, respectively), to ask the licensee to provide the V&V basis for the sprinkler activation time calculations in the FDS analyses.

  In its response to FM RAI 03(e) dated July 12, 2012 (Reference 7), the licensee explained that the FDS sprinkler activation calculations rely on the ability of FDS to predict ceiling jet/gas temperature, and the method of Heskestad and Bill to calculate the thermal response of sprinkler heads. The V&V for the former is provided in NUREG-1824, Volume 7. The Heskestad and Bill method is documented in an authoritative publication (Reference 74).

  Based on its review of the response to FM RAI 03(e), the NRC staff concludes that the use of FDS to determine sprinkler activation time is acceptable.

3.8.3.2.3 Post-Transition

The licensee 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 V&V. Revision of the applicable post-transition processes and procedures to include NFPA 805 requirements for V&V is an implementation item (see LAR Attachment S, Table S-3, "Implementation Items").

3.8.3.2.4 Conclusion for Section 3.8.3.2

Based on the licensee's description of the Callaway process for V&V of calculation models and numerical methods and its commitment for continued use of this process, 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

NFPA 805, Section 2.7.3.3 requires that acceptable engineering methods and numerical models only be used for applications to the extent that these methods have been subject to V&V; and that they only are applied within the scope, limitations, and assumptions prescribed for that method. The licensee stated that the engineering methods and numerical models used in support of the transition to NFPA 805 were used subject to the limitations of use outlined in NFPA 805, Section 2.7.3.3, and that the engineering methods and numerical models used post-transition will be subject to these same limitations of use. As an example, in LAR Section 4.5.2, "Fire Modeling," the licensee stated that the fire models developed to support the NFPA 805 transition at Callaway fall within their V&V limitations.

The NRC staff assessed the acceptability of each empirical correlation or other fire model in terms of the limits of its use. Tables 3.8.3.2-1 and 3.8.3.2-2 of Attachment A to this SE summarize the fire models used, how each was applied in the Callaway FRE, the V&V basis for each, and the NRC staff evaluation for each.

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 (see LAR Attachment S, Table S-3).

Based on the licensee's statements that the fire models used to support development of the FRE were used within their limitations, and the description of the Callaway 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.

3.8.3.4 Qualification of Users

NFPA 805, Section 2.7.3.4 requires that personnel performing engineering analyses and applying numerical methods (e.g., FM) 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).

Specifically, these requirements are being addressed through the implementation of an engineering qualification process at Callaway. The licensee has developed procedures that 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). These requirements are being addressed through the implementation of an engineering qualification process. The licensee has developed qualification or training requirements for personnel performing engineering analyses and numerical methods.

The NRC staff reviewed the engineering qualification process at Callaway, and concludes that appropriately competent and experienced personnel developed the Callaway FRE, including the supporting FM calculations and including 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 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, Section 2.7.3.5 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 ASME/ANS PRA standard (Reference 35) includes requirements to address uncertainty. Accordingly, the licensee addressed uncertainty as a part of the development of the Callaway FRE. 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," (Reference 30) there are three types of uncertainty associated with FM 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 FM 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.

(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 March 2, 2012 (Reference 11), the NRC staff asked RAIs concerning the FM conducted to support the Callaway FRE. By letter dated April 17, 2012 (Reference 6), the licensee provided a partial response to the first round of FM RAIs, and requested clarification of the RAIs that were not responded to. By emails dated June 6, 2012 (Reference 12) and June 19, 2012 (Reference 13), the NRC provided the requested clarification. The licensee responded to the remaining first round RAIs by letter dated July 12, 2012 (Reference 7). By letter dated December 11, 2012 (Reference 14), the NRC sent a second set of FM RAIs to the licensee. By letter dated February 19, 2013 (Reference 8) the licensee provided a response to the second round RAIs. The following paragraphs describe selected RAI responses related to the uncertainty of the FM results.

- The NRC staff issued FM RAI 02(a) dated March 2 and June 6, 2012 (References 11 and 12, respectively), to ask the licensee to explain in detail the uncertainty analyses for FM that were performed, to describe how the uncertainties of the input parameters (geometry, HRR, RTI, etc.) were determined and accounted for, and to substantiate the statement in Appendix J of the LAR that, "...the predictions are deemed to be within the bounds of experimental uncertainty..."

  In the response to FM RAI 02(a) dated July 12, 2012 (Reference 7), the licensee provided additional information about how uncertainty associated with FM was accounted for in the analysis. Most of this information was originally included in supporting documentation from each detailed FM report provided by the licensee's FM contractor. The uncertainty analysis performed with respect to FM was qualitative in nature and focused on the fact that conservative model input parameters were used in the FM calculations and that this, in turn, provides a substantial SM. The RAI response provided the following examples of conservative modeling assumptions that provide SM:

- 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 FRE target.

- The fire elevation in most cases is at the top of the 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 increase over the normally recommended value of 0.3.

- The convective heat release rate fraction used is 0.7. The normally recommended value is between 0.6 and 0.65.

- 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 (closest) target. 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.

- Cable trays are assumed to be filled to capacity, although in reality some are only partially filled.

- As the fire propagates to secondary combustibles, the fire is modeled as one single fire using the FM closed-form correlations. The resulting plume temperature used in this analysis are therefore likely to be over-estimated, since in actuality, 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 accounted for.

- The fire elevation for transient fires is 2 feet. Most transient fires are expected to be below this height or even at floor level.

- Oil fires are analyzed as both unconfined and confined spills with 20-minute duration. Although unconfined spills result in large heat release rates and usually burn for seconds, the oil fires have been analyzed for 20 minutes to account for the uncertainty in the oil spill size.

- High energy arcing fault scenarios are 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 (i.e., 20 minutes).

In addition to this SM discussion, the licensee's response to FM RAI 02(a) included a summary of the sections in NUREG-1824 concerning the degree to which each model used at Callaway and listed in Attachment J of the LAR falls within or outside of experimental uncertainty.

Based on its review and explanation, the NRC staff concludes that the licensee's response to FM RAI 02(a) provides reasonable assurance that the results of the FM performed at Callaway in support of the transition to NFPA 805 are within the bounds of experimental uncertainty.

- The NRC staff issued FM RAI 02(b) dated March 2, June 6, and June 19, 2012, (References 11, 12, and 13, respectively), to ask the licensee to justify why cable tray obstructions could be omitted in the FDS FM analysis for Fire Areas C-21 and C-22. This RAI relates specifically to an example of model and completeness uncertainty.

  The licensee's response to FM RAI 02(b) dated July 12, 2012 (Reference 7), provided justification for omitting cable obstructions by demonstrating that the cable obstructions would not significantly affect the output parameters of the analysis (i.e., the automatic detection and suppression system activation time).

  Based on its review and explanation, the NRC staff concludes that the licensee's response to FM RAI 02(b) is acceptable.

3.8.3.5.3 Post-Transition

The licensee 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 are implementation items listed in LAR Attachment S, Table S-3.

3.8.3.5.4 Conclusion for Section 3.8.3.5

The NRC staff reviewed the licensee's description of the process for performing an uncertainty analysis, and concludes that the licensee's approach for meeting the requirements of NFPA 805, Section 2.7.3.5 is acceptable.

3.8.3.6 Conclusion for Section 3.8.3

Based on the above discussions, the NRC staff concludes that the Callaway RI/PB FPP quality assurance (QA) 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. The individual sections of this SE provide the NRC staff's evaluation of the application of the NFPA 805 quality requirements to the licensee's FPP, as appropriate.

3.8.4 Fire Protection Quality Assurance Program

GDC 1 of Appendix A to 10 CFR Part 50 requires the following:

> 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 established its Fire Protection Quality Assurance Program in accordance with the guidelines Branch Technical Position (BTP) Chemical Engineering Branch (CMEB) Position C.4, "Quality Assurance Program" (Reference 36). In addition, the guidance in Appendix C to NEI 04-02 suggests that the LAR include a description of how the existing fire protection QA program will be transitioned to the new NFPA 805 RI/PB FPP, as discussed below.

The licensee stated that the FPP QA program is included within and implemented by the Callaway nuclear QA program, although certain aspects of that program are not applicable to the FPP. As discussed in SE Section 2.4.4, the licensee included implementation items for revising the QA program to reflect the applicable requirements of Section 2.7.3 of NFPA 805 in LAR Attachment S, Table S-3.

The NRC staff concludes that the licensee's changes to the fire protection QA program are acceptable because they include the expansion of the existing program to include those fire protection systems that were previously not included within the scope of the fire protection QA program that are required by NFPA 805 Chapter 4 and they include the applicable requirements of Section 2.7.3 of NFPA 805.

3.8.5 Conclusion for Section 3.8

The NRC staff reviewed the licensee's RI/PB FPP, as described in the LAR, as supplemented, to evaluate the NFPA 805 program documentation content, the associated configuration control process, and the appropriate QA requirements. The NRC staff concludes that the licensee's approach meets the requirements specified in NFPA 805, Section 2.7, regarding program documentation, configuration control, and quality.

4.0 FIRE PROTECTION LICENSE CONDITION

The licensee proposed a 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, 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 Callaway and is, therefore, acceptable.

The following license condition is included in the revised license for the Callaway, and will replace Operating License No. NPF-30 Condition 2.C(5):

Fire Protection Program

Union Electric shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated 8/29/2011 (and supplements dated 11/9/2011, 4/17/2012, 7/12/2012, 2/19/2013, 8/5/2013, 9/24/2013, and 12/19/2013) and as approved in the safety evaluation report dated xx/**xx/2014**. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

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

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

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

Other Changes that May Be Made Without Prior NRC Approval

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program 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 report dated **xx/xx/2014** 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.

<u>Transition License Conditions</u>

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

(2) The licensee shall implement the items listed in Enclosure 2, Attachment S, Table S-3, "Implementation Items," of Ameren Missouri letter ULNRC-06031, dated December 19, 2013, by 8 months from the issuance of the license amendment.

5.0 SUMMARY

The NRC staff reviewed the licensee's application, as supplemented by various letters, to transition to an RI/PB FPP in accordance with the requirements established by NFPA 805. The NRC 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 implementation items that must be completed 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. Before the licensee is able to fully implement the transition to an FPP based on NFPA 805 and apply the new fire protection license condition, to its full extent, the implementation items must be completed within the timeframe specified.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State official was notified on October 23, 2013, of the proposed issuance of the amendment. The state official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on February 14, 2012 (77 FR 8294). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

1. National Fire Protection Association, NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition, Quincy, Massachusetts.

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, December 2009 (ADAMS Accession No. ML092730314).

3. Nuclear Energy Institute, NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c), Revision 2, Washington, DC, April 2008 (ADAMS Accession No. ML081130188).

4. Maglio, Scott, Ameren Missouri, letter to U.S. Nuclear Regulatory Commission, "Callaway Plant, Unit 1, Union Electric Co., Docket Number 50-483, Request for License Amendment to Adopt NFPA 805, 'Performance –Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)'," dated August 29, 2011 (ADAMS Accession No. ML112420022).

5. Maglio, Scott, Ameren Missouri, letter to U.S. Nuclear Regulatory Commission, "Callaway Plant, Unit 1, Union Electric Co., Docket Number 50-483, Clarification of Information Contained in Request for License Amendment to Adopt NFPA 805, 'Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)' (LCDN 11-0012)," dated November 9, 2011 (ADAMS Accession No. ML113140044).

6. Maglio, Scott, Ameren Missouri, letter to U.S. Nuclear Regulatory Commission, "Callaway Plant, Unit 1, Union Electric Co., Docket Number 50-483, Response to Request for Additional Information Regarding Adoption of National Fire Protection Association Standard NFPA 805 (TAC No. ME7046)," dated April 17, 2012 (ADAMS Accession Nos. ML12108A239 and ML12108A240).

7. Maglio, Scott, Ameren Missouri, letter to U.S. Nuclear Regulatory Commission, "Callaway Plant, Unit 1, Union Electric Co., Docket Number 50-483, Response to Request for Additional Information Regarding Adoption of National Fire Protection Association Standard NFPA 805 (TAC No. ME7046)," dated July 12, 2012 (ADAMS Accession No. ML12194A624 and ML12194A638).

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11. Thadani, Mohan, U.S. Nuclear Regulatory Commission, letter to Adam C. Heflin, Union Electric Company, "Callaway Plant Unit 1-Request for Additional Information Regarding Adoption of National Fire Protection Association Standard on the Application for Amendment to Transition the Fire Protection Program to National Fire Protection Association (NFPA) Standard 805 (TAC No. ME7046)," dated March 2, 2012 (ADAMS Accession No. ML120600186).

12. Hon, Andy, U.S. Nuclear Regulatory Commission, email to Scott Maglio, Union Electric Company, "Callaway Plant Unit 1-Request for Additional Information Regarding Adoption of National Fire Protection Association Standard on the Application for Amendment to Transition the Fire Protection Program to National Fire Protection Association (NFPA) Standard 805 (TAC No. ME7046)," dated June 6, 2012 (ADAMS Accession No. ML12159A089).

13. Hon, Andy, U.S. Nuclear Regulatory Commission, email to Thomas B. Elwood, Union Electric Company, "Callaway Plant Unit 1-Request for Additional Information Regarding Adoption of National Fire Protection Association Standard on the Application for Amendment to Transition the Fire Protection Program to National Fire Protection Association (NFPA) Standard 805 (TAC No. ME7046)," dated June 19, 2012 (ADAMS Accession No. ML121710420).

14. Lyon, Carl F., U.S. Nuclear Regulatory Commission, letter to Adam C. Heflin, Union Electric Company, "Callaway Plant Unit 1-Request for Additional Information, Round 2, Regarding Adoption of National Fire Protection Association Standard NFPA 805 (TAC No. ME7046)," dated December 11, 2012 (ADAMS Accession No. ML12335A232).

15. Lyon, Carl F., U.S. Nuclear Regulatory Commission, letter to Adam C. Heflin, Union Electric Company, "Callaway Plant Unit 1-Request for Additional Information, Round 3, Regarding Adoption of National Fire Protection Association Standard NFPA 805 (TAC No. ME7046)," dated July 30, 2013 (ADAMS Accession No. ML13204A223).

16. Lyon, Carl F., U.S. Nuclear Regulatory Commission, email to Thomas B. Elwood, Union Electric Company, "Callaway Plant, Unit 1, Request for Additional Information Email, NRR/DRA/AFPB 18.01, License Amendment Request to Adopt NFPA 805, Performance-Based Standard for Fire Protection for LWR Generating Plants (2001 Edition) (TAC No. ME7046)" dated August 16, 2013 (ADAMS Accession No. ML13260A591).

17. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011 (ADAMS Accession No. ML100910006).

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19. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants," Revision 2, October 2009 (ADAMS Accession No. ML092580550).

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24. U.S. Nuclear Regulatory Commission, NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 1: Summary and Overview," September 2005 (ADAMS Accession No. ML052580075).

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Attachments:

A. Table 3.8.3.2-1 – V&V Basis for Fire Modeling Correlations Used at Callaway

B. Table 3.8.3.2-2 – V&V Basis for Fire Model Calculations of Other Models Used at Callaway

C. Abbreviations and Acronyms

Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at Callaway

Correlation	Application at Callaway	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 TAWs. 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 (Reference 27) NUREG-1824, Volume 3 (Reference 28) SFPE Handbook, 4th Edition, Chapter 2-1 (Reference 72)	• The licensee provided verification of the FMDB and Transient Workbook on basis of comparison with NUREG-1805 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in NUREG-1824 and an authoritative publication of the Society of Fire Protection Engineers (SFPE) Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(b), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Plume Centerline Temperature (Method of Heskestad)	The Plume Centerline Temperature. correlation was implemented in the FMDB and TAWs. The correlation was used to determine vertical separation distance, based on temperature, to a	NUREG-1805, Chapter 9 (Reference 27) NUREG-1824, Volume 3 (Reference 28) SFPE Handbook, 4th Edition, Chapter 2-1 (Reference 72)	• The licensee provided verification of the FMDB and TAWs on basis of comparison with NUREG-1805 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the

Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at Callaway

Correlation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
	target in order to determine the vertical extent of the ZOI.		correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(b), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Radiant Heat Flux (Point Source Method)	The Radiant Heat Flux (Point Source Method) correlation was implemented in the FMDB and TAWs. The correlation was used to calculate 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 (Reference 27) NUREG-1824, Volume 3 (Reference 28) SFPE Handbook, 4th Edition, Chapter 3-10 (Reference 72)	• The licensee provided verification of the FMDB and TAWs on basis of comparison with NUREG-1805 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(b), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Plume Radius (Method of Heskestad)	The Plume Radius (Method of Heskestad)	SFPE Handbook, 4th Edition, Chapter 2-1 (Reference 72)	• The licensee stated that the plume radius was not used as the sole basis for any target failures (Response to FM RAI 01(d), Reference 7).

Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at Callaway

Correlation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
	correlation was implemented in the FMDB and TAWs, to calculate the horizontal radius, based on temperature of the plume at a given height. The plume radius was used as the horizontal ZOI, based on heat flux.	NUREG-1805, Chapter 2 (Reference 27) NUREG-1824, Volume 3 (Reference 28)	• The licensee provided verification of the FMDB and TAWs on the basis of a comparison with NUREG-1805 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in an authoritative publication of the SFPE Handbook. • The plume radius correlation is derived from Heskestad's plume centerline temperature correlation, for which V&V is documented in NUREG-1824. The plume radius correlation is subject to the same validated ranges (Response to FM RAI 01(d) Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Hot Gas Layer (Method of McCaffrey, Quintiere, and Harkleroad)	The Hot Gas Layer (Method of McCaffrey, Quintiere, and Harkleroad) correlation was implemented in the FMDB and TAWs. The correlation was used to calculate the	NUREG-1805, Chapter 2 (Reference 27) NUREG-1824, Volume 3 (Reference 28) SFPE Handbook, 4th Edition, Chapter 3-6 (Reference 72)	• The licensee provided verification of the FMDB and TAWs on basis of comparison with NUREG-1805 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee

Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at Callaway

Correlation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
	hot gas layer temperature for a room with natural ventilation.		provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(b), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Hot Gas Layer (Method of Beyler)	The Hot Gas Layer (Method of Beyler) correlation was implemented in the FMDB and TAWs. The correlation was used to calculate the hot gas layer temperature for a room with no ventilation.	NUREG-1805, Chapter 2 (Reference 27) NUREG-1824, Volume 3 (Reference 28) SFPE Handbook, 4th Edition, Chapter 3-6 (Reference 72)	• The licensee provided verification of the FMDB and TAWs on basis of comparison with NUREG-1805 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(b), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Hot Gas Layer (Method of Foote,	The Hot Gas Layer (Method of Foote,	NUREG-1805, Chapter 2 (Reference 27)	• The licensee provided verification of the FMDB and TAWs on basis of comparison with

Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at Callaway

Correlation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
Pagni, and Alvares [FPA])	Pagni, and Alvares) correlation was implemented in the FMDB and TAWs. The correlation was used to calculate the hot gas layer temperature for a room with forced ventilation.	NUREG-1824, Volume 3 (Reference 28) SFPE Handbook, 4th Edition, Chapter 3-6 (Reference 72)	NUREG-1805 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(b), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Hot Gas Layer (Method of Deal and Beyler)	The Hot Gas Layer (Method of Deal and Beyler) correlation was implemented in the FMDB and TAWs. The correlation was used to calculate the hot gas layer temperature for a room with forced ventilation.	NUREG-1805, Chapter 2 (Reference 27) NUREG-1824, Volume 3 (Reference 28) SFPE Handbook, 4th Edition, Chapter 3-6 (Reference 72)	• The licensee provided verification of the FMDB and TAWs on basis of comparison with NUREG-1805 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI

Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at Callaway

Correlation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
			01(b), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Ceiling Jet Temperature (Method of Alpert)	The Ceiling Jet Temperature (Method of Alpert) correlation was implemented in the FMDB and TAWs. 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 (Reference 28) SFPE Handbook, 4th Edition, Chapter 2-2 (Reference 72)	• The licensee provided verification of the FMDB and TAWs on basis of comparison with FIVE, Rev1 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(b), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Smoke Detection Actuation Correlation (Method of Heskestad and Delichatsios)	Smoke Detection Actuation (Method of Heskestad and Delichatsios) correlation was implemented in the	NUREG-1805, Chapter 11 (Reference 27) NUREG-1824, Volume 3 (Reference 28)	• The licensee provided verification of the FMDB and TAWs on basis of comparison with NUREG-1805 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE

Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at Callaway

Correlation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
	FMDB and TAWs. 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.	SFPE Handbook, 4th Edition, Chapter 2-2 (Reference 72) SFPE Handbook, 4th Edition, Chapter 4-1, Custer R., Meacham B., and Schifiliti, R., 2008. (Reference 72)	Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(b), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Sprinkler Activation Correlation	Sprinkler Activation Correlation was implemented in the FMDB and TAWs. The correlation was used to estimate sprinkler actuation timing based on ceiling jet temperature,	NUREG-1805, Chapter 10 (Reference 27) NFPA Handbook, 19th Edition, Chapter 3-9 (Reference 75)	• The licensee provided verification of the FMDB and TAWs on basis of comparison with NUREG-1805 (Response to FM RAI 01(a), Reference 6). • The correlation is validated in an authoritative publication of the NFPA Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the

Attachment A: Table 3.8.3.2-1, V&V Basis for Fire Modeling Correlations Used at Callaway

Correlation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
	velocity, and thermal response of sprinkler.		correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(c), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.

Attachment B: Table 3.8.3.2-2, V&V Basis for Other Fire Models and Related Calculations Used at Callaway

Calculation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
Hot Gas Layer Calculations using Fire Dynamics Simulator (FDS) (Version 5)	FDS (Version 5) was used to calculate hot gas layer height and temperatures.	NUREG-1824, Volume 7 (Reference 28) NIST Special Publication 1018-5, Volume 2 (Reference 76) NIST Special Publication 1018-5, Volume 3 (Reference 76)	• The modeling technique is validated in NUREG-1824 and NIST Special Publication 1018-5 Volume 2 and 3 • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(e), Reference 7. Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Hot Gas Layer Calculations using Consolidated Model of Fire and Smoke Transport (CFAST) (Version 6)	CFAST (Version 6) was used to calculate upper and lower layer temperatures compartments, the layer height, and smoke obscuration for various conditions. It was also used to calculate abandonment time for the Callaway main control room.	NUREG-1824, Volume 5 (Reference 28) NIST Special Publication 1086 (Reference 77)	• The modeling technique is validated in NUREG-1824 and NIST Special Publication 1086 • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(f), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.

Attachment B: Table 3.8.3.2-2, V&V Basis for Other Fire Models and Related Calculations Used at Callaway

Calculation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
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 equipment.	NUREG-1824, Volume 5 (Reference 28) NIST Special Publication 1086 (Reference 77)	• The modeling technique is validated in NUREG-1824 and NIST Special Publication 1086 • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(f), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
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 (Reference 28) NIST Special Publication 1018-5, Volume 2 (Reference 76) NIST Special Publication 1018-5, Volume 3	• The modeling technique is validated in NUREG-1824 and NIST Special Publication 1018-5 • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(e), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the

Attachment B: Table 3.8.3.2-2, V&V Basis for Other Fire Models and Related Calculations Used at Callaway

Calculation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
		(Reference 76)	Callaway 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 (Reference 28) NIST Special Publication 1018-5, Volume 2 (Reference 76) NIST Special Publication 1018-5, Volume 3 (Reference 76)	• The modeling technique is validated in NUREG-1824 and NIST Special Publication 1018-5 • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 01(e), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Corner and Wall HRR	The corner and wall HRR was used to adjust the HRR for fires near a wall or corner	SFPE Handbook, 4th Edition, Chapter 2-14 (Reference 72) "Properties of Fire Plumes," Zukoski, 1995 (Reference 78) "Natural Convection Flows and Associated Heat Transfer Processes in Room	• The modeling technique is documented in the SFPE Handbook, Chapter 2-14. • The 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 78-81. The licensee provided justification for cases where the correlation was used outside validated range reported in these authoritative publications (Response to FM RAI 01(c), Reference 7).

Attachment B: Table 3.8.3.2-2, V&V Basis for Other Fire Models and Related Calculations Used at Callaway

Calculation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
		Fires," Sargent, 1983 (Reference 79) "Entrainment and Flame Geometry of Fire Plumes," Cetegen, 1982 (Reference 80) "Ignition Sources in Room Fire Tests and Some Implications for Flame Spread Evaluation," Williamson, 1991(Reference 81).	Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Correlation for HRRs 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 (Reference 72). "Heat Release Rate Characteristics of Some Combustibles Fuel Sources in Nuclear Power Plants," Lee, 1985 (Reference 82)	• The modeling technique is documented in Chapter 3-1 of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied in configurations similar to that reported by Lee (Reference 82). The licensee provided justification for cases where the correlation was used outside the configuration reported in the authoritative publication (Response to FM RAI 01(c), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.

Attachment B: Table 3.8.3.2-2, V&V Basis for Other Fire Models and Related Calculations Used at Callaway

Calculation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
Correlation for Flame Spread over Horizontal Cable Trays (FLASH-CAT)	The FLASH-CAT method was used to predict the growth and spread of a fire within a vertical stack of horizontal cable trays	NUREG/CR-7010, Section 9 (Reference 29)	• The modeling technique is validated in NUREG/CR-7010 • The licensee stated that in most cases, the correlation has been applied in configurations similar to that reported in NUREG/CR-7010. The licensee provided justification for cases where the correlation was used outside the configuration reported in the authoritative publication (Response to FM RAI 01(c), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this correlation in the Callaway application is acceptable.
Smoke Detector Response (Method of Cleary)	The smoke detector response model (method of Cleary) was used in conjunction with FDS (Version 5) to estimate smoke detector activation in fire areas C-21 and C-22	NIST Special Publication 965 (Reference 70) NIST GCR 07-911 (Reference 73)	• The modeling technique is validated in NIST Special Publication 965 and NIST GCR 07-911 • The licensee provided justification for using the method with a soot yield outside the validated range (Response to FM RAI 01(i), Reference 7). Based on its review and explanation, the NRC staff concludes that the use of this method as implemented in FDS in the Callaway application is acceptable.
Sprinkler Head Thermal Response (Method of Heskestad and Bill)	The sprinkler head thermal response model (method of Heskestad and Bill) was used in	"Quantification of Thermal Responsiveness of Automatic Sprinklers Including Combustion	• The modeling technique is validated in Vol 14 of 1988 Fire Safety Journal (Reference 74) Based on the observation that the method is described in a peer-reviewed professional journal article, the NRC

Attachment B: Table 3.8.3.2-2, V&V Basis for Other Fire Models and Related Calculations Used at Callaway

Calculation	Application at Callaway	V&V Basis	NRC Staff Evaluation of Acceptability
	conjunction with FDS (Version 5) to estimate sprinkler activation in fire areas A-11, C-30 and C-31	Effects," Fire Safety Journal, Vol. 14, 1988 (Reference 74)	staff concludes that the use of this method as implemented in FDS in the Callaway application is acceptable.

Attachment C: Abbreviations and Acronyms

AC	alternating current
ADAMS	Agencywide Documents Access and Management System
AFW	auxiliary feedwater
AHJ	authority having jurisdiction
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
BWR	boiling-water reactor
CCDP	conditional core damage probability
CCF	common-cause failure
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
CPT	control power transformer
CRS	control room supervisor
CT	current transformer
DC	direct current
DID	defense-in-depth
DID RA	defense-in-depth recovery action
ECA	equipment cabinet area
Epsilon (ε)	Non-zero but below truncation limit
EEEE	existing engineering equivalency evaluation
EPRI	Electric Power Research Institute
ERFBS	electrical raceway fire barrier system
ERO	emergency response organization
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
FHRA	Fire Human Reliability Analysis
FIVE	Fire Induced Vulnerability Evaluation Methodology
FLASH-CAT	Flame Spread over Horizontal Cable Trays
FM	fire modeling
FMDB	fire modeling database
FPE	fire protection engineering
FPP	fire protection program
FPRA	fire probabilistic risk assessment
FR	Federal Register
FRE	fire risk evaluation
FSAR	final safety analysis report
ft	foot
GDC	general design criteria
GL	generic letter
gpm	gallons per minute

HDPE	high-density polyethylene
HEP	human error probability
HFE	human failure event
HGL	hot gas layer
HRA	human reliability analysis
HRE	high(er) risk evolution
HRR	heat release rate
IN	information notice
in.	inches
IEEE	Institute of Electrical and Electronics Engineers
KSF	key safety function
kW	kilowatt
LAN	licensing action number
LAR	license amendment request
LERF	large early release frequency
LFS	limiting fire scenario
LOCA	loss-of-coolant accident
MAAP	Modular Accident Analysis Program
MCB	main control board
MCR	main control room
MEFS	maximum expected fire scenario
min	minute(s)
MSO	multiple spurious operation
NEI	Nuclear Energy Institute
NIST	National Institute of Standards and Technology
NFPA	National Fire Protection Association
No.	number
NPO	non-power operation
NPP	nuclear power plant
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
OQAM	Operating Quality Assurance Manual
OMA	operator manual action
P&ID	piping and instrumentation drawing
PACR	prior approval clarification request
PB	performance-based
PCE	plant change evaluation
PCS	primary control station
PORV	power-operated relief valve
POS	plant operational state
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
psi	pounds per square inch
PWR	pressurized-water reactor
PWROG	PWR Owner's Group
QA	quality assurance

RA	recovery action
RAI	request for additional information
RCP	reactor coolant pump
RCS	reactor coolant system
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RHR	residual heat removal
RI	risk-informed
RI/PB	risk-informed, performance-based
RIS	regulatory issue summary
RTI	response time index
SE	safety evaluation
SER	safety evaluation report
SFPE	Society of Fire Protection Engineers
SG	steam generator
SM	safety margins
SNUPPS	Standardized Nuclear Unit Power Plant System
SSA	safe shutdown analysis
SSC	structures, systems, and components
TAWs	Transient Analysis Worksheets
TDAFW	turbine-driving auxiliary feedwater
TR	technical/topical report
TS	technical specifications
UHS	ultimate heat sink
V&V	verification and validation
VAC	volts alternating current
VFDR	variance from deterministic requirements
WOG	Westinghouse Owners Group
YD	yard
yr	year
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/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Amendment No. 206 to NPF-30
2. Safety Evaluation

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