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

June 21, 2019

EA-14-094

Mr. Mark B. Bezilla
Site Vice President
FirstEnergy Nuclear Operating Company
Mail Stop A-DB-3080
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 298 TO ADOPT NATIONAL FIRE PROTECTION ASSOCIATION STANDARD 805 (CAC NO. MF7190, EPID L-2015-LLF-0001)

Dear Mr. Bezilla:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 298 to Renewed Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station (Davis-Besse), Unit No. 1. The amendment is in response to your application dated December 16, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15350A314), as supplemented by letters dated February 2, March 7, July 28, and December 16, 2016; January 17, June 16, and October 9, 2017; April 2, September 11, and November 20, 2018; and May 13, 2019 (ADAMS Accession Nos. ML16033A085, ML16067A195, ML16210A422, ML16351A330, ML17017A504, ML17170A000, ML17284A190, ML18094A798, ML18254A073, ML18324A677, and ML19134A032, respectively). The amendment transitions the current fire protection program at Davis-Besse to one based on the National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition, as incorporated into Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.48(c).

The application, as supplemented, also identified several current exemptions to 10 CFR Part 50, Appendix R, granted by the NRC on November 23, 1982; April 18, 1990; January 30, 1998; December 26, 2002; July 21, 2005; and July 21, 2005 (ADAMS Accession Nos. ML021160466, ML021190037, ML021190569, ML02126023, ML020100366, and ML050970136, respectively), that would no longer be applicable once the amendment is implemented. These exemptions are rescinded upon implementation of the amendment.

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

Sincerely,

Blake Purnell, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures:
1. Amendment No. 298 to NPF-3
2. Safety Evaluation

cc: Listserv

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

AND

FIRSTENERGY NUCLEAR GENERATION, LLC

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

DOCKET NO. 50-346

Amendment No. 298
Renewed License No. NPF-3

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

   A. The application for amendment filed by FirstEnergy Nuclear Operating Company (FENOC, the licensee) dated December 16, 2015, as supplemented by letters dated February 2, March 7, July 28, and December 16, 2016; January 17, June 16, and October 9, 2017; April 2, September 11, and November 20, 2018; and May 13, 2019, 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 paragraphs 2.C.(2) and 2.C(4) of Renewed Facility Operating License No. NPF-3 are hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 298, are hereby incorporated in the renewed license. FENOC shall operate the facility in accordance with the Technical Specifications.

2.C(4) Fire Protection

FENOC 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 December 16, 2015, as supplemented by letters dated February 2, March 7, July 28, and December 16, 2016; January 17, June 16, and October 9, 2017; April 2, September 11, and November 20, 2018; and May 13, 2019, and as approved by Amendment No. 298. 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 probabilistic risk assessment model, methods that have been approved by the 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 10^{-7}/year for core damage frequency and less than 10^{-8}/year for large early release frequency. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

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

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

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

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 by Amendment No. 298, 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) and (3) 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 modification described in Attachment S, Table S-1, "Plant Modifications Committed," to the FENOC letter dated November 20, 2018, within 2 years following issuance of the license amendment. The licensee shall maintain appropriate compensatory measures in place until completion of this modification.

(3) The licensee shall implement the items listed in Attachment S, Table S-2, "Implementation Items," to the FENOC letter dated November 20, 2018, within 2 years following issuance of the license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented in accordance with paragraph 2.C(4) of the license.

FOR THE NUCLEAR REGULATORY COMMISSION

Lisa M. Regner, Acting Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: June 21, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 298

RENEWED FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Renewed Facility Operating License and Appendix A, Technical Specifications (TSs), with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
Renewed License NPF-3	Renewed License NPF-3
L-5	L-5
L-7	L-7
--	L-7a
--	L-7b
TSs	TSs
5.4-1	5.4-1

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

(1) Maximum Power Level

FENOC is authorized to operate the facility at steady state reactor core power levels not in excess of 2817 megawatts (thermal). Prior to attaining the power level, Toledo Edison Company shall comply with the conditions identified in Paragraph (3) (o) below and complete the preoperational tests, startup tests and other items identified in Attachment 2 to this license in the sequence specified. Attachment 2 is an integral part of this renewed license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 298, are hereby incorporated in the renewed license. FENOC shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission:

(a) FENOC shall not operate the reactor in operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

(b) Deleted per Amendment 6

(c) Deleted per Amendment 5

2.C(4) Fire Protection

FENOC 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 December 16, 2015, as supplemented by letters dated February 2, March 7, July 28, and December 16, 2016; January 17, June 16, and October 9, 2017; April 2, September 11, and November 20, 2018; and May 13, 2019, and as approved by Amendment No. 298. 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 probabilistic risk assessment model, methods that have been approved by the 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 10^{-7}/year for core damage frequency and less than 10^{-8}/year for large early release frequency. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

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

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

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

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 by Amendment No. 298, 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) and (3) 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 modification described in Attachment S, Table S-1, "Plant Modifications Committed," to the FENOC letter dated November 20, 2018, within 2 years following issuance of the license amendment. The licensee shall maintain appropriate compensatory measures in place until completion of this modification.

(3) The licensee shall implement the items listed in Attachment S, Table S-2, "Implementation Items," to the FENOC letter dated November 20, 2018, within 2 years following issuance of the license amendment.

(5) Deleted per Amendment No. 279.

(6) Antitrust Conditions

FENOC and FirstEnergy Nuclear Generation, LLC shall comply with the antitrust conditions delineated in Condition 2.E of this renewed license as if named therein. FENOC shall not market or broker power or energy from the Davis-Besse Nuclear Power Station, Unit No. 1. FirstEnergy Nuclear Generation, LLC is responsible and accountable for the actions of FENOC to the extent that said actions affect the marketing or brokering of power or energy from the Davis-Besse Nuclear Power Station, Unit No. 1, and in any way, contravene the antitrust license conditions contained in the renewed license.

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 to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring; and

d. All programs specified in Specification 5.5.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c)

AMENDMENT NO. 298 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-3

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

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

Table of Contents

1.0 INTRODUCTION 1
1.1 Background 1
1.2 Requested Licensing Action 3
2.0 REGULATORY EVALUATION 5
2.1 Applicable Regulations 9
2.2 Applicable Guidance 10
2.3 NFPA 805 Frequently Asked Questions 15
2.4 Orders, License Conditions, and Technical Specifications 17
2.4.1 Orders 17
2.4.2 License Conditions 18
2.4.3 Technical Specifications 20
2.5 Rescission of Exemptions 21
2.6 Self-Approval Process for Fire Protection Program Changes 22
2.7 Modifications and Implementation Items 24
3.0 TECHNICAL EVALUATION 25
3.1 NFPA 805 Fundamental FPP Elements and Minimum Design Requirements 26
3.1.1 Compliance with NFPA 805, Chapter 3, Requirements 27
3.1.2 Identification of Power Block 37
3.1.3 Plant-Specific Treatments or Technologies 37
3.1.4 Approval Requests for Performance-Based Methods for NFPA 805, Chapter 3, Elements 37
3.2 Nuclear Safety Capability Assessment Methods 60
3.2.1 Compliance with NFPA 805 Nuclear Safety Capability Assessment Methods 61
3.2.2 Maintaining Fuel in a Safe and Stable Condition 67
3.2.3 Applicability of Feed-and-Bleed 70
3.2.4 Assessment of Multiple Spurious Operations 70
3.2.5 Establishing Recovery Actions 71
3.3 Fire Modeling 73
3.4 Fire Risk Assessments 74
3.4.1 Maintaining Defense-in-Depth 75
3.4.2 Safety Margins 76

3.4.3 Quality of the Fire Probabilistic Risk Assessment 77
3.4.4 Fire Risk Evaluations 94
3.4.5 Additional Risk Presented by Recovery Actions 95
3.4.6 Cumulative Risk and Combined Changes 96
3.4.7 Uncertainty and Sensitivity Analyses 98
3.4.8 Conclusion for Section 3.4 99
3.5 Nuclear Safety Capability Assessment Results 100
3.5.1 Nuclear Safety Capability Assessment Results by Fire Area 100
3.5.2 Fire Protection During Non-Power Operational Modes 110
3.5.3 Conclusion for Section 3.5 113
3.6 Radioactive Release Performance Criteria 113
3.7 NFPA 805 Monitoring Program 115
3.8 Post-Implementation Plant Change Evaluation Process 116
3.9 Program Documentation, Configuration Control, and Quality Assurance 118
3.9.1 Documentation 119
3.9.2 Configuration Control 119
3.9.3 Quality 120
4.0 FIRE PROTECTION LICENSE CONDITION 123
5.0 SUMMARY 126
6.0 STATE CONSULTATION 127
7.0 ENVIRONMENTAL CONSIDERATION 127
8.0 CONCLUSION 127
9.0 REFERENCES 128

ATTACHMENT: Abbreviations and Acronyms A-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED

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

AMENDMENT NO. 298 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-3

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

1.1 Background

The U.S. Nuclear Regulatory Commission (NRC or Commission) started developing fire protection requirements in the 1970s. In 1976, the NRC published comprehensive fire protection guidelines in Branch Technical Position (BTP), Auxiliary and Power Conversion Systems Branch (APCSB) 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" (Reference 1), and Appendix A to BTP APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976" (Reference 2). Subsequently, the NRC performed fire protection reviews for the operating reactors and documented the results in safety evaluations (SEs) or supplements to SEs.

To resolve issues identified in those SEs and supplements, the NRC amended its regulations with the publication of the final rule on the fire protection program (FPP) for operating nuclear power plants in the *Federal Register* (FR) on November 19, 1980 (45 FR 76602). This rule added Section 50.48, "Fire protection," and Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.

Section 50.48(a)(1) of 10 CFR requires, in part, each holder of an operating license to have a fire protection plan that satisfies General Design Criterion (GDC) 3, "Fire protection," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. Section 50.48(a)(1) states that the fire protection plan must describe the overall FPP; identify the positions responsible for the program and the authority delegated to those positions; and outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage.

Section 50.48(a)(2) of 10 CFR states that the fire protection plan must describe the specific features necessary to implement the program described in 10 CFR 50.48(a)(1), including administrative controls and personnel requirements for fire prevention and manual fire suppression activities; automatic and manual fire detection and suppression systems; and the means to limit fire damage to structures, systems, and components (SSC) important to safety to ensure the capability to safely shut down the plant.

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 3), which describes a methodology for establishing fundamental 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 and allows PB or deterministic approaches to be used to meet performance criteria.

Chapter 1 of NFPA 805 describes the performance goals, objectives, and criteria that must be met through the implementation of PB or deterministic approaches. The goals include ensuring that reactivity control, inventory and pressure control, decay heat removal (DHR), vital auxiliaries, and process monitoring are achieved and maintained. Plant fire protection requirements must be established using the methodology in Chapter 2 of NFPA 805, such that the minimum FPP elements and design criteria contained in Chapter 3 of NFPA 805 are satisfied. The fire areas and fire hazards must be identified though a plant-wide analysis, and either a PB or deterministic approach must be used to demonstrate that the performance criteria are met. Engineering evaluations, probabilistic risk assessments (PRAs),[1] and fire modeling (FM) calculations are used as part of a PB approach to show that the performance criteria are met. Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features necessary to meet the performance criteria. Chapter 4 also specifies that at least one success path to achieve the nuclear safety performance criteria (NSPC) shall be maintained free of fire damage by a single fire.

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, modifications, and supplementation (69 FR 33536). The amended rule allows licensees to apply for a license amendment to comply with the 2001 Edition of NFPA 805 as an alternative to complying with 10 CFR 50.48(b). The NFPA has issued subsequent editions of NFPA 805, but the NRC does not endorse them.

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

In parallel with the Commission's efforts to issue a rule incorporating the provisions of NFPA 805, the Nuclear Energy Institute (NEI) published implementing guidance for 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),"

---

[1] A PRA and a probabilistic safety assessment are considered synonymous. Although NFPA 805 uses the term probabilistic safety assessment, this SE will use the term PRA.

Revision 2 (Reference 4). Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1 (Reference 5), endorses portions of NEI 04-02, Revision 2, subject to certain exceptions, that provide methods acceptable to the NRC staff for adopting an FPP consistent with the 2001 Edition of NFPA 805 and 10 CFR 50.48(c).

1.2 Requested Licensing Action

By application dated December 16, 2015 (Reference 6), as supplemented by letters dated February 2, 2016 (Reference 7); March 7, 2016 (Reference 8); July 28, 2016 (Reference 9); December 16, 2016 (Reference 10); January 17, 2017 (Reference 11); June 16, 2017 (Reference 12); October 9, 2017 (Reference 13); April 2, 2018 (Reference 14); September 11, 2018 (Reference 15); November 20, 2018 (Reference 16); and May 13, 2019 (Reference 17), FirstEnergy Nuclear Operating Company (the licensee) submitted a license amendment request (LAR)[2] for Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS or Davis-Besse). The proposed amendment would transition the current FPP at DBNPS to a new FPP based on NFPA 805, as incorporated by reference into 10 CFR 50.48(c), and would change the license and the technical specifications (TSs). The supplemental letters were primarily in response to the NRC staff's requests for additional information (RAIs) dated October 18, 2016 (Reference 18), April 19, 2017 (Reference 19), and July 19, 2018, (Reference 20). The licensee's supplemental letters dated July 28 and December 16, 2016; January 17, June 16, and October 9, 2017; April 2, September 11, and November 20, 2018; and May 13, 2019, 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* on April 12, 2016 (81 FR 21599).

The December 16, 2015, application included multiple attachments. The supplements provided additional information, including changes to the original application and its attachments. In some cases, the licensee's supplements replaced specific attachments in their entirety. The LAR attachments and reference to the most recent complete version[3] are as follows:

Attachment A:	NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements (Reference 6)
Attachment B:	NEI 04-02 Table B-2 – Nuclear Safety Capability Assessment – Methodology Review (Reference 6)
Attachment C:	NEI 04-02 Table B-3 – Fire Area Transition (Reference 6)
Attachment D:	NEI 04-02 Non-Power Operational Modes Transition (Reference 6)
Attachment E:	NEI 04-02 Radioactive Release Transition (Reference 6)
Attachment F:	Fire-Induced Multiple Spurious Operations Resolution (Reference 6)
Attachment G:	Recovery Actions Transition (Reference 15)
Attachment H:	NFPA 805 Frequently Asked Question Summary Table (Reference 6)
Attachment I:	Definition of Power Block (Reference 6)
Attachment J:	Fire Modeling V&V (Reference 6)
Attachment K:	Existing Licensing Action Transition (Reference 13)
Attachment L:	NFPA 805 Chapter 3 Requirements for Approval (10 CFR 50.48(c)(2)(vii) (Reference 13)

[2] LAR is used in this SE to refer to the application, as supplemented.

[3] In some cases, the licensee revised portions of the LAR attachments in later supplements. The partial revisions to LAR attachments are not identified here, but are identified in the SE where applicable.

Attachment M:	License Condition Changes (Reference 6)
Attachment N:	Technical Specification Changes (Reference 6)
Attachment O:	Orders and Exemptions (Reference 6)
Attachment P:	RI-PB Alternatives to NFPA 805 10 CFR 50.48(c)(4) (Reference 6)
Attachment Q:	No Significant Hazards Evaluations (Reference 6)
Attachment R:	Environmental Considerations Evaluation (Reference 6)
Attachment S:	Plant Modifications and Items to be Completed During Implementation (Reference 16)
Attachment T:	Clarification of Prior NRC Approvals (Reference 6)
Attachment U:	Internal Events PRA Quality (Reference 6)
Attachment V:	Fire PRA Quality (Reference 6)
Attachment W:	Fire PRA Insights (Reference 15)

The existing deterministic fire protection licensing basis for DBNPS, which implements the fire protection requirements of 10 CFR 50.48 and 10 CFR Part 50, Appendix R, was approved by the NRC by letters dated July 26, 1979 (Reference 21), and May 30, 1991 (Reference 22), and is described in the DBNPS Updated Final Safety Analysis Report (UFSAR). The proposed amendment would transition the current deterministic FPP to an RI/PB FPP in accordance with 10 CFR 50.48(c). The proposed RI/PB FPP would use 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 DBNPS is often referred to as the RI/PB FPP in this SE. In its LAR, the licensee provided a description of the proposed new FPP at DBNPS that it will implement under 10 CFR 50.48(a) and (c), and the results of the evaluations and analyses required by NFPA 805.

The licensee also proposed a new fire protection license condition and a revision to the TSs to support the transition to the new FPP at DBNPS. SE Sections 2.4.2 and 4.0 discuss in detail the license condition, and SE Section 2.4.3 discusses the TS change.

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

1. The licensee has identified any orders, license conditions, and TSs that must be revised or superseded, as required by 10 CFR 50.48(c)(3)(i), and that the proposed revisions to the TSs and license conditions, as modified by the NRC, are adequate.

2. The licensee's approach, methods, and data are acceptable to establish, implement, and maintain the proposed RI/PB FPP in accordance with 10 CFR 50.48(c), subject to completion of the modification in Table S-1 and implementation items in Table S-2 of LAR Attachment S (Reference 16).

3. The proposed PB methods requested in accordance with 10 CFR 50.48(c)(2)(vii) are acceptable alternatives to the corresponding NFPA 805, Chapter 3, requirements.

4. The proposed RI/PB FPP for DBNPS is acceptable and that the licensee has demonstrated that the new FPP will meet the requirements of GDC 3, 10 CFR 50.48(a), and 10 CFR 50.48(c).

5. There is reasonable assurance that a fire in any plant area during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition.

2.0 REGULATORY EVALUATION

On July 11, 1967, the Atomic Energy Commission published for public comment in the *Federal Register* (32 FR 10213), a revised and expanded set of 70 draft general design criteria (GDC). On February 20, 1971, the Atomic Energy Commission published in the *Federal Register* (36 FR 3255) a final rule that added Appendix A (final GDC) to 10 CFR Part 50, which was amended on July 7, 1971 (36 FR 12733). Differences between the 1967 draft GDC and the final GDC included a consolidation from 70 to 64 criteria.

DBNPS received its construction permit on March 24, 1971 and was licensed for operation on April 22, 1977. Section 3.0, "Design Criteria – Structures, Systems and Components," of the NRC safety evaluation report (NUREG-0136) related to the operation of DBNPS describes the NRC staff's evaluation of the facility's conformance with the GDC for the original facility operating license. DBNPS was designed and constructed in accordance with the 1967 draft GDC. However, the applicant for the DBNPS operating license described the conformance with the final GDC, as amended on July 7, 1971, in its Final Safety Analysis Report. The NRC safety evaluation report concluded that the plant design conformed to the intent of the final GDC, as amended on July 7, 1971.

The fire protection rule (10 CFR 50.48 and 10 CFR Part 50, Appendix R), effective February 17, 1981, set forth fire protection features required to satisfy GDC 3. DBNPS UFSAR Section 3D.1.3, "Criterion 3 - Fire Protection," describes the facility conformance with GDC 3 as follows:

> Structures, systems, and components important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials are used wherever practical throughout the unit, particularly in locations such as the containment, control room and areas containing components of engineered safety features. Fire-detection and fighting systems of appropriate capacity and capability are provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire-fighting systems are designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.
>
> The system arrangements and component designs are such that water damage to critical systems does not prevent the safe shutdown [SSD] of the station.
>
> Equipment and facilities for fire protection (Subsection 9.5.1), including detection, alarm, and extinguishment, are provided to protect both station and personnel from fire, explosion, and the resultant release of toxic vapors. Both wet and dry type fire-fighting equipment are provided.
>
> Normal fire protection is provided by deluge systems, sprinklers, hose lines, and portable extinguishers.
>
> The fire protection system is designed in accordance with the requirements of the American Nuclear Insurers and Nuclear Electric Insurance Limited as a guide and applicable codes and regulations of the State of Ohio.

> The fire suppression system is provided with test hose valves that are accessible for periodic testing.

The current FPP for DBNPS is described in UFSAR Section 9.5.1, "Fire Protection Program."

The regulations in 10 CFR 50.48 establish requirements for nuclear power plant fire protection. Section 50.48 includes specific requirements for requesting approval for an RI/PB FPP based on the provisions of NFPA 805 (Reference 3). Section 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 [10 CFR 50.48(a)] 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," published in the *Federal Register* on June 16, 2004 (69 FR 33536). The statement of considerations states, in part:

> 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, 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 NFPA 805. Licensees who choose to adopt 10 CFR 50.48(c), but wish to use 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," must submit an LAR in accordance with 10 CFR 50.48(c)(2)(vii). Licensees may also request to use RI or PB alternatives to NFPA 805 in accordance with 10 CFR 50.48(c)(4). The LAR for DBNPS included several requests under 10 CFR 50.48(c)(2)(vii), which are discussed in LAR Attachment L, but the LAR for did not include any requests under 10 CFR 50.48(c)(4).

In addition to the conditions outlined by the rule that require licensees to submit an LAR for NRC review and approval in order to adopt an RI/PB FPP, a licensee may also submit additional elements of its FPP for which it wishes to receive specific NRC review and approval, as discussed in Section C.2.2.1 of RG 1.205 (Reference 5). Requesting approval for such additional elements can alleviate uncertainty in portions of the current FPP licensing bases. RGs are not substitutes for regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in RGs 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 10 CFR 50.48(c) requirements.

The 2001 Edition of NFPA 805 provides a comprehensive RI/PB standard for fire protection. It specifies the minimum fire protection requirements for existing light-water reactors during all phases of plant operations, including shutdown, degraded conditions, and decommissioning. The scope of NFPA 805 includes goals related to nuclear safety, radioactive release, life safety, and plant damage/business interruption. The goals, objectives, and criteria in NFPA 805, Chapter 1, related to life safety and plant damage/business interruption are not endorsed by the NRC. The specific SE sections identify the relevant parts of NFPA 805 considered in the NRC staff's review. The requirements in NFPA 805, Chapter 1, related to the scope, defense-in-depth (DID), and the goals, objectives, and criteria for nuclear safety and radioactive release are as follows:

- Section 1.1 "Scope"

  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.

- Section 1.2, "Defense-in-Depth"

  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 [referred to as DID Echelons 1, 2, and 3] is provided:

  (1) Preventing fires from starting

  (2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting 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 safety functions from being performed

- Section 1.3.1, "Nuclear Safety Goal"

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

- Section 1.3.2, "Radioactive Release Goal"

  The radioactive release goal is to provide reasonable assurance that a fire will not result in a radiological release that adversely affects the public, plant personnel, or the environment.

- Section 1.4.1, "Nuclear Safety Objectives"

  In the event of a fire during any operational mode and plant configuration, the plant shall be as follows:

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

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

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

- Section 1.4.2, "Radioactive Release Objective"

  Either of the following objectives shall be met during all operational modes and plant configurations.

  (1) Containment integrity is capable of being maintained.

  (2) The source term is capable of being limited.

- Section 1.5.1, "Nuclear Safety Performance Criteria"

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

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

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

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

(d) *Vital Auxiliaries.* Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.

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

- Section 1.5.2, "Radioactive Release Performance Criteria"

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

2.1 Applicable Regulations

The NRC staff considered the following regulations in its review of the LAR.

GDC 3 to 10 CFR Part 50, Appendix A, states:

> 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.

The regulations in 10 CFR 50.48 establish fire protection requirements for nuclear power plants. Paragraph 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. Paragraph 50.48(c) incorporates the 2001 Edition of NFPA 805 (Reference 3) by reference, with certain exceptions, modifications and supplementation. Paragraph 50.48(c) establishes the requirements for using an RI/PB FPP in conformance with NFPA 805 as an alternative to the requirements in 10 CFR 50.48(b) and Appendix R to 10 CFR Part 50 for plants licensed to operate prior to January 1, 1979.

Part 20, "Standards for Protection Against Radiation," of 10 CFR establishes radiation protection limits including the definition of "as low as reasonably achievable."

2.2 Applicable Guidance

In addition to NFPA 805, the NRC staff considered the following NRC RGs, guidance documents, technical reports, codes, and standards in its review.

RG 1.205, Revision 1 (Reference 5), provides guidance for complying with the RI/PB FPP requirements in 10 CFR 50.48(c), including the 2001 Edition of NFPA 805. RG 1.205 endorses portions of NEI 04-02, Revision 2 (Reference 4), that 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 and exceptions to the guidance provided in NEI 04-02. Should a conflict occur between NEI 04-02 and RG 1.205, the regulatory positions in RG 1.205 govern. This RG also states that Chapter 3 of NEI 00-01, "Guidance for Post Fire Safe Shutdown Circuit Analysis," Revision 2, issued May 2009 (Reference 23), when used in conjunction with NFPA 805 and the RG, provides an acceptable approach to circuit analysis for a plant implementing an FPP under 10 CFR 50.48(c).

NEI 04-02, Revision 2 (Reference 4), provides guidance for implementing the requirements of 10 CFR 50.48(c), and, to the extent endorsed by the NRC in RG 1.205, it provides methods acceptable to the NRC for implementing in whole or in part an RI/PB FPP. NEI 04-2 has two primary purposes: (1) to provide direction and clarification for adopting NFPA 805 as an acceptable approach to fire protection, consistent with 10 CFR 50.48(c), and (2) to provide additional supplemental technical guidance and methods for using NFPA 805 and its appendices to demonstrate compliance with fire protection requirements. The clarification and additional guidance in NEI 04-02 helps to ensure consistency and effective use of NFPA 805. The NEI 04-02 guidance focuses attention on the RI/PB FPP 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 incorporated guidance from the original version of RG 1.205 and from the closure of frequently asked questions (FAQs).

NEI 00-01, Revision 2 (Reference 23), provides a deterministic methodology for performing a post-fire safe shutdown analysis (SSA). In addition, NEI 00-01 includes information on RI 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 methods are intended for application by licensees to determine the risk significance of identified circuit failure issues related to MSOs.

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 24), provides guidance for using risk information to support nuclear power plant LARs. The guidance does not preclude other approaches for requesting licensing basis changes. Rather, RG 1.174 is intended to improve consistency in regulatory decisions when the results of risk analyses are used to help justify the 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 25), provides guidance for determining the technical adequacy of the base PRA used in an RI 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; and

4. documentation to support a regulatory submittal.

RG 1.200 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.

The American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) joint standard, 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 26), provides guidance related to PRAs used to support RI decisions for commercial light-water reactors and prescribes a method for applying this guidance to specific applications. The standard provides 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 the large early release frequency (LERF). The only hazards explicitly excluded from the scope are accidents resulting from intentional 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. The NRC position on ASME/ANS RA-Sa-2009 is described in RG 1.200.

NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2 (Reference 27), provides guidance for conducting and documenting a peer review for PRAs. The original intent of NEI 05-04 was to provide a methodology for PRA peer reviews as a follow-on to the NEI 00-02 methodology. With the release of ASME and ANS standards (to form the basis of a peer review), the emphasis of NEI 05-04 changed from follow-on peer reviews to simply peer reviews performed against an industry consensus standard. The NRC position on NEI 05-04 is described in RG 1.200.

NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Revision 1 (Reference 28), provides guidance for conducting and documenting an FPRA peer review. NEI 07-12 provides a method for reviewing an FPRA against part 4 of ASME/ANS RA-Sa-2009. The NRC position on NEI 07-12 is described in RG 1.200.

RG 1.189, "Fire Protection for Nuclear Power Plants," Revision 2, issued October 2009 (Reference 29), provides guidance to licensees on the proper content and quality of engineering equivalency evaluations used to support the FPP. This RG provides a comprehensive fire protection guidance document and identifies the scope and depth of fire protection that the NRC staff would consider acceptable for nuclear power plants.

NUREG-0800, Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection Program," Revision 0, issued December 2009 (Reference 30), 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 for Risk-Informed License Amendment Requests After Initial Fuel Load," Revision 3, issued September 2012 (Reference 31), provides the NRC staff with guidance for evaluating the technical adequacy of a licensee's PRA results when used to support 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 32), provides the NRC staff with guidance for evaluating the risk information used by a licensee to support permanent RI changes to the licensing basis.

To address the need for improved methods, the NRC Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI) embarked upon a program to develop state-of-art FPRA methodology. Both RES and EPRI provided specialists in fire risk analysis, FM, electrical engineering, human reliability analysis (HRA), 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 ensured that divergent technical views were fully considered, yet encouraged consensus at many points during the deliberation. The results of this program are documented in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," Volumes 1 (Reference 33) and 2 (Reference 34), and Supplement 1 (Reference 35), which reflects the current state-of-the-art in FPRA. NUREG/CR-6850 provides a compendium of methods, data, and tools to perform an FPRA and develop associated insights, and consensus was reached on all technical issues documented in the report.

However, as discussed in a June 14, 2013, NRC memorandum (Reference 36), new experimental information documented in NUREG/CR-6931, "Cable Response to Live Fire (CAROLFIRE)," issued April 2008 (Reference 37), and NUREG/CR- 7100, "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire): Test Results," issued April 2012 (Reference 38), indicates that the reduction factor for hot short probabilities in circuits with control power transformers identified in NUREG/CR-6850 cannot be repeated in experiments. Therefore, it was recommended that this reduction factor not be used for circuits with control power transformers.

NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)" (Reference 39), establishes good practices for performing HRAs and reviewing HRAs to assess the quality of those analyses. The HRAs in NUREG-1792 are of a generic nature and support implementation of RG 1.200 for level 1 and limited level 2 internal events PRAs with the reactor at full power.

NUREG-1805, "Fire Dynamics Tools (FDT^s): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program" (Reference 40), provides quantitative methods, known as FDT^s, to assist regional fire protection inspectors in performing fire hazard analysis. The FDT^s are used for RI analyses of credible fires that may cause critical damage to essential SSD equipment.

NUREG-1824, "Verification and Validation [V&V] of Selected Fire Models for Nuclear Power Plant Applications," Volumes 1 through 7 (Reference 41), provides technical documentation regarding the predictive capabilities of a specific set of fire models for the analysis of fire hazards in nuclear power plant scenarios. This report is the result of a collaborative program with EPRI and the National Institute of Standards and Technology (NIST). Volume 1 contains the comprehensive main report, and Volume 2 describes the experiments and associated experimental uncertainty used in developing this report. The remaining volumes describe the V&V of the following fire models:

Volume 3: FDT^s developed by the NRC;

Volume 4: Fire Induced Vulnerability Evaluation Methodology, Revision 1, developed by EPRI;

Volume 5: The Consolidated Fire Growth and Smoke Transport model developed by NIST;

Volume 6: The MAGIC zone model developed by Electricité de France; and

Volume 7: The Fire Dynamics Simulator computational fluid dynamics model developed by NIST.

NUREG/CR-7010, "Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE), Phase 1: Horizontal Trays," Volume 1 (Reference 42), describes the first phase of the CHRISTIFIRE testing program conducted by NIST. The overall goal of this multiyear program was to quantify the burning characteristics of grouped electrical cables installed in cable trays. This first phase of the program focused on horizontal tray configurations. CHRISTIFIRE addressed 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/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," Volume 1 (Reference 43), documents the results of a phenomena identification and ranking table exercise that was undertaken on fire-induced electrical circuit failures that may occur in nuclear power plants when cables are damaged by fires. NUREG/CR-7150, Volume 2 (Reference 44), documents the PRA expert elicitation results, and includes the best estimate conditional probabilities of hot-short-induced spurious operations of control circuits, given fire damage to associated cables.

NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Volume 1 (Reference 45), provides an overview of the RI decision-making process and guidance on how to treat uncertainties associated with PRA in RI decision-making.

NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines" (Reference 46), presents the state of the art in fire HRA. This report was developed jointly by RES and EPRI to provide a 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 47), describes the implications of the V&V results from NUREG-1824 for fire model users. The features and limitations of the fire models documented in NUREG-1824 are discussed relative to their use to support nuclear power plant fire hazard analyses. The report also provides information to assist fire model users in applying this technology in the nuclear power plant environment.

NUREG-2178, "Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE-FIRE)," Volume 1 (Reference 48), provides a refined approach for the characterization and modeling of fires in electrical enclosures. The report provides methods and data for the classification of electrical enclosures and determination of peak HRR probability distributions that are newer than what was provided in NUREG/CR-6850. The report also provides methods and data for the characterization of fire plumes associated with fires in electrical enclosures.

Generic Letter 2006-03. "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations" (Reference 49), identified concerns regarding the use of Hemyc and MT fire barrier systems used in nuclear power plants to protect circuits and other electrical components. The generic letter requested that operating reactor licensees evaluate their facilities to confirm compliance with the existing applicable regulatory requirements considering these concerns and take additional actions, if appropriate.

NFPA 13, "Standard for the Installation of Sprinkler Systems" (Reference 50), provides the minimum requirements for the design and installation of sprinkler systems to ensure that systems will work as intended to deliver adequate water in a fire emergency.

NFPA 14, "Standard for the Installation of Standpipe, Private Hydrant, and Hose Systems" (Reference 51), provides 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 30, "Flammable and Combustible Liquids Code" (Reference 52), provides safeguards to reduce the hazards associated with the storage, handling, and use of flammable and combustible liquids.

NFPA 50A, "Standard for Gaseous Hydrogen Systems at Consumer Sites" (Reference 53), provides the requirements for the installation of gaseous hydrogen systems on consumer sites where the hydrogen supply originates outside the consumer site and is delivered by mobile equipment.

NFPA 55, "Compressed Gases and Cryogenic Fluids Code" (Reference 54), facilitates protection from physiological, over-pressurization, explosive, and flammability hazards associated with compressed gases and cryogenic fluids

NFPA 58, "Liquified Petroleum Gas Code" (Reference 55), provides guidelines for safe liquid petroleum gas storage, handling, transportation, and use, which can mitigate risks and ensure safe installation to prevent failures, leaks, and tampering that could lead to fires and explosions.

NFPA 72, "National Fire Alarm Code" (Reference 56), provides requirements for the application, installation, location, performance, inspection, testing, and maintenance of fire alarm systems,

supervising station alarm systems, public emergency alarm reporting systems, fire warning equipment, emergency communications systems, and their components.

NFPA 101, "Life Safety Code" (Reference 57), provides the minimum requirements for egress; features of fire protection, sprinkler systems, alarms, emergency lighting, and smoke barriers; and special hazard protection.

2.3 NFPA 805 Frequently Asked Questions

In the LAR, the licensee referenced several documents commonly known as NFPA 805 FAQs. The following table provides the list of FAQs referenced in this SE, reference to the associated closeout memoranda, and a cross reference to the associated SE sections. This list does not include FAQs incorporated into NEI 04-02, Revision 2.

Table 2.3-1: NFPA 805 Frequently Asked Questions

FAQ #	FAQ Summary	Reference	SE Section
06-0008	This FAQ provides a general description of the fire protection engineering evaluation process, the different types of evaluations that may be used under NFPA 805, when prior NRC approval is needed, and how that approval is to be obtained.	(Reference 58)	3.1.1
07-0030	This FAQ provides an acceptable process for determining the recovery actions (RAs) for NFPA 805 Chapter 4 compliance. The process includes: ▪ Differentiating between RAs and activities in the main control room (MCR) or at primary control stations. ▪ Determining which RAs are required by the NFPA 805 FPP. ▪ Evaluating the additional risk presented by the use of RAs. ▪ Evaluating the feasibility of the RAs. ▪ Evaluating the reliability of the RAs.	(Reference 59)	3.2.1 3.2.5 3.4.5
07-0038	This FAQ describes 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 FPRA and nuclear safety capability assessment (NSCA) to include MSOs of concern. ▪ Step 4 – Evaluate for NFPA 805 compliance. ▪ Step 5 – Document the results.	(Reference 60)	3.2.4

FAQ #	FAQ Summary	Reference	SE Section
07-0039	This FAQ provides additional detail for the comparison of an SSD strategy to the guidance in NEI 00-01, Revision 1 (Reference 61). The process has the licensee: ▪ Assemble industry and plant-specific documentation; ▪ Determine which sections of NEI 00-01 are applicable; ▪ Compare the existing SSD methodology to the applicable guidance; and ▪ Document any discrepancies.	(Reference 62)	3.2.1
07-0040	This FAQ clarifies what is an acceptable NFPA 805 program for non-power operation (NPO). The process includes: ▪ Selecting NPO equipment and cabling. ▪ Evaluating higher risk evolutions (HREs) for NPO. ▪ Analyzing key safety functions (KSFs) for NPO. ▪ Identifying plant areas to protect or "pinch points" during HREs for NPO and actions to be taken if KSFs are lost.	(Reference 63)	3.5.2
08-0053	This FAQ provides guidance regarding the damage threshold for Kerite-FR cable.	(Reference 64)	3.1.4
08-0054[4]	This FAQ provides an acceptable process to demonstrate Chapter 4 compliance for transition: ▪ Step 1 – Assemble documentation. ▪ Step 2 – Document fulfillment of NSPC. ▪ Step 3 – Variance from deterministic requirements (VFDR) identification, characterization, and resolution considerations. ▪ Step 4 – PB evaluations. ▪ Step 5 – Final VFDR evaluation. ▪ Step 6 – Document required fire protection systems and features.	(Reference 65)	3.4.4 3.5.1
09-0056	This FAQ provides an acceptable level of detail and content for the radioactive release section of an LAR to adopt NFPA 805. The LAR should include: ▪ Justification of the radioactive release review compartmentation, if the 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.	(Reference 66)	3.6

[4] As noted in LAR Attachment H, FAQ 08-0054 was incorrectly identified as FAQ 07-0054 in an NRC closure memo and FAQ 07-0054 was used in the LAR. This SE uses the correct reference to FAQ 08-0054.

FAQ #	FAQ Summary	Reference	SE Section
10-0059	This FAQ provides clarification in the following areas regarding the implementation of an NFPA 805 monitoring program for transition: ▪ Monitoring program analysis units; ▪ Screening of low safety significant SSC; ▪ Action level thresholds; and ▪ Use of existing monitoring programs.	(Reference 67)	3.7
13-0004	This FAQ provides supplemental guidance for application of the damage criteria provided in Sections 8.5.1.2 and H.2 of NUREG/CR-6850 for solid-state control components.	(Reference 68)	3.4.3
13-0005	This FAQ provides additional guidance for detailed FPRA and FM concerning self-ignited cable fires and cable fires caused by welding and cutting.	(Reference 69)	3.4.3
13-0006	This FAQ provides guidance for modeling of junction box fires scenarios in an FPRA.	(Reference 70)	3.4.3
14-0009	This FAQ provides guidance regarding the treatment in an FRPA of well-sealed motor control centers operating at 440 volts or higher.	(Reference 71)	3.4.3

2.4 Orders, License Conditions, and Technical Specifications

Paragraph 50.48(c)(3)(i) of 10 CFR states that an LAR to transition to NFPA 805 must identify any orders and license conditions that must be revised or superseded and contain any necessary revisions to the plant's TSs and the bases thereof.

2.4.1 Orders

LAR Attachment O states that no NRC orders need to be superseded or revised. LAR Attachment O further states:

> A specific review was performed of the license amendment that incorporated the mitigation strategies required by Section B.5.b of Commission Order EA-02-026 (TAC No. MD4498) to ensure that any changes being made for compliance with 10 CFR 50.48(c) do not invalidate existing commitments applicable to the plant. The review of this order demonstrated that changes to the fire protection program will not affect measures required by B.5.b.

Based on its review, the NRC staff accepts the licensee's determination that no orders need to be superseded or revised to implement NFPA 805 at DBNPS. In addition, although NRC Order EA-02-026 has been withdrawn, the requirements of Section B.5.b are retained in DBNPS license condition 2.C(8) and are incorporated into 10 CFR 50.54(hh)(2).

2.4.2 License Conditions

LAR Section 5.2.1, "License Condition Changes," states that the current DBNPS fire protection license condition 2.C(4) is to be replaced by the following proposed license condition in LAR Attachment M:

> FENOC shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated _____ (and supplements dated _____) and as approved in the safety evaluation report dated _____ (and supplements dated _____). Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.
>
> Risk-Informed Changes that May Be Made Without Prior NRC Approval
>
> A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at 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 1E-7/year for CDF and less than 1E-8/year for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

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

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

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

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 _____ 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 modifications described in the _____ submittal of the (Davis-Besse) NFPA 805 Transition Report, Table S-1, "Plant Modifications Committed," to complete the transition to full compliance with 10 CFR 50.48(c) by _____.

(3) The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

The proposed license condition is based on the standard license condition in RG 1.205, Revision 1, with plant-specific information included. SE Sections 2.6 and 3.8 provide the NRC staff's evaluation of the self-approval process for FPP changes. SE Section 2.7 provides a discussion of the modification and implementation items. SE Section 4.0 provides the NRC staff's review of the proposed license condition.

2.4.3 Technical Specifications

LAR Section 5.2.2, "Technical Specifications," and LAR Attachment N state that the licensee conducted a review of the DBNPS TSs to determine which TSs will be impacted by the transition to NFPA 805. The licensee indicated that only TS 5.4.1 needs to be revised. In LAR Attachment N, the licensee proposed to revise TS 5.4.1 as follows (additions in bold, deletions in strikethrough):

> Written procedures shall be established, implemented, and maintained covering the following activities:
>
> a. The applicable procedures recommended in Regulatory Guide1.33, Revision 2, Appendix A, February 1978;
> b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;
> c. Quality assurance for effluent and environmental monitoring; **and**
> d. ~~Fire Protection Program implementation; and~~
> ~~e.~~ All programs specified in Specification 5.5.

Section 50.48(c) of 10 CFR incorporates NFPA 805 by reference. NFPA 805, Section 3.2.3, "Procedures," establishes requirements for procedures, and states, in part: "Procedures shall be established for implementation of the fire protection program." NFPA 805, Section 3.2.3, will become a requirement for DBNPS with the transition to NFPA 805. Maintaining the FPP in the list of activities requiring written procedures in TS 5.4.1 would be redundant to NFPA 805, Section 3.2.3. Therefore, the NRC staff concludes that the proposed change to TS 5.4.1 is acceptable.

2.5 Rescission of Exemptions

The DBNPS FPP is currently based on compliance with 10 CFR 50.48(a); 10 CFR 50.48(b); Sections III.G, III.J, and III.O of 10 CFR Part 50, Appendix R, unless specifically exempted; and the DBNPS fire protection license condition 2.C(4). LAR Section 2.2, "NRC Acceptance of the Fire Protection Licensing Basis," identifies exemptions from 10 CFR Part 50, Appendix R, for DBNPS, and provides a summary of each exemption (identified as Licensing Actions 1–14). These exemptions were granted by the NRC by letters dated November 23, 1982 (Reference 72), August 20, 1984 (Reference 73), April 18, 1990 (Reference 74), January 30, 1998 (Reference 75), December 26, 2002 (Reference 76), and July 21, 2005 (Reference 77). However, as discussed in the LAR, an exemption to Section III.L of 10 CFR Part 50, Appendix R, granted on August 20, 1984, was rescinded by the NRC on June 24, 2004 (Reference 78).

In LAR Attachment O, the licensee stated that the current DBNPS exemptions from 10 CFR Part 50, Appendix R, can be rescinded with the approval to transition to NFPA 805. These exemptions will no longer be needed because Appendix R will not be part of the licensing basis for DBNPS with the approval to transition to NFPA 805. Details regarding each of the current exemptions to 10 CFR Part 50, Appendix R, and justification for their disposition are provided in the October 10, 2017, version of LAR Attachment K (Reference 13).

For the following exemptions, LAR Attachment K indicates that the underlying condition has been evaluated using RI/PB methods and found to be acceptable, with no further actions needed, because DID and sufficient safety margin is maintained. Therefore, the bases for these exemptions will not be transitioned to the NFPA 805 FPP:

- Licensing Action 1: Exemption to Section III.G.3 of 10 CFR Part 50, Appendix R, granted on November 23, 1982. This exemption addressed the lack of a fixed fire suppression system in fire compartments FF-01, FF-02, and FF-03.

- Licensing Action 2: Exemption to Section III.G.2 of 10 CFR Part 50, Appendix R, originally granted on November 23, 1982, and superseded by an exemption granted on December 26, 2002. This exemption addressed the lack of separation of redundant SSD components by a 1-hour-rated fire barrier in fire compartment T-01.

- Licensing Action 4: Exemption to Section III.G.2 of 10 CFR Part 50, Appendix R, granted on April 18, 1990. This exemption addressed the lack of separation of redundant SSD components by a 3-hour-rated fire barrier in fire compartments A-04 and A-05.

- Licensing Action 5: Exemption to Section III.G.3 of 10 CFR Part 50, Appendix R, granted on April 18, 1990. This exemption addressed the lack of a fixed fire suppression system in fire compartment AB-01.

- Licensing Action 6: Exemption to Section III.G.2 of 10 CFR Part 50, Appendix R, granted on April 18, 1990. This exemption addressed the lack of separation of redundant SSD components by a 1-hour-rated fire barrier in fire compartment D-01.

- Licensing Action 7: Exemption to Section III.G.3 of 10 CFR Part 50, Appendix R, granted on April 18, 1990. This exemption addressed the lack of a fixed fire suppression system in fire compartment EE-01.

- Licensing Action 9: Exemption to Section III.G.3 of 10 CFR Part 50, Appendix R, granted on April 18, 1990. This exemption addressed the lack of a fixed fire suppression system in fire compartment R-01.

- Licensing Action 10: Exemption to Section III.J of 10 CFR Part 50, Appendix R, granted on April 18, 1990. This exemption addressed the lack of 8-hour battery-powered emergency lighting.

- Licensing Action 14: Exemption to Section III.G.3 of 10 CFR Part 50, Appendix R, granted on July 21, 2005. This exemption addressed the lack of a fixed fire suppression system in fire compartment HH-01.

For the following exemptions, the licensee stated, as discussed in LAR Attachment K, that the engineering evaluation of the underlying condition will be used as a qualitative engineering evaluation for the transition to NFPA 805. Therefore, the bases for these exemptions will be transitioned to the NFPA 805 FPP. The NRC staff's review of these licensing actions is discussed in SE Section 3.5.1.3.

- Licensing Action 3: Exemption to Section III.G.2 of 10 CFR Part 50, Appendix R, granted on August 20, 1984. This exemption addressed the lack of separation of redundant SSD components in fire compartments E-01 and F-01.

- Licensing Action 8: Exemption to Section III.G.2 of 10 CFR Part 50, Appendix R, granted on April 18, 1990. This exemption addressed the lack of separation of redundant train cables in fire compartment MA-01.

- Licensing Action 11: Exemption to Section III.G.2 of 10 CFR Part 50, Appendix R, granted on April 18, 1990. This exemption addressed the lack of separation of embedded conduits in concrete from redundant SSD components.

- Licensing Action 12: Exemptions to Section III.O of 10 CFR Part 50, Appendix R, granted on August 20, 1984, and January 30, 1998. These exemptions addressed the lack of capability for oil collection system (OCS) to contain the oil from the reactor coolant pumps (RCPs).

The NRC staff determined that the exemptions to 10 CFR Part 50, Appendix R, for DBNPS will no longer be applicable once the transition to NFPA 805 is completed. Therefore, the NRC staff finds it acceptable to rescind the current exemptions to 10 CFR Part 50, Appendix R, identified by the licensee with the approval of the proposed license amendment.

2.6 Self-Approval Process for Fire Protection Program Changes

Section 50.48(c) of 10 CFR allows certain changes to be made to the FPP without prior NRC review and approval (referred to as self-approval in this SE), following the transition to an RI/PB FPP. Section C.3.1 of RG 1.205 states that the NRC will provide this flexibility through a license condition for amendments authorizing the transition to an RI/PB FPP under 10 CFR 50.48(c).

The proposed license condition (see SE Section 2.4.2), which is consistent with the standard license condition in RG 1.205, includes the self-approval process. In addition, NFPA 805, Sections 2.2.9 and 2.4.4, specify requirements for plant change evaluations.

NFPA 805, Section 2.2.9, "Plant Change Evaluation," states:

> 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 [Section] 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.

The proposed license condition provides structure and detailed criteria to allow the licensee to self-approve changes to the DBNPS FPP if the requirements of NFPA 805 regarding engineering analyses, FREs, and plant change evaluations are met. The licensee intends to use an FPRA to evaluate the risk of proposed future plant changes at DBNPS. Risk assessments for plant change evaluations must use methods that are acceptable to the NRC staff, which include (1) methods that have been used in developing the peer-reviewed FPRA model, (2) 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 (3) methods that have been demonstrated to bound the risk impact.

Consistent with RG 1.205, the proposed license condition allows self-approval of changes that clearly result in a decrease in risk or that result in a risk increase less than 10^{-7}/year for CDF and less than 10^{-8}/year for LERF. In addition, the change must also be consistent with the DID philosophy and maintain sufficient safety margins. The NRC staff's review of the technical adequacy of the FPRA, including the licensee's process to ensure that the FPRA remains current, is discussed in SE Section 3.4. The NRC staff's review of the licensee's plant change evaluation process is discussed in SE Section 3.8.

The proposed license condition also includes a provision for self-approval of changes to the FPP that may be made on a qualitative, rather than quantitative basis. Specifically, the proposed license condition would allow the licensee to make changes to the FPP if an engineering evaluation demonstrates that an alternative to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement or adequate for the hazard. In either case, 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. In accordance with NFPA 805, Section 2.4, "Engineering Analysis," the use of a qualitative engineering analysis is an acceptable means of evaluating a change to the FPP against the NFPA 805 performance criteria.

The demonstration that an alternative is functionally equivalent to an NFPA 805, Chapter 3, element does not fall under NFPA 805, Section 1.7, "Equivalency," because the alternative must

meet the requirements in NFPA 805, Chapter 3. NFPA 805, Section 1.7, is a standard provision used throughout NFPA standards that allows owners or operators to use the latest state-of-the-art fire protection systems, methods, or devices, provided the alternatives are of equal or superior quality, strength, fire resistance, durability, and safety. However, Section 1.7 requires prior NRC approval to use such items because not all of these state-of-the-art items have relevant operating experience.

Prior NRC review and approval are not required to implement alternatives that an engineering evaluation has demonstrated are adequate for the hazard for the following sections of NFPA 805, Chapter 3:

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 must meet the requirements in Section 2.4 and Section 2.7, "Program Documentation, Configuration Control, and Quality," of NFPA 805. NFPA 805, Section 2.4, requires, in part, that: "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 2.4 for the plant being analyzed." The associated evaluations must also meet the documentation content and quality requirements of NFPA 805, Section 2.7, to be considered adequate. The NRC staff's review of the licensee's compliance with NFPA 805, Section 2.7, is provided in SE Section 3.9.

The proposed license condition also defines limitations on self-approval 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 necessary because NFPA 805 requires that the risk analyses be based on the as-built, as-operated and maintained plant, and reflect the operating experience at the plant. Until the proposed implementation items and plant modification listed in LAR Attachment S (Reference 16) are completed, the risk analysis will not be consistent with the as-built, as-operated and maintained plant. Therefore, for changes made during the transition period, the proposed license condition will require the licensee to use its plant change evaluation screening process (see SE Section 3.8) and ensure that fire protection DID and safety margins are maintained.

2.7 Modifications and Implementation Items

As discussed in Section C.2.1 of RG 1.205, Revision 1 (Reference 5), when the NRC issues an NFPA 805 license amendment it will impose a license condition requiring the use of NFPA 805 along with an implementation schedule. Section C.3.1 of RG 1.205 provides a sample license condition that includes: (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.

By letter dated November 20, 2018 (Reference 16), the licensee provided the final version of LAR Attachment S and a revised implementation schedule. Table S-1 identifies one uncompleted plant modification that is necessary to bring DBNPS into compliance with NFPA 805. Table S-2 provides the list of implementation items, which are actions that the licensee will complete during implementation of the license amendment to transition to NFPA 805 (e.g., procedure changes or development of NFPA 805 programs). Table S-3 provides a description of the plant modifications that have been completed. The licensee proposed to complete the plant modification in LAR Table S-1 and the implementation items in LAR Table S-2 within 2 years of issuance of the license amendment. The licensee also stated that appropriate compensatory measures will be maintained until the plant modification is completed.

The NRC staff confirmed that the modifications identified in LAR Tables S-1 and S-3 are consistent with the modifications listed in LAR Attachment C. Aside from the one modification listed in LAR Table S-1, the LAR did not identify any other uncompleted modifications credited for the transition to NFPA 805. The NRC staff also confirmed that the implementation items identified in LAR Table S-2 are the same as those identified in the LAR (primarily LAR Attachments A, C, G, and J). The NRC staff also determined that completion of the plant modification and implementation items in accordance with the proposed schedule is acceptable because it does not change or impact the NRC staff's conclusions in this SE.

The plant modification in LAR Table S-1 and the implementation items in LAR Table S-2 were credited by the licensee in its analysis supporting the LAR and must be completed for DBNPS to fully transition to NFPA 805. In addition, the NRC staff relied on this analysis and the implementation items in its review. Therefore, the NRC staff will require their completion within the proposed schedule through a license condition (see SE Section 4.0). The NRC staff, through an onsite audit or during a future fire protection inspection, may examine the completion of the implementation items.

3.0 TECHNICAL EVALUATION

The following SE sections provide the NRC staff's evaluation of the technical aspects of the LAR to transition the FPP to one based on NFPA 805 in accordance with 10 CFR 50.48(c). The NRC staff used the guidance provided in NUREG-0800, Section 9.5.1.2 (Reference 30), to review the LAR to determine whether the licensee adequately demonstrated that it will comply with the requirements of NFPA 805, 10 CFR 50.48, and GDC 3.

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

Section 3.2 provides the NRC staff's evaluation of the methods used by the licensee to demonstrate the ability to meet the nuclear safety performance criteria (NSPC) in NFPA 805, Section 1.5.1.

Section 3.3 provides the NRC staff's evaluation of the fire modeling (FM) methods used by the licensee to demonstrate the ability to meet the NSPC using a PB FM approach.

Section 3.4 provides the NRC staff's evaluation of the fire risk assessments used to demonstrate the ability to meet the NSPC using a PB fire risk evaluation (FRE) approach.

Section 3.5 provides the NRC staff's evaluation of the licensee's nuclear safety capability assessment (NSCA) results by fire area.

Section 3.6 provides the NRC staff's evaluation of the methods used by the licensee to demonstrate an ability to meet the radioactive release performance criteria in NFPA 805, Section 1.5.2.

Section 3.7 provides the NRC staff's evaluation of the NFPA 805 monitoring program developed as a part of the transition to NFPA 805.

Section 3.8 provides the NRC staff's evaluation of the licensee's post-implementation plant change evaluation process.

Section 3.9 provides the NRC staff's evaluation of the licensee's program documentation, configuration control, and quality assurance.

3.1 NFPA 805 Fundamental FPP Elements and Minimum Design Requirements

Chapter 3 of NFPA 805 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, industrial fire brigades, and fire protection structures, systems, and components (SSC). However, 10 CFR 50.48(c) provides the following exceptions, modifications, and supplementation to certain aspects of NFPA 805, Chapter 3:

- 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 (which would allow existing cable in place prior to adoption of NFPA 805 to remain as is) 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) – PB methods. Section 3.1 of NFPA 805 prohibits the use of PB methods to demonstrate compliance with the NFPA 805, Chapter 3, requirements. Notwithstanding this prohibition, the FPP elements and minimum design requirements of NFPA 805, Chapter 3, may be subject to the PB methods permitted elsewhere in the standard.

In addition, Section 3.1 of NFPA 805 specifically allows the use of alternatives to the fundamental FPP requirements in NFPA 805, Chapter 3, that have been previously approved by the NRC.

The LAR describes the process the licensee used to assess the proposed DBNPS FPP against the NFPA 805, Chapter 3, requirements. The LAR summarizes how the licensee will comply

with each of the Chapter 3 requirements, including the use of previously approved alternatives where applicable. The NRC staff reviewed the LAR to evaluate the licensee's compliance with each applicable Chapter 3 requirement. The NRC staff also evaluated whether the previously approved alternatives remain valid. The NRC staff did not review the licensee's compliance with Chapter 3 sections that have no technical requirements.

3.1.1 Compliance with NFPA 805, Chapter 3, Requirements

The licensee used the systematic process described in NEI 04-02 (Reference 4), as endorsed by the NRC in RG 1.205 (Reference 5), with some modifications, to assess the proposed DBNPS 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 DBNPS FPP. LAR Attachment A (Reference 6) provided specific compliance statements for each NFPA 805, Chapter 3, attribute that contained applicable requirements. Some sections of NFPA 805, Chapter 3 do not contain requirements or are not applicable to DBNPS. In addition, the licensee made multiple compliance statements to fully document compliance with some subsections of NFPA 805, Chapter 3.

The following compliance categories were used by the licensee to demonstrate compliance with the fundamental FPP elements and minimum design requirements:

1. The existing FPP element directly complies with the requirement (noted as "Complies" in LAR Attachment A).

2. The existing FPP element complies using an explanation or clarification. There were none of these identified in the LAR.

3. 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 as "Complies by Previous NRC Approval" or "Complies by Previous Approval" in LAR Attachment A).

4. The existing FPP element complies using existing engineering equivalency evaluations (EEEEs) whose bases remain valid and are of sufficient quality (noted as "Complies with Use of EEEE" in LAR Attachment A).

5. The existing FPP element does not comply with the requirement, but the licensee is requesting specific NRC approval to use a PB method in accordance with 10 CFR 50.48(c)(2)(vii) (noted as "Submit for NRC Approval" in LAR Attachment A).

6. The existing FPP element does not comply with the requirement but will be in compliance with the completion of a required action as part of implementation of the new FPP (noted as "Will Comply with the Use of Commitment" or "Action Required Prior to Bases Being Acceptable for Transition" in LAR Attachment A). These actions are identified as implementation items in LAR Attachment S (Reference 16), which is discussed in SE Section 2.7. This category was not included in NEI 04-02.

For existing DBNPS FPP elements that comply by previous NRC approval, the licensee modified the process outlined in Appendix B, "Detailed Transition Assessment of FPP," of

NEI 04-02. For these elements, rather than providing excerpts from both the associated submittal and approval documents, the licensee provided only an excerpt from the NRC approval document as a part of the compliance basis statement if the excerpt included sufficient information to fully understand the basis for previous approval.

The NRC staff has determined that the process the licensee used to document compliance with the NFPA 805, Chapter 3, requirements, is acceptable. The licensee followed the compliance strategies identified in the NRC-endorsed NEI 04-02 guidance, with some modifications. The process provided an organized structure to document how the licensee will comply with each attribute of NFPA 805, Chapter 3, and the licensee provided significant detail on how the program will meet the requirements. The use of additional categories aside from "Complies," provided clarification regarding the acceptability of particular attributes.

In LAR Section 4.2.2, "Existing Engineering Equivalency Evaluation Transition" (Reference 6), the licensee stated that it evaluated the EEEEs used to demonstrate compliance with the NFPA 805, Chapter 3, requirements to ensure continued appropriateness, quality, and applicability to the current DBNPS plant configuration. The licensee determined that no EEEEs used to support compliance with NFPA 805 required NRC approval. EEEEs were 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 will be conducted using a PB approach to demonstrate that specific plant configurations meet the performance criteria in the standard.

Additionally, the licensee stated in LAR Section 4.2.3, "Licensing Action Transition" (Reference 6), that the bases for previously approved licensing actions that will be transitioned to the new FPP have been evaluated to determine if they remain valid. The results of these licensing action evaluations are provided in LAR Attachment K (Reference 13).

LAR Attachment A provided 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 Category: Complies

For certain NFPA 805, Chapter 3, requirements, as modified by 10 CFR 50.48(c)(2), the licensee determined that the existing FPP element complies with the fundamental FPP element and can be transitioned directly to the RI/PB FPP. In these instances, based on the information provided by the licensee in LAR Attachment A (Reference 6), the NRC staff concludes that the licensee's statements of compliance are acceptable. A more detailed review of the licensee's compliance with Sections 3.2.3(1) and 3.4.1(c) of NFPA 805 is provided below.

Section 3.2.3(1) of NFPA 805

NFPA 805, Section 3.2.3(1), requires procedures to be established for inspection, testing, and maintenance for fire protection systems and features credited by the FPP. Under the Phase 3 discussion in LAR Section 4.6.2, "Overview of Post-Transition NFPA 805 Monitoring Program" (Reference 6), the licensee stated that the EPRI Technical Report (TR) 1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide" (Reference 79), will be used as input for establishing reliability targets, action levels, and monitoring frequency for fire protection systems and features. The methodology in EPRI TR 1006756 is a PB approach

for determining acceptable surveillance frequencies for fire protection systems and features that are different than the surveillance frequencies described in the appropriate NFPA code or standard. The licensee indicated that its current FPP revised the frequency of performing inspection, testing, and maintenance of certain fire protection systems and features using a PB method, and that some surveillance frequencies are different than the frequencies required by the appropriate NFPA 805 codes and standards.

In its October 9, 2017, revision to LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 11) to use the PB methods in EPRI TR 1006756 for establishing inspection, testing, and maintenance frequencies as an alternative to the NFPA 805, Chapter 3, requirements. In addition, Implementation Item DB-2063 in LAR Attachment S (Reference 16), Table S-2, states that the licensee will perform a review of the PB methods described in EPRI TR 1006756 for establishing the appropriate frequencies for inspection, testing, and maintenance procedures, and adjust the site program to address any differences identified. The NRC staff's evaluation of Approval Request 11 is provided in SE Section 3.1.4.11.

Section 3.4.1(c) of NFPA 805

NFPA 805, Section 3.4.1(c), requires the fire brigade leader and at least two brigade members to have sufficient training and knowledge of nuclear safety systems in order to understand the effects of fire and fire suppressants on NSPC. The licensee stated in LAR Attachment A that the fire brigade members are qualified through a training program and that the qualification requirements detailed in qualification manuals include knowledge of plant systems, layout, and general operation, as well as firefighting skills and attack strategy. In response to fire protection engineering (FPE) RAI 02 (Reference 10), the licensee stated that its qualification manuals describe the qualifications for the fire brigade captain and members, and the licensee provided details of the training that is provided. The RAI response also changed the compliance statement for NFPA 805, Section 3.4.1(c), in LAR Attachment A from "Complies" to "Will Comply with the Use of Commitment." Implementation Item DB-2062 in LAR Attachment S, Table S-2 (Reference 16), states that the licensee will update the conduct of operations procedure to clarify the fire brigade qualifications necessary to meet NFPA 805, Section 3.4.1(c). As discussed in SE Section 3.1.1.5, the NRC staff concluded that the licensee's statements of compliance using commitments are acceptable because the proposed actions for achieving compliance will incorporate the provisions of NFPA 805 into the FPP.

3.1.1.2 Compliance Category: Complies by Previous NRC Approval

NFPA 805, Section 3.1, states that previously approved alternatives from the fundamental protection program attributes in NFPA 805, Chapter 3, take precedence over the Chapter 3 requirements. The licensee identified the Chapter 3 requirements where it will comply by previous NRC-approval, as provided in the following documents:

1. Amendment No. 18 to the DBNPS license, including the associated SE, issued on July 26, 1979, which approved the original DBNPS FPP (Reference 21);

2. NRC letter dated July 17, 1980 (Reference 80) approving the use of sprinklers in lieu of fireproof coating of structure steel for certain rooms;

3. NRC SE of the DBNPS FPP issued on May 30,1991 (Reference 22);

4. Exemptions from 10 CFR Part 50, Appendix R, Section III.G, dated November 23, 1982 (Reference 72), April 18, 1990 (Reference 74), and July 21, 2005 (Reference 77), approving alternative shutdown capabilities without a full, fixed fire suppression system in fire areas where the alternative shutdown capabilities are provided;

5. Exemption from 10 CFR Part 50, Appendix R, Section III.G, dated August 20, 1984 (Reference 73), approving fire barriers that do not have 3-hour fire rating;

6. Exemptions from 10 CFR Part 50, Appendix R, Section III.O, dated August 20, 1984, and January 30, 1998 (Reference 75), approving the lack of a lube OCS that can hold the entire lube oil inventory for the RCPs and remote fill locations;

7. Exemptions from 10 CFR Part 50, Appendix R, Sections III.G and III.J, dated April 18, 1990, approving the lack of a 3-hour fire barrier separating redundant trains of safe shutdown (SSD) circuits, embedded conduits, and lack of 8-hour battery backed lights in access and egress routes; and

8. Exemptions from 10 CFR Part 50, Appendix R, Section III.G, dated December 26, 2002 (Reference 76), approving separation of redundant SSD components without a 1-hour fire-rated barrier where less than 20 feet of separation exists.

The guidance in Section 2.3.1 of RG 1.205, Revision 1, states that exemptions or deviations identified as previously approved alternatives to NFPA 805, Chapter 3, will be considered acceptable to the NRC if the licensee demonstrates that the exemption or deviation remains valid. In LAR Attachment A (Reference 6), the licensee identified several sections of NFPA 805, Chapter 3, where it will comply by previous NRC approval. However, for several sections, the licensee did not clearly identify that the exemption or deviation remains valid. In its letter dated October 9, 2017 (Reference 13), the licensee submitted a revised LAR Attachment K that included a new Licensing Action 16, which indicated that for Sections 3.3.4, 3.3.7.1, 3.3.8, 3.5.1, 3.5.2, 3.5.10, 3.5.13, 3.6.1, 3.6.2, and 3.6.4 of NFPA 805 the previous NRC approval cited remains valid. The NRC staff determined that the previous NRC approval cited in Licensing Action 16.03, and the licensee's statement that this approval remains valid, is also applicable to NFPA 805, Section 3.3.1.2(5).

In its response to FPE RAI 05 (Reference 10), the licensee changed its compliance statement for NFPA 805 Sections 3.5.3, 3.7, 3.8.2, and 3.9.1 from "Complies by Previous NRC Approval" to "Will Comply with Use of Commitment." The licensee stated that it will perform a code compliance review in an EEEE. If the design requirement is not met or cannot be demonstrated functionally equivalent, then the licensee will either modify the plant to ensure conformance with the design code or request a deviation from the NRC. In its letter dated November 20, 2018 (Reference 16), the licensee submitted a revised version of LAR Attachment S that included Implementation Items DB-2053 through DB-2057 to update the code of record compliance reviews to the format of EEEEs for NFPA 805, Sections 3.5.3, 3.7, 3.8.2, 3.9.1, and 3.11.3. The NRC staff's review of NFPA 805, Chapter 3, requirements where the licensee will comply by use of commitments is discussed in SE Section 3.1.1.5.

The NRC staff reviewed the information provided by the licensee and finds that the licensee has demonstrated previous NRC approval using suitable documentation. The NRC staff found the licensee's justification of the continued validity of the previously approved alternatives to the NFPA 805, Chapter 3, requirements to be adequate. Therefore, the NRC staff concludes that the previously approved alternatives to the specific NFPA 805, Chapter 3, requirements, as

discussed in the LAR, are acceptable. A more detailed review of the licensee's compliance with Sections 3.3.5.1, 3.3.5.3, 3.3.7.1, 3.8.1, and 3.11.2 of NFPA 805 is provided below.

Section 3.3.5.1 of NFPA 805

NFPA 805, Section 3.3.5.1, requires that wiring above suspended ceilings be kept to a minimum, and, where installed, electrical wiring must 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. In LAR Attachment A, the licensee stated that it complies with NFPA 805, Section 3.3.5.1, by previous approval, which is documented in the NRC SE dated May 30, 1991 (Reference 22). However, in response to FPE RAI 05 (Reference 10), the licensee changed its compliance statement for NFPA 805, Section 3.3.5.1, from "Complies by Previous NRC Approval" to "Complies". The licensee stated in its RAI response that the previous approval was due to the lack of a fixed fire suppression system above the main control room (MCR) suspended ceiling, as required by Appendix A of BTP 9.5-1, but the BTP acceptance criteria does not apply for transition to NFPA 805. The NRC staff determined that the previous approval of lack of a fixed fire suppression system above the MCR suspended ceiling is not needed to demonstrate compliance with NFPA 805, Section 3.3.5.1.

Section 3.3.5.3 of NFPA 805

NFPA 805, Section 3.3.5.3, requires electric cable construction to comply with a flame propagation test acceptable to the NRC. In LAR Attachment A, the licensee stated, in part, that it complies by previous approval, which is documented in the NRC SE dated May 30, 1991 (Reference 22). However, in response to FPE RAI 05 (Reference 10), the licensee deleted the statement for NFPA 805, Section 3.3.5.3, that it "Complies by Previous NRC Approval." Additionally, in response to FPE RAI 04.a (Reference 10), the licensee stated that some electrical cables installed at DBNPS were not tested to the standards referenced in NFPA 805 or the associated FAQ 06-0002 (Reference 81).[5] The licensee also revised LAR Attachment L, Approval Request 2, to state that the electrical cable material properties have been evaluated in FM, PRA, and in the FREs to determine the locations where fire suppression or detection is required for risk reduction or DID. The NRC staff's review of Approval Request 2 is documented in SE Section 3.1.4.2.

Section 3.3.7.1 of NFPA 805

NFPA 805, Section 3.3.7.1, requires the storage of flammable gas to be located outdoors, or in separate detached buildings, so that a fire or explosion will not adversely impact systems, equipment, or components important to nuclear safety. Section 3.3.7.1 also requires NFPA 50A (Reference 53) to be followed for hydrogen storage. In LAR Attachment A, the licensee stated it complies with the NFPA 805 requirement that flammable gas storage is located outdoors or in separate detached buildings. The licensee also stated that it complies by previous NRC approval, which is documented in the NRC SE dated May 30, 1991 (Reference 22). In response to FPE RAI 09 (Reference 10), the licensee stated:

> The bulk storage of hydrogen is located outside in the north yard area, and propane tanks are located outside in the west yard. Hydrogen cylinder storage is also located outside in the yard, west of the low level radwaste storage facility building.

[5] FAQ 06-0002 was incorporated into NEI 04-02, Revision 2.

The FPE RAI 09 response also states that the hydrogen storage installation was evaluated to the requirements of NFPA 50A-1973, as described in DBNPS letters dated July 31, 1989 (Reference 82) and October 11, 1989 (Reference 83). The licensee stated that the NRC staff's acceptance of the NFPA 50A installation was documented in NRC SE dated May 30, 1991 (Reference 22). In its October 9, 2017, revision to LAR Attachment K (Reference 13), the licensee indicated that the basis for previous approval remains valid (Licensing Action 16.02). The NRC staff concludes that the licensee adequately demonstrated that it complies with the NFPA 805 requirement by previous approval, as documented in the May 30, 1991, NRC SE, and that the basis for the previous approval remains valid.

Section 3.8.1 of NFPA 805

NFPA 805, Section 3.8.1, requires fire alarm initiating devices to be installed in accordance with NFPA 72 (Reference 56). In LAR Attachment A, the licensee indicated that the NRC staff previously approved deviations from the related NFPA standards for the fire detection, alarm, and signaling systems. In response to FPE RAI 05 (Reference 10), the licensee stated that the code of record for the fire alarm system is NFPA 72, and the licensee deleted the statement that it complies with NFPA 805, Section 3.8.1, by previous NRC approval. The RAI response stated that compliance with NFPA 72 is documented in an EEEE, which determined that the fire alarm system complies with or is functionally equivalent to the code. The installed fire alarm panel is not the same one previously evaluated by the NRC, and the EEEE review supersedes the previous code review. The NRC staff concludes the licensee adequately explained how it will meet the NFPA 805, Section 3.8.1, requirement with an engineering evaluation.

Section 3.11.2 of NFPA 805

NFPA 805, Section 3.11.2, requires that fire barriers required by NFPA 805, Chapter 4, include a specific fire-resistance rating and be designed and installed to meet this rating using assemblies qualified by fire tests. In LAR Attachment A, the licensee stated that for several rooms where the structural steel could not be adequately fire proofed it will comply by previous NRC approval, as documented in an NRC letter dated July 17, 1980 (Reference 80). Overhead sprinklers have been provided in these rooms, rather than applying the fireproof coating. In LAR Table 4-3, the licensee identified fire protection systems and features that are credited to meet the NSPC in NFPA 805, Chapter 4. In its response to FPE RAI 07 (Reference 10), the licensee revised LAR Table 4-3 to indicate that the automatic fire suppression system in rooms 208, 236, 303, 402, 405, and 427 will be transitioned to the new FPP by a new Licensing Action 15. In its letter dated October 9, 2017 (Reference 13), the licensee revised LAR Attachment K to include the new Licensing Action 15, which indicates that the previous NRC approval remains valid. The NRC staff concludes that the licensee adequately demonstrated that it complies with NFPA 805, Section 3.11.2, for these rooms by prior NRC approval, as documented in the NRC letter dated July 17, 1980, and that the basis for the previous approval remains valid.

3.1.1.3 Compliance Category: Complies with Use of EEEEs

For certain NFPA 805, Chapter 3, requirements, the licensee demonstrated compliance with the fundamental FPP element using EEEEs. The NRC staff reviewed the licensee's statements of continued validity, quality, and appropriateness for the EEEEs, and concludes that the licensee's statements of compliance using EEEEs are acceptable. A more detailed review of the licensee's compliance with Section 3.3.7.2 of NFPA 805 is provided below.

Section 3.3.7.2 of NFPA 805

NFPA 805, Section 3.3.7.2, requires outdoor high-pressure flammable gas storage containers to be located so that the long axis is not pointed at buildings. In LAR Attachment A (Reference 6), the licensee stated that the hydrogen and propane storage tanks are oriented with the long axis toward buildings. The licensee stated that the flammable gas storage orientation was evaluated for compliance with NFPA 50A-1973 (Reference 53) and NFPA 58-2004 (Reference 55), and concluded that the orientation of the tanks is acceptable. However, NFPA 805, Section 3.3.7.2, is not associated with code compliance. Additionally, NFPA 50A and NFPA 58 do not contain directional requirements that are similar to the requirements of NFPA 805, Section 3.3.7.2.

In response to FPE RAI 08 (Reference 10), the licensee stated that the EEEE for NFPA 805, Section 3.3.7.2, was revised to clarify that the existing plant configuration is functionally equivalent to the NFPA 805 requirement and that approval is not required. The licensee based its determination of functional equivalency on the application of safe distances from structures for flammable gases in the specific codes (e.g., NFPA 50A-1973 and NFPA 58-2004), substantial securement of the tanks, and installation of relief devices. The licensee also provided the stand-off distances for the hydrogen trailer (239 feet to nearest building) and propane tanks (43 feet to nearest building), which exceeded the minimum required distances in NFPA 50A (25 feet) and NFPA 58 (10 feet), respectively. The licensee also stated that the propane tanks are secured to concrete foundations and the hydrogen cylinders are secured between two heavy steel bulkheads that are mounted to the frame of the trailer.

In response to FPE RAI 08.01 (Reference 12), the licensee stated, in part:

> The long axis end of a horizontal storage container is structurally the weaker point of a cylinder's design; therefore, if it were to potentially rupture it would be most likely to propel material along the length of the axis. The NFPA codes (30, 55 [which superseded 50A], and 58) inherently recognize this and recommend safe distances from structures and property lines based on consideration of this along with other factors.
>
> FENOC has determined that a fire or potential storage container rupture will not result in penetration of an adjacent wall based on a consideration of factors such as: separation distance, building construction and other physical obstacles, storage container size and pressure, relief devices, and tank anchorage.

In addition, the licensee stated:

> Exceeding the minimum distances required by NFPA 55 and NFPA 58 by factors of nine and four, respectively, results in a conclusion that any nearby structures would be safe from an exposure fire or missiles projected due to a tank rupture. In addition, the EEEE addresses several other considerations contributing to the conclusion that the location of the storage tanks relative to the nearest exposed PRA fire compartment and equipment is a minimal risk for radiant heat exposure and for missiles. Those considerations include:
>
> - The tanks are securely mounted to steel bulkheads or to concrete, thereby reasonably precluding the tanks from becoming a missile,

- The tanks have pressure relief devices that reasonably preclude the tanks from overpressurization and becoming potential missiles,
- Surfaces of the nearest exposed structures are of concrete and steel construction, and
- The hydrogen tanks have additional obstacles between the tanks and building that would absorb potential missile damage.

The NRC staff reviewed the information provided by the licensee and considered the guidance in FAQ 06-0008 (Reference 58) regarding the use of engineering evaluations to demonstrate functional equivalency. The NRC staff concludes that the licensee adequately demonstrated functional equivalency to NFPA 805, Section 3.3.7.2, and that the location of the storage tanks poses a minimal risk to exposed equipment and buildings located along the long axis of the tanks.

3.1.1.4 Compliance Category: Submit for NRC Approval

In accordance with 10 CFR 50.48(c)(2)(vii), the licensee requested approval to use several PB methods to demonstrate compliance with fundamental FPP elements. The NRC staff review of these methods is provided in SE Section 3.1.4. The following NFPA 805 sections were identified in LAR Attachment A as complying via this method:

- Section 3.3.5.1 requires wiring above suspended ceilings to be kept to a minimum, and, where installed, the wiring shall be listed for plenum use, routed in armored cable, routed in metallic conduit or routed in cable trays with solid metal top and bottom covers. The licensee requested NRC approval of a PB method to justify the limited use of low power data and communications cables above suspended ceilings which are neither plenum-rated nor routed in armored cable, metal conduit, or trays with metal covers (SE Section 3.1.4.1).
- Section 3.3.5.3 requires electrical cable construction to comply with flame propagation tests acceptable to the NRC. The licensee requested NRC approval of a PB method used to justify the type of cable insulation used throughout the plant (SE Section 3.1.4.2).
- Section 3.3.12(1) requires the OCS for each RCP to be capable of collecting lubricating oil from all potential pressurized and nonpressurized leakage sites in each RCP oil system. Section 3.3.12(4) requires that leakage points on a RCP motor to be protected to include but not be limited to the lift pump and piping, overflow lines, oil cooler, oil fill and drain lines and plugs, flanged connections on oil lines, and the oil reservoirs, where such features exist on the RCPs. The licensee requested NRC approval for the use of a PB method to justify the RCP lube oil misting that is not captured by the RCP lube OCS (SE Section 3.1.4.3).
- Section 3.5.5 requires each fire pump and its driver and controls to be separated from the remaining fire pumps and from the rest of the plant by rated fire barriers. The licensee requested NRC approval of a PB method to justify the remote start circuit separation configuration of the remote control circuits to each fire pump (SE Section 3.1.4.4).

- Section 3.5.14 requires all fire protection water supply and fire suppression system control valves to be under a periodic inspection program and to be supervised by one of the methods specified in this section. The licensee requested NRC approval of a PB method for the use of curb box valves in the fire protection water supply that do not meet the requirement for supervision (SE Section 3.1.4.5).

- Section 3.6.3 requires the proper type of hose nozzles to be supplied to each power block area based on the area fire hazards. The licensee requested NRC approval of a PB method to justify the use of adjustable fog nozzles at hose stations which are located outside the high-voltage electrical switchgear rooms (SE Section 3.1.4.6).

- Section 3.9.4 requires diesel-driven fire pumps to be protected by automatic sprinklers. The licensee requested NRC approval for the use of a PB method to justify the lack of automatic sprinkler coverage over the diesel-driven fire pump (SE Section 3.1.4.7).

- Section 3.6.1 requires Class III standpipe and hose systems to be installed for all power block buildings. The licensee requested NRC approval for the use of a PB method to justify the lack of a fire hose standpipe system in the containment building (SE Section 3.1.4.9).

- Section 3.3.8 requires the bulk storage and use of flammable and combustible liquids to, at a minimum, comply with NFPA 30. The licensee requested NRC approval for the use of a PB method to justify several deviations from NFPA 30 related to the bulk storage of flammable and combustible liquids, including:

    - the diesel fuel tank inside the station blackout (SBO) diesel generator building (SE Section 3.1.4.10),
    - the lube oil storage tank (SE Section 3.1.4.12),
    - the emergency diesel generator (EDG) fuel oil day tanks in the auxiliary building (SE Section 3.1.4.13), and
    - the diesel fuel oil tank in the emergency feedwater facility (EFWF) (SE Section 3.1.4.14)

- Section 3.2.3(1) requires the inspection, testing, and maintenance of fire protection systems and features credited by the FPP. The licensee requested NRC approval to use a PB method (EPRI TR 1006756) to establish acceptable surveillance frequencies that are different than the frequencies described in the appropriate NFPA code (SE Section 3.1.4.11).

In the original LAR Attachment L (Reference 6), the licensee requested approval (Approval Request 8) of changes to fire protection elements described in a previously approved exemption from 10 CFR Part 50, Appendix R, requirements for separation of equipment required for SSD. However, Approval Request 8 was associated with a deviation from the deterministic separation requirements in NFPA 805, Chapter 4, and cannot be approved under 10 CFR 50.48(c)(2)(vii) because this regulation applies only to NFPA 805, Chapter 3, requirements. In response to SSA RAI 12 (Reference 10), the licensee withdrew Approval Request 8, and stated it will not transition the exemption for component cooling water pump separation described in LAR

Attachment K, Licensing Action 2. The licensee further stated that a variance from deterministic requirements (VFDR) was developed to address the potential for a fire in compartment T-01 to affect all component cooling water pumps. In its November 20, 2018, letter (Reference 16), the licensee stated that the FRE of the VFDR determined that a modification is not necessary and that the risk of not performing a modification is minimal.

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

3.1.1.5 Compliance Category: Will Comply with the Use of Commitment

For several NFPA 805, Chapter 3, requirements, the licensee stated that compliance with the fundamental FPP element will be achieved using commitments. These commitments are identified as implementation items in Table S-2 of LAR Attachment S (Reference 16). The NRC staff concludes that the licensee's statements of compliance using commitments are acceptable because the proposed actions for achieving compliance will incorporate the provisions of NFPA 805 into the FPP. However, to ensure that these actions are completed, the NRC will require the implementation items in Table S-2 to be completed during implementation of the new FPP as part of the FPP license condition (see SE Section 2.7).

3.1.1.6 Compliance Category: Action Required Prior to Bases Being Acceptable for Transition

In LAR Attachment A (Reference 6), the licensee identified fire compartment EF-01 (emergency feedwater) as a new construction and identified that a compliance basis review against the requirements of NFPA 805, Sections 3.8.2 and 3.9.1, will follow design and construction of the EFWF. This plant modification is identified as Item DB-1421 in Table S-1 of LAR Attachment S (Reference 16). The NRC staff concludes that the licensee's statement of compliance is acceptable because the action for achieving compliance will incorporate the provisions of NFPA 805 in the FPP. However, to ensure that these actions are completed, the NRC will require Implementation Item DB-1421 in Table S-1 to be completed during implementation of the new FPP as part of the FPP license condition (see SE Section 2.7).

3.1.1.7 Conclusion Regarding Compliance with Chapter 3 Requirements

As discussed above, the NRC staff reviewed the results of the licensee's assessment of the proposed DBNPS RI/PB FPP against the NFPA 805, Chapter 3, fundamental FPP elements and minimum design requirements, as modified by the exceptions, modifications, and supplementation in 10 CFR 50.48(c)(2). Based on this review, the NRC staff concludes that the RI/PB FPP is acceptable with respect to these requirements because the licensee:

(1) Used an acceptable process to determine the state of compliance with each of the applicable NFPA 805, Chapter 3, requirements.

(2) Adequately demonstrated that DBNPS will comply with each NFPA 805, Chapter 3, requirement by describing how:

- the existing FPP complies with the requirement and can be directly transitioned to the new FPP;

- the new FPP will comply with the requirement after the completion of the modification LAR Attachment S (Reference 16), Table S-1 or an implementation item listed in LAR Attachment S, Table S-2;
- the new FPP will comply via previous NRC staff approval of an alternative to the requirement;
- the new FPP will comply with the requirement using an EEEE;
- the new FPP will comply using a PB method that the licensee has requested NRC staff approval, in accordance with 10 CFR 50.48(c)(2)(vii), with this LAR; or
- it will comply using a combination of the above methods.

3.1.2 Identification of Power Block

NFPA 805 defines the power block as: "Structures that have equipment required for nuclear plant operations." The NRC staff reviewed the DBNPS structures identified in LAR Attachment I (Reference 6) as comprising the power block for the proposed DBNPS FPP. The power block at DBNPS includes the auxiliary building, containment building, EFWF, intake structure, plant office building, turbine building, water treatment building, and outside areas. The NRC staff concludes that the licensee appropriately evaluated the structures and equipment at DBNPS, and adequately identified those structures that fall under the NFPA 805 definition for the power block.

3.1.3 Plant-Specific Treatments or Technologies

Generic Letter 2006-03 (Reference 49) identified concerns regarding the use of Hemyc and MT fire barrier systems used in nuclear power plants to protect circuits and other electrical components. In a letter dated June 8, 2006 (Reference 84), the licensee stated that Hemyc and MT fire barrier materials are not used at DBNPS. Therefore, the issues identified in Generic Letter 2006-03 are not applicable to DBNPS.

3.1.4 Approval Requests for 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 to use 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. According to 10 CFR 50.48(c)(2)(vii), the NRC staff may approve the use of the PB method in this circumstance, if the staff determines that the PB 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 DID (fire prevention, fire detection, fire suppression, mitigation, and post-fire SSD capability).

As discussed in SE Section 3.1.1.4, the licensee requested NRC approval of PB methods to demonstrate an equivalent level of fire protection for certain NFPA 805, Chapter 3, elements. The licensee's requests are identified in the October 9, 2017, revision to LAR Attachment L (Reference 13) as Approval Requests 1–14. The NRC staff evaluation of these proposed methods is provided below. The revised approval requests contain annotated text, where text that is struck through is considered deleted and text that is underlined was added. When quoting text from the approval requests, this SE does not show the annotations and only indicates the intended revision (e.g., quoted text does not include struck though text).

The NRC staff considered the nuclear safety and radioactive release goals, objectives, and performance criteria in NFPA 805, Chapter 1 (see SE Section 2.0) in its review of the licensee's requests to use PB methods to demonstrate an equivalent level of fire protection for the NFPA 805, Chapter 3, elements. In addition, the NRC staff considered the three echelons of DID in NFPA 805, Section 1.2 (see SE Section 2.0) in its review:

(1) preventing fires from starting;

(2) rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting damage; and

(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 safety functions from being performed.

3.1.4.1 Approval Request 1: Performance-Based Approach for NFPA 805, Section 3.3.5.1

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 1) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.3.5.1, requirement that wiring installed above suspended ceilings 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. Specifically, the licensee requested approval for the limited use of low power data and communication cables above suspended ceilings which are neither plenum-rated nor routed in armored cable, metal conduit, or trays with metal covers.

Within the power block at DBNPS, the licensee stated in Approval Request 1 that fire area CC-01 (old radiologically restricted area access and chemistry lab areas) is the only area with cables routed above suspended ceilings that are not rated for plenum use and not enclosed. The licensee further stated:

Nuclear Safety and Radiological Release Performance Criteria:

The location of non-enclosed or non-plenum rated low power data and communication cable above the suspended ceiling of CC-01 does not pose a significant fire hazard. The low power cable is not susceptible to shorts that would result in a fire and does not constitute an ignition source capable of challenging nuclear safety. Cables performing a nuclear safety function are routed and protected in metallic conduit in this space. Therefore, the presence of non-plenum-rated cables located above suspended ceilings does not affect nuclear safety performance criteria.

> The presence of non-plenum-rated cables located above the suspended ceiling of fire compartment CC-01 has no impact on fire suppression activities, nor impact on radiological release performance criteria. The radiological review was performed based upon the potential location of radiological concerns and is not dependent on the type of wiring or location of suspended ceilings. The cables do not change the results of the radiological release evaluation that concluded potentially contaminated water is contained and smoke is monitored. The cables do not add additional radiological materials to the fire compartment. The existing use of non-enclosed or non-plenum-rated low voltage wiring above suspended ceilings at DBNPS has no impact on nuclear safety and radiological release performance criteria.
>
> **Safety Margin and Defense-in-Depth:**
>
> The non-enclosed or non-plenum rated low power data and communication cable above suspended ceilings carry insufficient energy to self-ignite. Such cables do not compromise automatic or manual fire suppression functions nor adversely impact safety margin given the absence of a credible ignition source and the low likelihood of occurrence of fire. The cables have been analyzed in their current configuration. The amount of non-rated and non-enclosed cable above the ceiling in CC-01 is minor and does not present a significant fire hazard. In addition, the cable used for nuclear safety are routed and protected in metallic conduit. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

In Approval Request 1, the licensee addressed all three echelons of DID. For DID Echelon 1, the licensee stated that "the subject low power data and communication cable does not carry sufficient energy to self-ignite, which prevents potential fires from starting." For DID Echelon 2, the licensee stated that pre-fire plans and training ensure that the fire brigade will promptly respond to a fire, and that fire extinguishers and hose stations are located within or nearby the fire areas. For DID Echelon 3, the licensee stated:

> Finally, the DBNPS fire program provides robust fire barriers and circuit separation of redundant trains of essential safety related equipment that preserve one train free of fire damage to enable safe shutdown of the plant and preserve DID echelon 3. Limited fixed ignition sources and the low potential energy in the exposed non-shutdown data and communication cables located above the suspended ceiling make it unlikely for a significant fire event to originate in the space above the suspended ceilings. In the unlikely event of a fire, it would not be capable of challenging the cable contained in the metallic conduits that support the nuclear safety performance criteria, which further supports echelon 3.

The licensee also stated that procedure changes will be made to ensure future cable installations above suspended ceilings will be listed for plenum use or enclosed per NFPA 805, Section 3.3.5.1, which will support DID Echelon 1. These procedural changes are included as Implementation Item DB-1964 in LAR Attachment S (Reference 16).

The NRC staff reviewed the information provided by the licensee related to Approval Request 1 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection

DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 1 is an acceptable alternative to the corresponding NFPA 805, Section 3.3.5.1, requirement.

3.1.4.2 Approval Request 2: Performance-Based Approach for NFPA 805, Section 3.3.5.3

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 2) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.3.5.3, requirement that electrical cable construction comply with a flame propagation test acceptable to the NRC. Specifically, the licensee requested approval for recognition that the type of cable insulation throughout the plant meets the intent of NFPA 805, Section 3.3.5.3.

The licensee stated that the cables installed as part of the original construction of DBNPS predate the issuance of the Institute of Electrical and Electronics Engineers (IEEE) Standard 383, "Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations" (Reference 85). The NRC staff previously approved (Reference 22) alternative testing measures for DBNPS that were used to qualify cables based on the levels of fire protection (e.g., fire detection, fire suppression, and fire barriers) provided for SSD systems and hazardous areas. The licensee further stated that the majority (over 90 percent) of the cables in trays at DBNPS are equivalent to qualified cables because they have thermoset insulation that limits fire propagation along the cable.

The licensee stated that cables categorized as having thermoplastic insulation were included into the fire models. Less than 2 percent of the cables in trays at DBNPS were identified as thermoplastic (polyvinylchloride or Teflon) insulated. The licensee could not determine the actual cable insulation type for approximately 4 percent of the cables, so they were conservatively assumed to be thermoplastic insulated. The licensee also stated (Reference 17) that DBNPS has cables manufactured by Kerite, and, consistent with the guidance in FAQ 08-0053, these cables were modeled using the damage thresholds for thermoplastic cables and the fire growth and spread values for thermoset cables. The known thermoplastic insulated cables at DBNPS are used in low-voltage applications that do not carry enough electrical energy for self-ignition. These cables were found only in non-essential trays and are routed separately from safety-related cables. The licensee stated: "Cables in conduit are not a concern and not considered as an ignition source, as cables in conduit do not significantly contribute to fire growth and spread because they are enclosed, have limited oxygen, and less surface area for fire spread."

The licensee stated in Approval Request 2:

> **Nuclear Safety and Radiological Release Performance Criteria:**
>
> Thermoplastic insulated cables are used primarily in non-safety and low voltage applications; therefore, they are not considered an ignition source. The thermoplastic insulated cables identified by the DBNPS Condition Report 10-74188 were found only in non-essential cable trays. These cables are routed separately from safety-related cables. In the event of an externally initiated fire that may ignite these cables, the separation will minimize the potential of flame spread to safety-related cables. Therefore there is no impact on the nuclear safety performance criteria.

The radiological review was performed based on the potential location of radiological concerns and is not dependent on the type or location of thermoplastic cables. The results of the radiological release evaluation concluded that potentially contaminated water is contained and smoke is monitored. The thermoplastic cables do not add additional radiological materials to the areas or challenge systems boundaries. Therefore, non-qualified thermoplastic (this includes cables of unknown construction) cable currently in use at DBNPS has no impact on radiological release performance criteria.

Safety Margin and Defense-in-Depth:

Thermoset insulation is used in the majority of cables in the plant. This type of insulation will char and retain its shape when heated, limiting flame propagation. The cables not confirmed to be of thermoset insulation account for less than 10% of all cables within the plant, with the majority of these thermoplastic cables being low-voltage and non-safety related. The non-thermoset cables are all treated as if they are of thermoplastic construction. Damage to these cables is not expected to impact safety-related cables due to cable separation. Flame spread to adjacent cable trays in high density safety-related areas is reduced by the use of solid-bottom trays. Some cable trays enclose the cable with a layer of ceramic fiber on top; thereby further reducing the likelihood of fire spread to additional cable trays.

The detailed fire models as well as the probabilistic risk assessment incorporated the information from DBNPS Condition Report 10-74188 to identify and conservatively account for thermoplastic cable impacts. The electrical cable material properties were included in the fire modeling evaluations, in the PRA models, and in the fire risk evaluations to determine the locations where additional fire protection features (such as fire suppression and/or detection) are required for risk reduction or defense-in-depth. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

In Approval Request 2, the licensee addressed all three echelons of DID. The licensee stated that Echelon 1 is met through plant fire prevention procedures and is not affected by use of non-qualified thermoplastic cables because these cables are used primarily in nonsafety and low-voltage applications and are not considered an ignition source. The licensee further stated that these cables do not affect Echelons 2 or 3 because they are found only in nonessential trays and are routed separately from safety-related cables. Therefore, these cables do not directly result in compromising automatic fire suppression systems, manual fire suppression functions, or post-fire SSD capability.

The NRC staff reviewed the information provided by the licensee related to Approval Request 2 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 2 is an acceptable alternative to the corresponding NFPA 805, Section 3.3.5.3, requirement.

3.1.4.3 Approval Request 3: Performance-Based Approach for NFPA 805, Section 3.3.12

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 3) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.3.12(1) and 3.3.12(4), requirements. Section 3.3.12(1) requires the OCS for each RCP to be capable of collecting lubricating oil from all potential pressurized and nonpressurized leakage sites in each RCP oil system. Section 3.3.12(4) requires that leakage points on an RCP motor be protected, including the lift pump and piping, overflow lines, oil cooler, oil fill and drain lines and plugs, flanged connections on oil lines, and the oil reservoirs. Specifically, the licensee requested approval of a PB method that has not been previously approved by the NRC staff for RCP oil mist that is not captured within the OCSs.

The NRC staff previously considered the RCP OCS and remote oil fill system design with the exemptions granted on August 20, 1984 (Reference 73), and January 30, 1998 (Reference 75). In Licensing Action 12 (Reference 13), the licensee requested these exemptions to be transitioned to the RI/PB FPP, which is discussed in SE Section 3.5.1.3. However, RCP oil mist was not considered with these exemptions.

The reactor coolant system (RCS) at DBNPS is configured with two reactor coolant loops, which are partitioned at various levels by reinforced concrete walls. Each reactor coolant loop contains two nonsafety-related RCPs, a steam generator, and associated piping. The OCSs are constructed of oil drip pans with spray shields and enclosures surrounding the RCP lubricating oil system and are designed to prevent the RCP lubricating oil system from becoming a potential ignition source. Enclosures surround the oil cooler, fill pipe, and bearing lift pump, and each reactor coolant loop has a single oil collection tank located and sized to handle the total lube oil inventory from one of the two RCP motors in the loop. The enclosures are designed to contain oil from leaks or pipe failures and drains to a vented collection tank.

The licensee stated that some oil mist will be created as part of normal RCP motor operation. The OCS cannot contain this mist because the system does not completely enclose the RCP motor since adequate air circulation is required for safe motor cooling. The licensee further stated that the quantity of oil that may be found in areas of the containment due to the RCP oil vapor mist is very small and does not contribute to any significant fire loading nor create potential fire propagation potential between fire compartments. Nearby hot surfaces of the RCS piping are protected by seismically designed mirror insulation such that any spilled lube oil would contact only outer surfaces of the insulation, which have temperatures below the lube oil flash point. The licensee also stated that sump and splash shields would be capable of preventing a fire during normal and design basis accident conditions.

The licensee stated in Approval Request 3:

> Historically, there have been no fires attributed to oil misting based on normal operation in the industry. Fires have occurred due to oil leakage from equipment failure such as cracked welds on piping or inadequate collection pan design. DBNPS does not have a history, since its licensing for commercial operation was issued on April 22, 1977, of significant oil loss from the RCPs as a result of oil misting or oil leakage that is not contained by the properly designed and installed oil leakage collection system.

The licensee further stated in Approval Request 3:

> **Nuclear Safety and Radiological Release Performance Criteria:**
>
> The nuclear safety performance criteria are met because RCPs are available as necessary, and the RCPs are not required to achieve or maintain post-fire SSD. The radiological release performance criteria are met because (1) the containment vessel during power operations is an environmentally sealed radiological area, (2) the potential for oil mist from the RCPs does not change the radiological release evaluation performed for each fire zone where potentially contaminated water and smoke is contained and monitored, (3) the oil mist does not add additional radiological materials to the area or challenge systems boundaries that contain such materials, and (4) the fire brigade control of water runoff and smoke' is not hindered because of the existence of the misting.
>
> **Safety Margin and Defense-in-Depth:**
>
> RCP oil mist resulting from normal operation will not adversely impact the ability of the plant to achieve and maintain fire safe shutdown, even if ignition occurs. There are redundant RCPs available as necessary, and RCPs are not required to achieve and maintain safe and stable conditions. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

In Approval Request 3, the licensee addressed all three echelons of DID. The licensee stated:

> Echelon 1 is maintained by the OCS and by the RCP motor design, and is not affected by this configuration. The introduction of small amounts of oil misting does not affect Echelons 2 and 3 as oil misting will not serve as either an ignition source or a secondary combustible. Furthermore, Echelon 3 is preserved by the redundant reactor coolant loop. The oil misting does not result in compromising fire detection, manual fire suppression functions, or post-fire safe shutdown capability.

The NRC staff reviewed the information provided by the licensee related to Approval Request 3 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 3 is an acceptable alternative to the corresponding NFPA 805, Section 3.3.12(1) and 3.3.12(4), requirements.

3.1.4.4 Approval Request 4: Performance-Based Approach for NFPA 805, Section 3.5.5

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 4) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.5.5, requirement that each fire pump and its driver and controls be separated from the remaining fire pumps and from the rest of the plant by rated fire barriers. Specifically, the licensee requested approval for the circuit separation configuration of the remote start control circuits for each fire pump at DBNPS.

DBNPS has both an electric fire pump and a diesel-driven fire pump, which supply water to the fire protection systems in the structures within the power block and outlying buildings. Each of these pumps can supply the largest sprinkler system plus a hose stream allowance. Additional capability to supply the fire water system using an onsite or municipal portable fire water pumper is available. The fire pumps are designed to automatically start upon a drop in fire-main water pressure caused by a fire suppression system actuation or the use of a fire hose in manual firefighting. In addition, remote start switches for each fire pump are located in the MCR.

The licensee stated in Approval Request 4 that the remote start circuit cables for the fire pumps are not separated from each other by rated fire barriers throughout their routing. The licensee stated that the design is in conformance with NFPA 20-1974, "Standard for the Installation of Centrifugal Fire Pumps" (Reference 86), and the previous BTP 9.5-1 licensing commitments. The licensee further stated that the configuration meets the intent of NFPA 805, Section 3.5.5.

The cables for the remote start switches are routed from the MCR (fire compartment FF-01) to the cable spreading room (fire compartment DD-01) through penetrations in the floor of the MCR. From the cable spreading room, the cables are routed through separate conduits embedded in concrete to fire compartment BG-01, and from there the cables go to their respective pumps in separate fire compartments. The circuits are installed as separate channels and are not routed through the same raceways. The circuits are routed through cable trays, except in short sections where the cables are routed in conduits. Within these rooms, the cable trays are configured with solid bottoms with a layer of ceramic fiber blanket on the top.

In Approval Request 4, the licensee stated that a fire that causes a short to ground on the remote start cable for either fire pump could result in the loss of the associated pump's starting circuits. If a short to ground were to occur on the remote start cable for the electric fire pump, the licensee stated that this would not affect the ability to perform an emergency start of the pump locally. If a short to ground were to occur on the remote start cable for the diesel fire pump, the licensee stated that under certain conditions the start circuit for the diesel-driven fire pump would be disabled, preventing the local starting of the pump. However, if a fire pump started upon a drop in water pressure and the remote start circuit is subsequently damaged, the fire pump will continue to run.

The licensee stated that a fire may cause a loss of both fire pumps in fire compartments BG-01, DD-01, and FF-01. The licensee stated that the fire pumps and controls are in compliance with NFPA 805, Section 3.5.5, in the remaining fire compartments of the plant. In response to FPE RAI 06.a (Reference 11), the licensee stated that these are the only fire compartments that contain control cables for both the electric and diesel-driven fire pumps. The fire scenarios in these compartments that could damage the control circuits do not credit water-based manual or automatic fire suppression, except for one fire scenario for fire compartment DD-01. For that fire scenario, the licensee stated that damage is expected to be limited to a single fire pump remote start circuit.

In response to FPE RAI 06.a, the licensee also stated that credit for fire suppression sprinklers in fire compartment DD-01 is not needed to compensate for a lack of structural steel fireproofing. Implementation Item DB-1825 in LAR Attachment S (Reference 16) indicates that the licensee will revise the EEEEs to remove credit for sprinklers in fire compartment DD-01. The RAI response also stated that the fire model for fire compartment BG-01 credits water-based fire suppression, but the fire scenarios do not result in damage to the fire pump control circuits. The fire protection DID in the FREs for BG-01, DD-01, and FF-01 will be updated by

the licensee to limit credit of fire hydrants and hose stations and, where necessary for risk mitigation, use of portable extinguishers will be credited. Based on the information provided by the licensee, the NRC staff concludes that a water-based fire suppression system is not required for fire compartments BG-01, DD-01, and FF-01 to meet the NSPC or DID.

In response to FPE RAI 06.b (Reference 11), the licensee stated that the detailed fire models for fire compartments BG-01 and DD-01 take credit for all cable trays having a solid metal bottom and a ceramic fire layer on the top to prevent them from being ignited during a fire. The licensee also stated that these passive fire protection features preclude additional heat release contribution of the cable trays to the associated fire scenarios and limit the damaging zone of influence (ZOI). Additionally, the licensee stated that these features limit the possibility of simultaneous damage to both channels of cable trays that route the fire pump cables.

The licensee stated in Approval Request 4:

> **Nuclear Safety and Radiological Release Performance Criteria:**
>
> The unlikely event of a fire in Fire Compartment BG-01, DD-01, or FF-01 that renders the remote start circuits for both fire pumps inoperable would not affect nuclear safety functions since the fire pumps are not relied upon for these fire scenarios and the nuclear safety performance goals of NFPA 805 Section 1.5 are still achieved with available safe shutdown equipment. In the event of damage to both fire pumps' starting circuits, the ability to perform a local emergency start of the electric fire pump would not be affected. Also, offsite agencies are available to provide firefighting assistance and portable equipment is available to pressurize and supply the fire protection water supply system. Therefore, there is no impact on the nuclear safety performance criteria.
>
> The radiological release reviews were performed based on the fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the location of the pump controls. The location of the fire pumps' remote start cables does not change the radiological release evaluation performed that concluded that potentially contaminated water is contained and smoke is monitored. The location of the remote start circuit cables for the fire pumps do not add additional radiological materials to the area or challenge systems boundaries. Therefore, remote start circuit cables for the fire pumps currently in use at DBNPS has no impact on radiological release performance criteria.
>
> **Safety Margin and Defense-in-Depth:**
>
> The lack of rated fire barriers between the remote start circuits for the electric fire pump and the diesel fire pump does not negate the ability to suppress a fire using credited means. Even in the unlikely event that both sets of remote start cables are damaged and impair both fire pumps' starting circuits prior to automatic start due to low fire main pressure, the ability to perform an emergency start of the electric fire pump or provide an alternate water supply would not be affected. Therefore, the inherent safety margin and conservatisms in these analysis methods remain unchanged. The lack of tested fire-rated separation between fire pump circuits does not result in compromising credited automatic fire suppression functions, manual fire suppression functions, or post-fire SSD

capability. Therefore, the safety margin inherent in the analysis for the fire has been preserved.

In Approval Request 4, the licensee addressed all three echelons of DID. The licensee stated:

> The location of the fire pumps and the associated starting circuits does not affect Echelon 1. Echelons 2 and 3 are maintained by a) analysis showing that water-based fire suppression is not needed in fire scenarios that affect both fire pumps, and b) alternate provisions for establishing a fire protection water supply. The location for the starting circuits does not result in compromising the credited manual fire suppression functions or post-fire nuclear safety capability.

The NRC staff reviewed the information provided by the licensee related to Approval Request 4 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 4 is an acceptable alternative to the corresponding NFPA 805, Section 3.5.5, requirement.

3.1.4.5 Approval Request 5: Performance-Based Approach for NFPA 805, Section 3.5.14

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 5) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.5.14, requirement that all fire protection water supply and fire suppression system control valves be under a periodic inspection and be supervised. Specifically, the licensee requested approval for curb box valves in the fire protection water supply that do not meet the NFPA 805, Section 3.5.14, requirement for electrical supervision, locking, or sealing. These valves are included in a quarterly fire valve alignment verification, as allowed by NFPA 24, "Standard for Outside Protection" (Reference 87).

At DBNPS, several control valves in the supply lines to the individual fire hydrants and a portion of the south loop are underground valves that are not supervised. These valves do not have an extended permanently attached method of changing the valve's position. In Approval Request 5, the licensee stated that these valves are not designed to accept monitoring switches, locks and chains, or sealing devices. In addition, these valves are not subject to inadvertent closure or tampering because they require the use of a special T-wrench to be operated.

The licensee stated in Approval Request 5:

> **Nuclear Safety and Radiological Release Performance Criteria:**
>
> The non-supervision of curb valves for the underground yard fire main loop does not affect nuclear safety performance criteria (NSPC) since the valves are located underground and are not subject to inadvertent closure or tampering because of their inaccessibility and requirement to use a special T-wrench for operation. The valves are operated by trained personnel to ensure that water is available to plant fire protection systems as required; therefore, there is no impact on the NSPC. Periodic inspections ensure that the valves are in the correct position.

> Similarly, the non-supervision of curb box valves has no impact on the radiological release performance criteria. The radiological review was performed based on the potential location of radiological concerns and is not dependent on the fire protection water system. The use of non-supervised curb box valves does not change the radiological release evaluation performed that concluded that potentially contaminated water is contained and smoke is monitored. The fire protection water system does not add additional radiological materials to the areas or challenge systems boundaries.
>
> **Safety Margin and Defense-in-Depth:**
>
> The non-supervised curb valves for the underground fire main loop require a special T-wrench for operation, and are operated by trained, authorized personnel. The curb valves are periodically inspected to ensure they are in the correct position. The code edition of NFPA 24 referenced in NFPA 805 and subsequent editions allows an exception not to lock or electrically supervise curb box valves. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

In Approval Request 5, the licensee addressed all three echelons of DID. The licensee stated:

> Echelon 1 is met through plant fire prevention procedures and is not affected by this configuration. Echelons 2 and 3 are met since the non-supervised valves do not adversely affect the system pressure or flow, nor compromise fire suppression functions, manual fire suppression functions, or post-fire safe shutdown capability.

The NRC staff reviewed the information provided by the licensee related to Approval Request 5 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 5 is an acceptable alternative to the corresponding NFPA 805, Section 3.5.14, requirement.

3.1.4.6 Approval Request 6: Performance-Based Approach for NFPA 805, Section 3.6.3

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 6) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.6.3, requirement to supply the proper type of hose nozzle to each power block area based on the area fire hazards. Specifically, the licensee requested approval to use adjustable fog nozzles at hose stations located outside the high-voltage electrical switchgear rooms.

The licensee stated in Approval Request 6 that the current nozzles allow trained fire brigade members to select the desired spray pattern manually in lieu of limiting the nozzle's capabilities to a fixed fog, giving the trained firefighter more control over the various types of fires that could potentially occur near these hose stations. The licensee also stated:

> The adjustable fog nozzles allow the Fire Brigade to extinguish potential electrical fires in other areas. The nozzles currently in use at DBNPS allow the Fire Brigade to adapt manual suppression techniques quickly to potentially

changing fire sources and scenarios. Limiting the nozzle capability to fixed fog will not enhance plant safety and could delay the Fire Brigade response time if the nozzle is required to be interchanged to match the source and scenario.

The licensee stated in Approval Request 6:

Nuclear Safety and Radiological Release Performance Criteria:

The use of adjustable nozzles in lieu of fixed fog nozzles at hose stations does not affect nuclear safety performance criteria. The fire brigade members are experienced and trained on using the nozzles. Hose stations can be used on multiple types of fires in the Turbine, Auxiliary Buildings, other locations, as well as the high voltage switchgear rooms. It is critical to maintain a nozzle that can quickly be adjusted to adapt to any expected fire source, as well as potentially changing fire scenarios.

The use of adjustable nozzles at hose stations has no impact on the radiological release performance criteria. The radiological review was performed based on the potential location of radiological concerns and is not dependent on the fire protection water system. The use of adjustable nozzles at hose stations does not change the radiological release evaluation performed that concluded that potentially contaminated water is contained and smoke is monitored. The use of adjustable nozzles at hose stations does not add additional radiological materials to the areas or challenge systems boundaries.

Safety Margin and Defense-in-Depth:

Fire brigade members are trained to use the appropriate spray pattern for different types of fires. It is also essential for the fire brigade members to have the appropriate tools to extinguish fires using practiced techniques. The nozzles currently in use allow the fire brigade to adapt manual suppression techniques to changing fire sources and scenarios. Limiting the nozzle capability to fixed fog will not enhance plant safety. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

In Approval Request 6, the licensee addressed all three echelons of DID. The licensee stated:

Echelon 1 is met through fire brigade training procedures and is not affected by this configuration. Training includes both classroom and on-the-job practical training to ensure that the Fire Brigade members have the necessary knowledge, skills, and abilities to successfully fight fires. Echelon 2 is met since the use of adjustable nozzles at these hose stations will allow the fire brigade members to adapt the manual suppression capability to the appropriate fire scenarios. Echelon 3 is met since the use of adjustable nozzles do not adversely affect the system pressure or flow, nor compromise fire suppression functions, manual fire suppression functions, or post-fire safe shutdown capability.

The NRC staff reviewed the information provided by the licensee related to Approval Request 6 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection

DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 6 is an acceptable alternative to the corresponding NFPA 805, Section 3.6.3, requirement.

3.1.4.7 Approval Request 7: Performance-Based Approach for NFPA 805, Section 3.9.4

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 7) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.9.4, requirement that diesel-driven fire pumps be protected by automatic sprinklers. Specifically, the licensee requested approval to not have automatic sprinkler coverage over the diesel-driven fire pump.

In Approval Request 4, the licensee provided a basis for accepting the lack of automatic sprinkler coverage over the diesel-driven fire pump. The licensee stated that the diesel-driven fire pump is located in a separate room, divided by a 3-hour-rated fire wall, from the electric fire pump. In addition, the diesel-driven fire pump is separated from adjacent compartments by 3-hour-rated fire walls. The room for the diesel-driven fire pump has a relatively low combustible loading and combustibles in the area are controlled. The diesel-driven fire pump is protected by an early warning smoke detection system that alarms in the MCR, which is constantly attended. Manual fire suppression means are available, including a manually isolated sprinkler system, portable fire extinguishers, and local hose stations. Plant pre-fire plans and associated training ensures that the fire brigade and operations personnel are adequately trained regarding manual actuation and response to a fire in this area.

The licensee stated in Approval Request 7:

> **Nuclear Safety and Radiological Release Performance Criteria:**
>
> The lack of automatic sprinklers to protect the diesel-driven fire pump does not affect nuclear safety. Both the diesel-driven and electric fire pump individually have the ability to supply the required fire water and the diesel engine-driven fire pump is not relied upon for other water requirements. Therefore there is no impact on the nuclear safety performance criteria.
>
> The lack of automatic sprinklers to protect the diesel engine-driven fire pump has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the location of the fire pumps. The radiological release evaluation concluded that potentially contaminated water is contained and smoke is monitored. The configuration of the diesel-driven fire pump room does not add additional radiological materials to the area or challenge systems boundaries.
>
> **Safety Margin and Defense-in-Depth:**
>
> The lack of automatic sprinklers to protect the diesel-driven fire pump does not affect the ability to supply the required fire water in a fire event. Only one fire pump is required for fires in safety-related areas. The use of the diesel-driven fire pump has been defined by the limitations of the analysis of the fire. Therefore, the safety margin inherent in the analysis for the fire has been preserved.

In Approval Request 7, the licensee addressed all three echelons of DID. For Echelon 1, the licensee stated that the lack of automatic sprinklers to protect the diesel engine-driven fire pump does not affect administrative controls for preventing fires from starting. For Echelons 2 and 3, the licensee stated that this condition does not impact the ability of the automatic fire suppression systems to perform their functions because the electric fire pump is capable of supplying the necessary fire water to safety-related areas. Additionally, the licensee stated that this condition does not compromise automatic fire suppression functions, manual fire suppression functions, or post-fire SSD capability.

The NRC staff reviewed the information provided by the licensee related to Approval Request 7 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 7 is an acceptable alternative to the corresponding NFPA 805, Section 3.9.4, requirement.

3.1.4.8 Approval Request 8

In its December 16, 2016, response to SSA RAI 12 (Reference 10), the licensee withdrew LAR Attachment L, Approval Request 8.

3.1.4.9 Approval Request 9: Performance-Based Approach for NFPA 805, Section 3.6.1

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 9) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.6.1, requirement that, for all power block buildings, Class III standpipe and hose systems shall be installed in accordance with NFPA 14 (Reference 51). Specifically, the licensee requested approval for the lack of a fire hose standpipe system for the containment building.

In Approval Request 9, the licensee stated that the NRC staff previously reviewed and approved the DBNPS fire hazards analysis (Reference 21), with the absence of a fire hose standby system for the containment building. The licensee stated that the basis for the previous approval has changed due to modification of the ionization detectors to photoelectric smoke and heat detectors, which are functionally equivalent for this application. The licensee stated that the remaining basis from the previous approval remains the same.

The licensee stated that the RCP lube oil is the major combustible in containment. As discussed in SE Section 3.1.4.3, the OCS is designed to prevent the RCP lubricating oil system from becoming a potential ignition source. The licensee stated that there is a low quantity of combustibles in containment. In addition, the early warning fire detection system is installed at various areas within the containment. The licensee also stated:

> The fire suppression strategy inside the containment still relies on manual fire fighting operation. Portable fire extinguishers are provided inside of Containment during non-operating modes as stated in the previous approval. However, portable fire extinguishers are removed to the outside of Containment during reactor power operating modes due to potential containment sump clogging issues and exposure to high ambient temperatures. There are fire extinguishers readily available for fire brigade use outside containment near the containment

access door in the event of a fire. Hose stations (HCR-27 and HCS-28) and additional lengths of fire hose and low pressure nozzles are available just outside of containment near the personnel hatch.

The licensee stated in Approval Request 9:

Nuclear Safety and Radiological Release Performance Criteria:

The lack of fire hose standpipe system for the containment building does not affect nuclear safety due to limited combustible material in the containment building and the existence of the reactor coolant pump oil collection system. In addition, the fire detection inside of containment provides early warning of fire development and allows for fire brigade notification and response to suppress a containment fire as trained. Therefore, there is no impact on the nuclear safety performance criteria.

The radiological release review was performed based on the manual fire suppression in areas containing or potentially containing radioactive materials. A fire that originates in this fire area will be contained in this fire area. This fire area is separated from Fire Areas A and AB (the East and West halves of the Annulus) by the steel pressure vessel. Based on the low combustible loading and the design of the pressure vessel, a fire will not spread from one side of the pressure vessel to the other. The radiological release evaluation concluded that due to engineered controls to contain both smoke and fire suppression water runoff and use of revised pre-fire plans, there is no impact on the radiological release performance criteria.

Safety Margin and Defense-in-Depth:

The fire brigade at Davis-Besse is trained to manually extinguish fires and have access to containment pre-fire plans. Manual fire suppression equipment is also maintained specifically for containment near the containment access point; therefore, the safety margin inherent in the analysis for the fire event has been preserved.

In Approval Request 9, the licensee addressed all three echelons of DID. For Echelon 1, the licensee stated that the lack of a fire hose standpipe system in the containment building does not affect administrative controls for preventing fires from starting. In addition, the RCP OCS also prevents fires from starting. For Echelon 2, the licensee stated that extinguishers are maintained readily available, and photoelectric-type smoke and heat detectors are in containment areas. For Echelon 3, the licensee stated:

The lack of a fire hose standpipe system in the containment building does not allow or increase its potential for fire propagation through the fire compartment's steel pressure vessel barrier. Due to low combustible loading and the design of the pressure vessel, a fire will not spread from the fire compartment. Impacts to equipment due to manual fire fighting activities are not expected to result in additional fire damage states. The plant fire brigade is trained to discharge water in a judicious manner and instructed to direct hose streams and portable extinguishers at the base of the fire to limit the amount of overspray beyond the immediate fire source. A fire does not result in compromising essential safe

shutdown functions from being performed or post-fire SSD capability; therefore, echelon 3 is met.

The NRC staff reviewed the information provided by the licensee related to Approval Request 9 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 9 is an acceptable alternative to the corresponding NFPA 805, Section 3.6.1, requirement.

3.1.4.10 Approval Request 10: Performance-Based Approach for NFPA 805, Section 3.3.8

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 10) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.3.8, requirement that bulk storage of flammable and combustible liquids shall, at a minimum, comply with NFPA 30 (Reference 52). Specifically, the licensee requested approval of several deviations from NFPA 30 for the station blackout (SBO) diesel fuel oil tank (T210) inside the SBO building.

In Approval Request 10, the licensee stated that SBO diesel fuel tank and the associated piping complies with the 1990 edition of NFPA 30, with certain exceptions. The licensee further stated that in the instances where the installed configuration of tank T210 is not functionally equivalent to the requirements of NFPA 30, it concluded that the configuration is adequate for the hazard. The licensee identified the following exceptions from the requirements of the 1990 edition of NFPA 30 for the SBO diesel fuel tank, and provided justification that the conditions are adequate for the hazard:

- Drainage is not provided to the curbed area under tank T210 for the removal of any sprinkler system actuation discharge (NFPA 30, Sections 1-1.6 and 5-3.4.1).

  The licensee stated the existing curb arrangement captures and delays the water and oil runoff, and the fire brigade is trained to contain any runoff. Following the event, the water and oil retained in the tank T210 pit will be pumped out and processed using normally available plant equipment and procedures.

- Emergency vent capability for tank T210 is not met (NFPA 30, Sections 1-1.6 and 2-5.2).

  The licensee stated that the prompt actions of the fire brigade to provide additional fire suppression will minimize tank T210 heat-up. This is expected to be sufficient to prevent any internal pressure increase from exceeding the capability of the existing vent to atmosphere.

- Self-actuated closing valves are not provided for the piping connections that are below the level of tank T210 (NFPA 30, Section 2-5.4.3).

  The licensee stated that the full area automatic sprinkler system will provide adequate fire suppression coverage and cooling, the pipe and fittings are seismically designed, the piping is welded, and the fire brigade will provide

additional manual fire suppression. Any maintenance issues, such as pipe or valve packing leaks, would be identified for repair.

- Tank T210 does not have the required fire protection coating on the tank supports (NFPA 30, Sections 2-6.2 and 2-6.3).

  The licensee stated that the full area automatic sprinkler coverage will aid in reducing the heat exposure to the tank supports, and the fire brigade is expected to provide additional fire suppression. These actions make it unlikely the tank supports would be exposed to sufficient heat to cause a collapse.

The licensee stated in Approval Request 10:

> **Nuclear Safety and Radiological Release Performance Criteria:**
>
> The lack of NFPA 30 compliance for the T210 diesel fuel tank does not affect nuclear safety due to the automatic suppression system and the rapid fire brigade response. A fire in and/or loss of the SBO building, based on the NFPA 805 safe shutdown analysis does not affect Davis-Besse's ability to shutdown; therefore, there is no impact on the nuclear safety performance criteria.
>
> The lack of a drain in the curbed area under Tank T210 does not impact radiological release because there is no radiological material located within the SBO building and the fire brigade will direct any runoff to either the outside or to the SBO sump; therefore there is no impact on the radiological release performance criteria.
>
> **Safety Margin and Defense-in-Depth:**
>
> The deviations from the requirements of NFPA 30 were reviewed and the safety margin is maintained based on the low risk of a fire occurring, the mitigating automatic fire protection systems and features, and the response of the fire brigade. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

In Approval Request 10, the licensee addressed all three echelons of DID. The licensee stated:

> Echelon 1 is met through plant fire prevention procedures and is not affected by this configuration. Echelon 2 is met by maintaining automatic and manual fire suppression functions. In addition, the fire brigade is trained to rapidly respond to and extinguish fires. Echelon 3 is met since the deviations still provide an adequate level of fire protection suppression and cooling to ensure that a fire will not prevent essential safety functions from being performed. The deviations are adequate for the hazard and will not increase the likelihood of damage since the fire protective coating is intact on the structural steel except in a localized location where a joint between the top layer of protective material is not sealed. and the full area automatic sprinkler system and prompt fire brigade actions will minimize the tank heat-up. Additionally, the SBO building is in a remote area of the site such that the reliability of onsite emergency [alternating-current] power sources (e.g., emergency diesel generators), the frequency of loss of offsite

> power, and the probable time to restore offsite power ensures that a temporary loss of the SBO Diesel Building will not result in compromising post-fire SSD capability.

The NRC staff reviewed the information provided by the licensee related to Approval Request 10 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 10 is an acceptable alternative to the corresponding NFPA 805, Section 3.3.8, requirement.

3.1.4.11 Approval Request 11: Performance-Based Approach for NFPA 805 Section 3.2.3(1)

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 11) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.2.3(1), requirement that procedures to accomplish inspection, testing, and maintenance for fire protection systems and features credited by the FPP be established. Specifically, the licensee requested approval to use EPRI TR 1006756 (Reference 79) to establish the appropriate inspection, testing, and maintenance frequencies for fire protection systems and features credited by the new FPP. The licensee stated that this request does not involve the use of the EPRI TR 1006756 to establish the scope of the activities determined by the required systems review.

In Approval Request 11, the licensee stated:

> The target tests, inspections, and maintenance will be for the fire protection systems and features required by NFPA 805. The failure criterion in each case will be based on the credited functions of the required fire protection systems and features and will ensure those functions are maintained (or appropriate actions are implemented). Data collection and analysis will follow the EPRI TR-1006756 document guidance. The failure probability will be determined based on the TR-1006756 guidance, and a 95% confidence level will be used. The 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.

The licensee stated in Approval Request 11 that: "Use of performance-based test frequencies established per EPRI TR 1006756 methods combined with NFPA 805 Section 2.6, 'Monitoring,' 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." Based on this, the licensee concluded that the use of the PB methods in EPRI TR 1006756 would have no adverse impact on the nuclear safety and radioactive release performance criteria.

The licensee stated that use of the methods in EPRI TR 1006756 will maintain the inherent safety margins contained in the codes used for design and maintenance of fire protection systems and features. The licensee further stated:

> Safety margins are maintained if codes and standards are met and safety analysis acceptance criteria in the licensing basis (e.g., UFSAR, supporting analyses) are met or provide sufficient margin to account for analysis and data

> uncertainty. An adequate safety margin for each of the surveillance and maintenance frequencies evaluated in accordance with EPRI TR-1006756 methods is preserved by establishing reliability and availability standards for the affected equipment. The performance parameters for the evaluated fire protection systems and features are created to support nuclear performance criteria contained in the plant-specific accident analyses. These analyses established component and system performance criteria necessary to establish safe and stable plant operation as well as safe shutdown of the unit in the event of a fire. These performance parameters were not modified as a result of this approval request. Therefore, the safety margin inherent to, and credited in, the analysis has been preserved.

In Approval Request 11, the licensee addressed all three echelons of DID. The licensees stated:

> Echelon 1 is not affected by the use of EPRI TR-1006756 methods. Echelons 2 and 3 are not affected by the use of performance-based test frequency guidance established per EPRI TR-1006756 report when combined with NFPA 805 Section 2.6, "Monitoring" requirements. This will ensure for Echelons 2 and 3 that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analysis (or the analysis will be updated accordingly).

The NRC staff reviewed the information provided by the licensee related to Approval Request 11 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 11 is an acceptable alternative to the corresponding NFPA 805, Section 3.2.3(1), requirement.

3.1.4.12 Approval Request 12: Performance-Based Approach for NFPA 805, Section 3.3.8

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 12) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.3.8, requirement that bulk storage of flammable and combustible liquids shall, at a minimum, comply with NFPA 30 (Reference 52). Specifically, the licensee requested approval of a deviation from NFPA 30 for the lube oil storage tank (T32).

In Approval Request 12, the licensee stated that the lube oil storage tank complies with the 1973 edition of NFPA 30, with one exception. Specifically, the lube oil storage tank is not provided with an automatic closing heat actuated valve on each withdrawal connection below the liquid level (NFPA 30, Section 2343). The licensee stated that this deviation is acceptable because: (1) the full area automatic sprinkler system will provide adequate fire suppression coverage and cooling; (2) the piping and fittings are seismically designed; (3) the tank is enclosed in a room with 3-hour fire-rated barriers; and (4) manual fire suppression is available from the fire brigade. In addition, leaks identified during regular operator rounds will be identified for repair. Therefore, the licensee concluded this configuration is adequate for the hazard.

The licensee stated in Approval Request 12:

> The NFPA 30 deviations for the Lube Oil Storage Tank does not affect nuclear safety due to the automatic suppression system and the rapid fire brigade response. A fire in and/or loss of the Lube Oil Storage Tank would not affect any safe shutdown trains based on the NFPA 805 safe shutdown analysis, and does not affect Davis-Besse's ability to safely shutdown; therefore, there is no impact on the nuclear safety performance criteria.

The licensee also stated that the radiological release performance criteria are not impacted because there is no radiological material located within the lube oil storage tank room and the fire brigade monitors effluents for contamination. The licensee stated that the safety margin inherent in the analysis for the fire event is maintained because of the low risk of a fire occurring, the mitigating automatic fire protection systems and features, and the response of the fire brigade.

In Approval Request 12, the licensee addressed all three echelons of DID. The licensee stated:

> Echelon 1 is met through plant fire prevention procedures and is not affected by this configuration. Echelon 2 is met by maintaining automatic sprinkler system and manual suppression. In addition, the fire brigade is trained to rapidly respond to and extinguish fires. Echelon 3 is met since there are 3-hour fire-rate[d] barriers, an adequate level of automatic and manual fire suppression is provided, and a fire will not prevent essential safety functions from being performed.

The NRC staff reviewed the information provided by the licensee related to Approval Request 12 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 12 is an acceptable alternative to the corresponding NFPA 805, Section 3.3.8, requirement.

3.1.4.13 Approval Request 13: Performance-Based Approach for NFPA 805, Section 3.3.8

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 13) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.3.8, requirement that bulk storage of flammable and combustible liquids shall, at a minimum, comply with NFPA 30 (Reference 52). Specifically, the licensee requested approval of several deviations from NFPA 30 for the EDG fuel oil day tanks (T46-1 and T46-2) in the auxiliary building.

In Approval Request 13, the licensee stated that the EDG fuel oil day tanks comply with the 1973 edition of NFPA 30, with certain exceptions. For instances where the installed configuration of the EDG fuel oil day tanks is not functionally equivalent to the requirements of NFPA 30, the licensee concluded that the configuration is adequate for the hazard. The licensee identified the following exceptions from the requirements of the 1973 edition of NFPA 30 for the EDG fuel oil day tanks and provided justification that the conditions are adequate for the hazard:

- The two connections below the liquid level of the tanks are not provided with an automatic closing heat actuated valve (NFPA 30, Section 2343).

  The licensee stated that: (1) there are no fixed ignition sources; (2) the automatic sprinkler system will provide adequate fire suppression coverage for cooling the hot gas layer (HGL) and effective cooling to the exposed oil and to the exposed top portion of the tanks; (3) the enclosed tank rooms are located in separate 3-hour-rated fire compartments; (4) the piping below the liquid level is seismically constructed; and (5) manual suppression is available from the fire brigade. In addition, leaks identified during regular operator rounds will be identified for repair. Therefore, the licensee concluded this configuration is adequate for the hazard.

- A flame arrestor on the emergency vent for the tanks restricts the flow rate such that it does not comply with NFPA 30, Section 2154.

  The licensee stated that the enclosed tank rooms are located in separate 3-hour-rated fire compartments with an automatic sprinkler system. In addition, manual fire suppression is available from the fire brigade. The piping and fittings are seismically designed. In the unlikely event of a tank rupture, the licensee stated that the contents would be contained within the room. If the released oil ignites, the sprinkler system would activate and rapidly raise the liquid (oil and water) level in the room to the bottom of the tank, such that the tank's presence would cause any further increase in level to decrease the exposed surface area of any potential ignited pool fire. Therefore, the licensee concluded that the sprinkler system will provide cooling to the HGL, and effective cooling to the exposed oil and to the exposed top portion of the tank. Since both tanks are electrically grounded, it is not expected that any static charge would develop to initiate a fire inside the tank from the vapor. Leaks identified during regular operator rounds would be identified for repair. Therefore, the licensee concluded that this configuration is adequate for the hazard.

The licensee stated in Approval Request 13:

> The NFPA 30 deviations for the EDG Fuel Oil Day Tanks do not affect nuclear safety due to fire rated separation of each of the EDG Fuel Oil Day Tank rooms, the automatic suppression system and the rapid fire brigade response. A fire in and/or loss of an EDG Fuel Oil Day Tank would not affect the safe shutdown capabilities of the redundant train since the tank contents would be contained within the fire compartment containing the tank, and prevent fire migration to other areas. The loss of an EDG Fuel Oil Day Tank would only affect one safe shutdown train and based on the NFPA 805 safe shutdown analysis, does not affect Davis-Besse's ability to safely shutdown; therefore, there is no impact on the nuclear safety performance criteria.

The licensee also stated that the radiological release performance criteria are not impacted because there is no radiological material located within the EDG area of the auxiliary building and the fire brigade will direct water runoff appropriately. The licensee stated that the safety margin inherent in the analysis for the fire event is maintained because of the low risk of a fire occurring, the mitigating automatic fire protection systems and features, and the response of the fire brigade.

In Approval Request 13, the licensee addressed all three echelons of DID. The licensee stated:

> Echelon 1 is met through plant fire prevention procedures and is not affected by this configuration. Echelon 2 is met by maintaining automatic sprinkler system and manual suppression. In addition, the fire brigade is trained to rapidly respond to and extinguish fires. Echelon 3 is met since each EDG Fuel Oil Day Tank fire compartment is enclosed in 3-hour fire-rate[d] barriers, an adequate level of automatic and manual fire suppression is provided, and in the unlikely event of a condition resulting in inadequate emergency vent capacity that results in the EDG Fuel Oil Day Tank becoming over-pressurized and ruptured, the contents would be contained within the fire compartment containing the affected EDG Day Tank. A fire will not prevent essential safety functions from being performed since only one safe shutdown train would be affected.

The NRC staff reviewed the information provided by the licensee related to Approval Request 13 and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 13 is an acceptable alternative to the corresponding NFPA 805, Section 3.3.8, requirement.

3.1.4.14 Approval Request 14: Performance-Based Approach for NFPA 805, Section 3.3.8

In LAR Attachment L (Reference 13), the licensee requested approval (Approval Request 14) of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.3.8, requirement that bulk storage of flammable and combustible liquids shall, at a minimum, comply with NFPA 30 (Reference 52). Specifically, the licensee requested approval of deviations from NFPA 30 associated with the diesel fuel oil tank (T145) in the EFWF.

The licensee stated that its EFWF complies with the 2015 edition of NFPA 30, with certain exceptions. For instances where the EFWF is not functionally equivalent to the requirements of NFPA 30, the licensee concluded that the condition is adequate for the hazard. The licensee identified the following exceptions from the requirements of 2015 edition of NFPA 30 associated with the diesel fuel oil tank for the EFWF, and provided justification that the conditions are adequate for the hazard:

- Prompt notification in the event of a fire is not provided for the EFWF (NFPA 30, Section 6.6.1).

  The licensee stated that the EFWF room which houses the diesel fuel oil tank and the EFWF external barriers are of robust and fire resistant construction that is expected to contain the effects of a fire. The tank is manufactured with double wall construction which protects it from physical damage and limits heat transferred to the internal primary tank when exposed to an external fire. The top-side oil transfer connections make a diesel fuel oil spill highly unlikely. Leaks identified during regular operator rounds will be identified for repair. In addition, fire damage to the EFWF does not affect either train of the SSD capabilities within the remaining areas of the plant. The licensee stated that the fire brigade can provide manual fire suppression. Therefore, the licensee concluded that the

absence of a prompt notification for a fire in the EFWF is adequate for the hazard.

- The construction and fire protection of the steel supports for the EFWF diesel fuel oil tank does not comply with NFPA 30, Sections 22.5.2.3 and 22.5.2.4.

  The licensee stated that the tank supports are seismically designed. The postulated pool fire would be confined due to the oil containment provided by the room and the robust and fire resistive design of the EFWF building. Transient combustibles are controlled and the tank room is in a location without significant ignition sources. Leaks identified during regular operator rounds will be identified for repair. In addition, damage to the tank from a fire does not affect either SSD trains or SSD capabilities within the remaining areas of the plant. The licensee stated that the fire brigade can provide manual fire suppression. Therefore, the licensee concluded that the supports for the diesel fuel oil tank are adequate for the hazard.

- The emergency vent piping for the EFWF diesel fuel oil tank is directly routed outside of the tank room and out of the building through the roof. However, the vent piping capability is not sized to meet the NFPA 30, Section 22.7.3.2, emergency venting flow requirements. A flame arrestor is installed at the vent discharge to atmosphere that further restricts any effluent discharge.

  The licensee stated that the walls, floor, and ceiling of the tank room are of robust 3-hour fire-rated construction. The tank is separated from other ignition sources outside the tank room. The tank is electrically grounded, which reduces the likelihood of vapor ignition inside the tank. Leaks identified during regular operator rounds are identified for repair. In addition, damage to the tank from a fire does not affect either SSD trains or SSD capabilities within the remaining areas of the plant. The licensee stated that the fire brigade can provide manual fire suppression. Therefore, the licensee concluded that the existing flame arrestor and vent for the tank are adequate for the hazard.

- Class II liquid transfer connections on the EFWF diesel fuel oil tank are not provided with remotely activated or heat-activated valves or other approved device (NFPA 30, Section 24.14.3).

  The licensee stated that the tank is spatially separated from other hazards in the EFWF and means of controlling the flow of Class II liquid are provided. The piping and fittings are seismically designed, and piping and tubing are well maintained. Leaks identified during regular operator rounds are identified for repair. In addition, damage to the tank from a fire does not affect either SSD trains or SSD capabilities within the remaining areas of the plant. The licensee stated that the fire brigade can provide manual fire suppression. Therefore, the licensee concluded that the liquid transfer connections below the liquid level for the tank that do not have remotely activated or heat-activated valves or other approved devices installed are adequate for the hazard.

The licensee stated in Approval Request 14:

> The NFPA 30 deviation for the EFWF Diesel Fuel Oil Storage Tank T145 does not affect nuclear safety since the EFWF is non-safety related and it has spatial separation from the other operating areas of the plant. In the event of a fire in the Auxiliary Building, the availability of a diversely powered, automatically initiated auxiliary feedwater supply located in the EFWF would significantly reduce the likelihood of core damage; however, the loss of the EFWF equipment from a fire does not affect the normal or emergency shutdown capabilities used in plant systems, structures, and components.

The licensee also stated that the radiological release performance criteria are not impacted because there is no radiological material located within the EFWF building and the fire brigade monitors effluents for contamination. The licensee stated that the safety margin inherent in the analysis for the fire event is maintained because of the low risk of a fire occurring, the separation from the other operating areas of the plant, and the response of the fire brigade.

In Approval Request 14, the licensee addressed all three echelons of DID. The licensee stated:

> Echelon 1 is met through plant fire prevention procedures and since the EFWF building design is hardened to resist both tornado missiles and design basis earthquakes, the construction is non-combustible and fire resistant. Echelon 2 is met by manual suppression. In addition, the fire brigade is trained to rapidly respond to and extinguish fires. Echelon 3 is met since there are 3-hour fire-rated barriers and an adequate level of manual fire suppression is provided. Damage to Tank T145 from a fire does not affect either train of the safe shutdown capabilities within the remaining areas of the plant.

The NRC staff reviewed the information provided by the licensee related to Approval Request 14, and determined that the proposed PB approach: (1) satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (2) maintains safety margins; and (3) maintains fire protection DID. Therefore, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concludes that the proposed PB method in Approval Request 14 is an acceptable alternative to the corresponding NFPA 805, Section 3.3.8, requirement.

3.2 Nuclear Safety Capability Assessment Methods

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 performance criteria of NFPA 805, Section 1.5 (see SE Section 2.0). NFPA 805, Section 2.4, "Engineering Analyses," states:

> 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.

The NSCA is performed by the licensee to determine what equipment, and associated electrical cables and controls, is needed and available to safely shut down the plant in the event of a fire. The requirements for the NSCA are described in NFPA 805, Section 2.4.2, "Nuclear Safety Capability Assessment," which states:

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

In LAR Section 4.2.1, "Nuclear Safety Capability Assessment Methodology" (Reference 6), the licensee stated that its NSCA methodology review consisted of four processes:

1. Establishing compliance with NFPA 805, Section 2.4.2.
2. Establishing the safe and stable conditions for the plant.
3. Establishing recovery actions (RAs).
4. Establishing multiple spurious operations (MSOs).

SE Section 3.2.1 provides the NRC staff's review of the LAR as it relates to the first three steps of NFPA 805, Section 2.4.2, and SE Section 3.5 provides the NRC staff's review of the fourth step. SE Section 3.2.2 provides the NRC staff's review regarding the establishment of safe and stable conditions for the plant. SE Section 3.2.3 discusses the applicability of feed-and-bleed. SE Section 3.2.4 provides the NRC staff's review of the licensee's assessment of MSOs. SE Section 3.2.5 provides the NRC staff's review regarding the establishment of RAs.

In addition to NFPA 805, Sections 2.4 and 2.4.2, the NRC staff considered the nuclear safety goals, objectives, and performance criteria of NFPA 805 (see SE Section 2.0) in its review of NSCA methods.

3.2.1 Compliance with NFPA 805 Nuclear Safety Capability Assessment Methods

As noted above, this SE section addresses the first three steps of NFPA 805, Section 2.4.2. RG 1.205, Revision 1 (Reference 5), through the endorsement of NEI 04-02, Revision 2 (Reference 4), and Chapter 3 of NEI 00-01, Revision 2 (Reference 23), provides a method acceptable to the NRC for conducting an NSCA. The NRC staff considered this guidance in its review of LAR Section 4.2.1 and LAR Attachment B (Reference 6), as updated by the licensee's supplements.

As noted in NEI 04-02, the nuclear safety goals, objectives, and performance criteria of NFPA 805 (see SE Section 2.0) are similar to the requirements in Sections III.G, and III.L of 10 CFR Part 50, Appendix R. Thus, a substantial part of the existing FPP can be transitioned to

a RI/PB FPP by performing a transition review and addressing NFPA 805 topics not previously approved.

The guidance in NEI 00-01, Revision 2, provides a framework to evaluate the impact of fires on the ability to maintain post-fire SSD. It provides detailed guidance for: (1) selecting systems and components required to meet the NSPC; (2) selecting the cables necessary to achieve the NSPC; (3) identifying the location of nuclear safety equipment and cables; (4) identifying dependencies between equipment, systems, and components; and, (5) conservative assumptions to be used in the performance of the NSCA. NEI 04-02 provides the methodology worksheet (Table B-2) which includes implementing guidance for each subsection of NFPA 805, Section 2.4.2. Specifically, Table B-2 of NEI 04-02 recommends that the current analysis be compared to the methodology in NEI 00-01 to demonstrate compliance with the subsections of NFPA 805, Section 2.4.2, as follows:

- For Subsection 2.4.2.1, "Nuclear Safety Capability Systems and Equipment Section," compare the methodology of the current SSD equipment list to NEI 00-01.

- For Subsection 2.4.2.2.1, "Circuits Required in Nuclear Safety Functions," compare the methodology of the current circuit analysis to NEI 00-01.

- For Subsection 2.4.2.2.2, "Other Required Circuits," compare the methodology of the current associated circuits analysis to NEI 00-01.

- For Subsection 2.4.2.3, "Nuclear Safety Equipment and Cable Location," and Subsection 2.4.2.4, "Fire Area Assessment," compare the methodology of the current equipment and cable location analysis to NEI 00-01.

In addition, FAQ 07-0039 (Reference 62) provides one acceptable method for documenting the comparison of the SSA against the NFPA 805 requirements. This method first maps the existing SSA to the NEI 00-01, Chapter 3, methodology, which, in turn, is mapped to the NFPA 805, Section 2.4.2, requirements.

In LAR Section 4.2.1.1, the licensee stated it compared its existing SSA methodology against the guidance in Chapter 3 of NEI 00-01, Revision 2, using the guidance NEI 04-02, Appendix B-2. The licensee stated that each subsection of NFPA 805, Section 2.4.2, was correlated to the corresponding section of NEI 00-01, Chapter 3. For each section of NEI 00-01, Chapter 3, applicable to DBNPS, the licensee determined whether the existing SSA aligns with the section; aligns with the intent of the section; or is not in alignment, but there are no adverse consequences. The licensee documented the results of this review in LAR Attachment B, as updated by RAI responses. The licensee did not identify any sections of NEI 00-01 where there was prior NRC approval of an alternative for an SSA that was not in alignment.

3.2.1.1 Aligns with NEI 00-01 Sections

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 an FPP under 10 CFR 50.48(c). For the majority of the NEI 00-01, Chapter 3, sections, the licensee determined that the SSA aligns directly with the section. For these cases, the NRC staff verified that the licensee's analyses, as described in LAR Attachment B

(Reference 6), are consistent with the regulatory guidance for: (1) selecting systems and equipment and their interrelationships necessary to achieve the NSPC, (2) selection of the cables necessary to achieve the NSPC, and (3) the identification of the location of nuclear safety equipment and cables.

3.2.1.2 Aligns with Intent of NEI 00-01 Sections

For certain sections of NEI 00-01, the licensee determined that the SSA aligns with the intent of these sections. The licensee described how it aligns with the intent of these sections in LAR Attachment B (Reference 6), as updated by its RAI responses. The NRC staff's review of the licensee's evaluation for each of these sections is discussed below.

NEI 00-01, Section 3.2.1.2

NEI 00-01, Section 3.2.1.2, states, in part, that when identifying equipment necessary to perform the required SSD functions:

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

The licensee stated in LAR Attachment B that it assumes that passive mechanical components, including heat exchangers, manual valves, relief valves, and check valves, remain available after a fire. In LAR Attachment G (Reference 15), the licensee stated that each of the feasibility criteria in FAQ 07-0030 (Reference 59) were assessed for the RAs listed in LAR Table G-1, "Davis-Besse Recovery Actions and Activities Occurring at the Primary Control Station(s)," and demonstrated that the RAs are creditable and feasible. The NRC staff determined that the licensee aligns with the intent of NEI 00-01, Section 3.2.1.2, because the licensee demonstrated the feasibility of RAs credited to meet NFPA 805, Chapter 4.

NEI 00-01, Section 3.2.2.2

NEI 00-01, Section 3.2.2.2, states that, as part of the methodology for selecting SSD equipment for a post-fire SSA, equipment in each system's flow path should be identified. This should include identifying equipment that could spuriously operate and adversely affect the desired system functions.

The licensee stated in LAR Attachment B that the method used to develop the list of SSD components included identifying: (1) all active and passive components required to function on the normal flow paths; (2) alternate and backup equipment that could perform the same function; and (3) any equipment that could spuriously actuate to a position detrimental to proper system operation, or whose maloperation could result in a breach of the reactor coolant boundary. The NRC staff determined that this method aligns with the intent of NEI 00-01, Section 3.2.2.2, because the licensee evaluated the effects of spurious operation and its impact on meeting the nuclear safety functions.

NEI 00-01, Section 3.3.1.1.4.1

NEI 00-01, Section 3.3.1.1.4.1, is associated with evaluating the effects of fire-induced failure of automatic-initiation logic circuits and the potential to adversely affect any post-fire SSD system function.

The licensee stated in LAR Attachment B that automatic initiation of systems required to achieve and maintain SSD is not credited, and that fire-induced automatic-initiation signals are evaluated for the possibility of spurious component operation and their subsequent adverse impact on SSD. The NRC staff determined that the licensee aligns with the intent of NEI 00-01, Section 3.3.1.1.4.1, because the licensee evaluated the effects of fire-induced automatic signals and their impact on meeting the NSPC.

NEI 00-01, Section 3.3.3.3

NEI 00-01, Section 3.3.3.3, states that, as part of the methodology for cable selection and location, cables required to operate or that may result in maloperation of each piece of SSD equipment should be identified. In addition, the list of cables potentially affecting each piece of equipment should be tabulated in a relational database that includes the respective drawing numbers, revision identification, and any interlocks that are investigated to determine their impact on the operation of the equipment.

The licensee stated in LAR Attachment B that the tracing of all SSD circuits "culminated in the development of a database that identifies all raceway routings (by fire area) for all circuits associated with safe shutdown components." However, the licensee's database does not reference the drawing revision numbers. The NRC staff determined that the licensee aligns with the intent of NEI 00-01, Section 3.3.3.3, because the licensee included the cables associated with equipment in the database.

NEI 00-01, Section 3.5.1.1

NEI 00-01, Section 3.5.1.1, provides circuit failure criteria for performing fire-induced circuit failure evaluations, which address the effects of multiple fire-induced circuit failures due to hot shorts, open circuits, and shorts to ground on unprotected circuits. This section states that "for ungrounded [direct-current (dc)] circuits, a single hot short from the same source is assumed to occur unless it can be demonstrated that the occurrence of a same source short is not possible in the affected fire area."

The licensee stated in LAR Attachment B that its SSA identified all cables in the fire area that may adversely affect SSD equipment as a result of open circuits, shorts, or hot shorts. Where necessary to demonstrate availability of credited SSD functions, the licensee performed a circuit analysis as part of the fire area compliance assessment. However, for multiple high impedance faults, the licensee stated that existing circuit analyses have relied upon ungrounded dc circuits of proper polarity not faulting in some cases. In response to SSA RAI 03 (Reference 10), the licensee clarified that this statement was referring to certain cases of fire damage documented in the fire hazards analysis report for compliance with 10 CFR Part 50, Appendix R, which will not be transitioned to the NFPA 805 licensing basis. The RAI response also stated: "When the component is required to maintain long term safe and stable conditions at hot standby, but had cable damage, a VFDR was assigned and a fire risk evaluation was performed."

In response to SSA RAI 03.01 (Reference 12), the licensee stated:

> The [fire hazards analysis report] methodology for resolving circuit failures for ungrounded direct-current circuits based on the number of required shorts was not credited for NFPA 805 transition. To support the new licensing basis for these circuits under NFPA 805, the nuclear safety capability assessment (NSCA) assigned a VFDR to the analysis failures of these ungrounded direct-current circuits. The VFDR was then dispositioned during the performance of the fire risk evaluation (FRE) process, by using risk insights to evaluate the circuit failure consequence.

The NRC staff determined that the licensee aligns with the intent of NEI 00-01, Section 3.5.1.1, because the licensee evaluated the effects of fire damage to unprotected circuits (i.e., hot shorts, open circuits, and shorts to ground).

NEI 00-01, Sections 3.5.2.1 and 3.5.2.3

NEI 00-01, Sections 3.5.2.1 and 3.5.2.3, provide guidance for addressing the effects of circuit failures due to an open circuit and a hot short, respectively. The licensee stated in LAR Attachment B that its SSA identified all cables in the fire area that may adversely affect SSD equipment as a result of open circuits, shorts, or hot shorts. Where necessary to demonstrate availability of credited SSD functions, the licensee performed circuit analysis as part of the fire area compliance assessment, which considered the potential impact of open circuits, hot shorts, and shorts to ground on unprotected cables. The NRC staff determined that the licensee aligns with the intent of the NRC-endorsed guidance, which is to ensure that the post-fire SSA addresses the guidance in NEI 00-01, Sections 3.5.2.1 and 3.5.2.3, because the licensee evaluated the effects of fire-induced circuit failures due to open circuits and hot shorts and their impact on meeting the NSPC.

NEI 00-01, Section 3.5.2.4

NEI 00-01, Section 3.5.2.4, states: "The evaluation of circuits of a common power source consists of verifying proper coordination between the supply breaker/fuse and the load breakers/fuses for power sources that are required for hot shutdown."

The licensee stated in LAR Attachment B that it credits a power source only when adequate coordination can be shown, and that it analyzed power supplies for additional equipment credited in the FPRA for risk reduction. This analysis included consideration of common enclosure effects on circuit failures, consistent with the methodology described in NEI 00-01. The NRC staff determined that the licensee aligns with the intent of NEI 00-01, Section 3.5.2.4, because the licensee evaluated the power supplies required to meet the NSPC.

NEI 00-01, Section 3.4.1.4

NEI 00-01, Section 3.4.1.4, provides guidance for classifying cables and components as either required or important to SSD. The licensee stated in LAR Attachment B, Section 3.4.1.4, that it did not use this classification since component identification and cable selection was performed prior to the issuance of NEI 00-01. The licensee stated that this does not impact the selection of systems, equipment, cables, and the identification of their location for use in the transitioning nuclear safety analysis. The NRC staffs determined that the licensee aligns with the intent of NEI 00-01, Section 3.4.1.4, because the DBNPS fire hazards analysis report provides

justification when the protection of cables and components in a fire area of systems whose function is required for hot shutdown does not meet 10 CFR Part 50, Appendix R.

NEI 00-01, Section 3.4.1.7

NEI 00-01, Section 3.4.1.7, provides guidance for Section III.G of 10 CFR Part 50, Appendix R, which requires one train of each system necessary to achieve and maintain hot shutdown conditions, from either the control room or emergency control stations, to be free of fire damage. The licensee stated that it will meet the NFPA 805 requirement to achieve a safe and stable condition, instead of the 10 CFR Part 50, Appendix R, requirement to achieve and maintain cold shutdown within 72 hours. The NRC staffs determined that licensee aligns with the intent of NEI 00-01, Section 3.4.1.7, because the methodology in the DBNPS fire hazards analysis report is consistent with the guidance in NEI 00-01, Section 3.4.1.7, and the NFPA 805 requirement to achieve a safe and stable condition.

NEI 00-01, Section 3.4.2.4

NEI 00-01, Section 3.4.2.4, provides guidance for developing a compliance strategy to mitigate the effects of fire damage to each required component or cable. The licensee stated that the DBNPS fire hazards analysis report describes the fire area assessment process and contains the requirements, assumptions, methodology, and results of the evaluation. The report includes an evaluation of the fire protection features that assure safe shutdown capability as required by Section III.G of 10 CFR Part 50, Appendix R. The licensee clarified that emergent industry issues related to operator manual actions and MSOs are being addressed during the transition to NFPA 805 using the guidance provided in NEI 00-01 and NEI 04-02. The NRC staff determined that the licensee aligns with the intent of NEI 00-01, Section 3.4.2.4, because the licensee has developed compliance strategies to mitigate the effects of fire damage to required components or cables.

3.2.1.3 Not in Alignment with NEI 00-01 Sections, but No Adverse Consequences

In LAR Attachment B (Reference 6), the licensee indicated that it was not in alignment with NEI 00-01, Sections 3.4.1.5 and 3.5.1.2, but this did not result in any adverse consequences. However, in response to SSA RAI 04 (Reference 10), the licensee revised LAR Attachment B to state that it aligns with NEI 00-01, Section 3.5.1.2. Therefore, Section 3.5.1.2 of NEI 00-01 was considered in SE Section 3.2.1.1.

NEI 00-01, Section 3.4.1.5, provides criteria for the use of operator manual actions to mitigate the consequences of circuit failures. This section also refers to the criteria in NEI 00-01, Appendix H, for determining whether an operator manual action may be used for flow diversion off the primary flow path. The licensee stated that it did not classify cables in accordance with NEI 00-01, Appendix H, guidance. In LAR Attachment B (Reference 6), Section 3.4.1.5, the licensee stated that emergent industry issues related to operator manual actions and MSOs are being addressed during the transition to NFPA 805 using the guidance provided in NEI 00-01 and NEI 04-02. In addition, pre-transition operator actions will be transitioned as RAs consistent with NEI 04-02 and applicable FAQs. The NRC staff determined that the licensee aligns with NEI 00-01, Section 3.4.1.5, because the licensee will evaluate the effects of operator manual actions and MSOs in accordance with the guidance in NEI 04-02.

3.2.1.4 NFPA 805 Nuclear Safety Capability Assessment Methods Conclusion

As discussed above, the NRC staff reviewed LAR Section 4.2.1, LAR Attachment B, and the associated RAI responses regarding the process the licensee used to perform the NSCA required by NFPA 805, Section 2.4.2. The licensee followed the approach in NEI 04-02, as endorsed by the NRC. The process provided an organized structure that documented how the SSA aligns with each section of NEI 00-01, Chapter 3, which correspond to the requirements in NFPA 805, Section 2.4.2. For each section of NEI 00-01, Chapter 3, applicable to DBNPS, the licensee adequately justified that the SSA aligns with the section; aligns with the intent of the section; or is not in alignment, but there are no adverse consequences. Based on this review, the NRC staff determined that the process the licensee used to demonstrate compliance with NFPA 805, Section 2.4.2, is acceptable with respect to the selection of systems and equipment, selection of cables, and identification of the location of nuclear safety equipment and cables.

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. In the event of a fire, NFPA 805 requires the licensee to achieve and maintain the fuel in a safe and stable condition, which is more flexible than the requirement in 10 CFR Part 50, Appendix R, to achieve and maintain cold shutdown in 72 hours. NFPA 805 defines safe and stable conditions as follows:

> For fuel in the reactor vessel, head on and tensioned, safe and stable conditions are defined as the ability to maintain K_{eff} <0.99 [K_{eff} is the effective neutron multiplication factor], with a reactor coolant temperature at or below the requirements for hot shutdown for a boiling water reactor and hot standby for a pressurized water reactor. For all other configurations, safe and stable conditions are defined as maintaining K_{eff} <0.99 and fuel coolant temperature below boiling.

In LAR Section 4.2.1.2 (Reference 6), as supplemented, the licensee described how it will meet the NFPA 805 nuclear safety goal to achieve and maintain the fuel in a safe and stable condition. The licensee described its methods to maintain safe and stable conditions by addressing the five NSPC in NFPA 805, Section 1.5.1 (see SE Section 2.0): reactivity control, inventory and pressure control, decay heat removal, vital auxiliaries, and process monitoring. The licensee performed an at-power analysis (LAR Section 4.2.4 and Attachment C) and a non-power operation (NPO) analysis (LAR Section 4.3 and Attachment D) to demonstrate that it will meet the NSPC, the results of which are reviewed in SE Section 3.5.

In LAR Section 4.2.1.2, the licensee stated, in part:

> The NFPA 805 licensing basis for DBNPS for a safe and stable condition in the event of a fire starting with the reactor in at-power operating Mode 1 (Power Operation), Mode 2 (Startup), and Mode 3 (Hot Standby), is to maintain safe and stable conditions in Hot Standby up to the point at which the Decay Heat (DH) Loop is placed into service. DBNPS will maintain Hot Standby conditions until a decision is made to either place the reactor in a non-power operating mode, (i.e., Hot Shutdown Mode 4 or Cold Shutdown Mode 5) or to return to power operations. Determination of the final state will be based upon the extent of the fire damage, the inventory remaining in the Borated Water Storage Tank (BWST), the ability to provide makeup water to the BWST, and the ability to

re-establish inventory in the Condensate Storage Tanks (CSTs) or realign Auxiliary Feedwater (AFW) to its alternate sources. Refer to [LAR] Attachment C (Table B-3) for the systems and components credited with supporting safe and stable plant conditions by fire area.

As part of the transition to NFPA 805, each fire compartment was evaluated for maintaining safe and stable Hot Standby conditions. The evaluation has determined that DBNPS can achieve and maintain safe and stable conditions with the minimum shift operating staff. If required, the [auxiliary feedwater] pump suctions can be supplied from the Service Water (SW) System. The necessary valve manipulations to align these sources have adequate procedural guidance and are within the skills and training of the minimum shift operating staff.

With these required actions to maintain the plant in a safe and stable condition performed by the shift operating staff, there is sufficient time for the Emergency Response Organization (ERO) to respond and be available to assess plant conditions and determine the required actions necessary to extend safe and stable Hot Standby conditions. In the event it is determined that a plant cooldown to a non-operating mode is required, the ERO will determine the necessary actions, including maintenance and repairs that are necessary....

Reactivity Control

The licensee stated in LAR Section 4.2.1.2:

DBNPS reactor core design ensures that Keff [the effective neutron multiplication factor] is maintained < 0.99 while the plant is in a safe and stable condition, including compensation for any positive reactivity increases as a result of Xenon-135 decay and reactor coolant temperature decreases. Gravity insertion of the control rods into the reactor core will ensure reactivity control is achieved.

Reactor Coolant System (RCS) makeup will be from the BWST, which is a borated source that will ensure the Keff is maintained < 0.99 in all operating and non-operating modes.

Inventory and Pressure Control

The licensee stated in LAR Section 4.2.1.2:

Inventory makeup to the RCS will be required to account for nominal RCS leakage and RCS shrinkage due to cooldown as well as RCP seal injection. DBNPS has design features and procedures to ensure that an adequate source of borated inventory is provided for RCS inventory control from the BWST utilizing the makeup pumps (MUPs) or the high pressure injection (HPI) pumps. If BWST inventory is depleted, it will be refilled using a combination of makeup from the spent fuel pool or makeup tank.

With fuel in the reactor vessel, head on and tensioned, DBNPS has design features and procedures to ensure 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 reactor water level such that fuel cladding damage as a result of a fire is prevented.
>
> DBNPS has design features and procedures to ensure that excess RCS inventory is released from the RCS utilizing letdown or the pressurizer vent header.
>
> DBNPS has design features and procedures to ensure that excess RCS pressure relief is provided utilizing the pressurizer pilot-operated relief valve (PORV) or the pressurizer vent header.

In LAR Attachment G (Reference 15), the licensee identified a number of RAs associated with VFDRs that involve a loss of RCP seal cooling via the RCP seal injection flow path (e.g., VFDRs DB-1029, DB-1381, DB-1383). The licensee stated that within 8 hours of the loss of RCP seal cooling, the RAs require either manually aligning seal injection flow to all the RCP seals, manually align component cooling water flow to the RCP thermal barrier, or cooling down the RCS to a temperature between 280 degrees Fahrenheit (°F) and 350 °F. In response to SSA RAI 07 (Reference 10), the licensee stated that the three options to cool the RCP seals are considered DID RAs because the only action credited for risk reduction is tripping the RCPs. A discussion of the RCP seal modeling for the FPRA is provided in SE Section 3.4.3.2.

Decay Heat Removal

The licensee stated in LAR Section 4.2.1.2:

> DBNPS has design features and procedures to ensure reactor core [decay heat] will be rejected to the secondary plant through the steam generators (SG). The heat will be rejected to the atmosphere through the Atmospheric Vent Valves (AVVs).
>
> DBNPS has design features and procedures to provide adequate [auxiliary feedwater] to the credited [steam generators] for DHR [decay heat removal].

Vital Auxiliaries

The licensee stated in LAR Section 4.2.1.2:

> Each EDG is provided with a storage tank having a fuel oil capacity sufficient to operate that diesel for a period of seven days while the EDG is supplying continuous rating load demand. The EDG will provide power to the shutdown equipment for Reactivity Control, Inventory and Pressure Control, DHR, and Process Monitoring. Each EDG will also provide power to the other vital auxiliary systems.

Process Monitoring

The licensee stated in LAR Section 4.2.1.2:

> Adequate indications will be provided to the shift operating staff and ERO to ensure assessment can be made of plant conditions.

Based on the information in the LAR, the NRC staff concludes that the licensee has adequately described how it will meet the NFPA 805 nuclear safety goal to achieve and maintain the fuel in a safe and stable condition by describing how it will accomplish each of the NSPC. The licensee's NSCA identified the equipment, including associated electrical cables and controls, needed to meet the NSPC (see SE Section 3.2.1). In SE Section 3.5, the NRC staff provides its review of the results of the licensee's at-power and NPO analyses to demonstrate that it will meet the NSPC.

3.2.3 Applicability of Feed-and-Bleed

In 10 CFR 50.48(c)(2)(iii), the NRC limited the use of feed-and-bleed for FPPs base on NFPA 805, as follows:

> In demonstrating compliance with the performance criteria of [NFPA 805] 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 licensee stated in LAR Table 5-3 (Reference 6) that feed-and-bleed is not used as the sole fire-protected SSD methodology. The NRC staff reviewed the designated SSD path listed in LAR Attachment C for each fire area and confirmed that all fire compartment analyses included the SSD equipment necessary to provide DHR without relying on feed-and-bleed. Therefore, the NRC staff determined the licensee will meet 10 CFR 50.48(c)(2)(iii).

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 [Section] 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: "The [PRA] evaluation shall address the risk contribution associated with all potentially risk-significant fire scenarios." The licensee used FREs for its RI/PB approach, so its PRA evaluation must address all potentially risk-significant fire scenarios including potential MSO combinations to meet NFPA 805, Section 4.2.4.2, "Use of Fire Risk Evaluation."

In LAR Section 4.2.1.4, "Evaluation of Multiple Spurious Operations," and LAR Attachment F (Reference 6), the licensee stated that its process for identification and evaluation of fire-induced MSOs was conducted in accordance with NEI 04-02, RG 1.205, and FAQ 07-0038 (Reference 60), and included the following five steps:

1. identify potential MSOs of concern;

2. conduct an expert panel to assess plant specific vulnerabilities;
3. update the FPRA model and NSCA to include the MSOs of concern;
4. evaluate for NFPA 805 compliance; and,
5. document the results.

LAR Attachment F provided a summary for each step of the process, identifying information such as information sources, guidance, and results, as appropriate.

The NRC staff reviewed LAR Section 4.2.1.4 and Attachment F to determine whether the licensee adequately addressed MSO concerns. The licensee used a systematic and comprehensive process for identifying MSOs to be analyzed using available industry guidance. Based on the information provided in the LAR, the NRC staff determined that the process the licensee used provides reasonable assurance that the FRE appropriately identifies and includes risk-significant MSO combinations. Therefore, the licensee's approach for assessing the potential for MSO combinations is acceptable.

3.2.5 Establishing Recovery Actions

The NRC staff reviewed LAR Section 4.2.1.3, "Establishing Recovery Actions" (Reference 6), and LAR Attachment G (Reference 15) to evaluate whether the licensee will meet the associated requirements in NFPA 805 for the use of RAs.

NFPA 805 defines an RA as: "Activities to achieve the nuclear safety performance criteria that take place outside of the main control room or outside of the primary control station(s) for the equipment being operated, including the replacement or modification of components."

NFPA 805, Section 4.2.3.1, states:

> 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 [Sections] 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 [Section] 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 [PB] approach, the additional risk presented by their use shall be evaluated."

As described in LAR Section 4.2.1.3 and LAR Attachment G, the licensee used the guidance in NEI 04-02, RG 1.205, and FAQ 07-0030 (Reference 59) to establish the population of RAs being carried forward to the RI/PB FPP. As described in FAQ 07-0030, the process consisted of the following steps:

Step 1: Define the primary control stations and determine which pre-transition operator manual actions are taken at primary control stations. Activities that occur in the MCR are not considered pre-transition operator manual actions. Activities that take place at primary control stations or in the MCR are, by definition, not RAs.

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.

In LAR Attachment G, the licensee stated that it determined the primary control station based on the definition provided in RG 1.205 and by following the additional guidance in FAQ 07-0030. The licensee identified the auxiliary shutdown panel located in fire compartment R-01 as the only primary control station.

Operator manual actions meeting the definition of an RA are required to comply with the NFPA 805 requirements outlined above. Some of these operator manual actions may not be required to demonstrate the availability of a success path but may still be required to be retained in the RI/PB FPP because of DID considerations (NFPA 805, Section 1.2). Accordingly, the licensee indicated that DID RAs have been retained to provide plant operations with written guidance where such actions will enhance DID Echelon 3. This will provide additional assurance that one success path of safe shutdown capability can be restored if DID Echelons 1 and 2 become degraded or rendered ineffective.

In LAR Attachment G, the licensee stated that it performed a feasibility review of the risk reduction and DID RAs, listed in LAR Table G-1, using the guidance in NEI 04-02, RG 1.205, and FAQ 07-0030. The licensee stated that each of the feasibility criteria in FAQ 07-0030 were assessed for these RAs. FAQ 07-0030 provides criteria for demonstrations, systems and indications, communications, emergency lighting, tools and equipment, procedures, staffing, actions in the fire area, time, training, and drills.

LAR Attachment G, Table G-1, describes each RA associated with the disposition of a VFDR from the fire compartment assessments documented in LAR Attachment C. The licensee stated that a DID expert panel determined that all RAs listed in LAR Table G-1 are acceptable. The licensee completed a feasibility study that assessed all RAs (i.e., RAs in current procedures and new RAs needed for risk reduction or DID) against the NFPA 805 acceptance criteria. As part of Implementation Item DB-1941 (LAR Attachment S, Table S-2), the following actions will be completed during implementation of the proposed amendment:

- Procedures will be updated for the credited NFPA 805 RAs and fire compartment analysis results.
- Confirmatory demonstration of the feasibility for the credited NFPA 805 RAs will be performed after procedures are updated.
- Training will be updated after completion of the procedures.
- Fire brigade drills will be updated after completion of the procedures and training.

The activities performed by the fire brigade are generally focused on firefighting and fire suppression of a fire in the plant. In response to SSA RAI 06 (Reference 10), the licensee stated:

> Recovery actions credited to meet NSPCs will not be performed by any fire brigade member actively engaged in firefighting or post-fire cleanup. All credited recovery actions will be performed by assigned operations department personnel, and they will be directed by a licensed operations unit supervisor using approved safe shutdown procedures.
>
> Implementation item DB-1941 includes non-fire brigade changes to safe shutdown operations procedures and incorporate all new recovery actions required for transition to NFPA 805. In addition, it identifies updates to fire brigade training materials and drills. The following implementation items also track changes to fire brigade materials and are documented in LAR Attachment S, Table S-2: [DB-0341, DB-0538, DB-0557, DB-1074, DB-1093, and DB-1095.]
>
> These implementation items are collectively summarized as follows:
>
> The fire brigade pre-fire plans and training materials will be updated to include instruction for the containment and monitoring of potentially-contaminated fire suppression water and products of combustion. Updates of the pre-fire plans and training lesson plans will include guidance for the judicious use of fire hose spray and an awareness item for the potential of flooding within the fire compartment and water run-off to adjacent areas. There will also be updates of pre-fire plan information to include safe shutdown components and power supplies identified in the safe shutdown analysis. This may be used by operators as an aid to identify fire-affected equipment.

The NRC staff concludes that the licensee appropriately identified and evaluated RAs. As discussed above, the licensee used acceptable guidance in RG 1.205, NEI 04-02, and FAQ 07-0030 for the identification and evaluation of RAs. In addition, the licensee's proposed implementation items are acceptable.

3.3 Fire Modeling

NFPA 805 allows both FM and FREs 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 FREs 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 defines FM as a mathematical prediction of fire growth, environmental conditions, and potential effects on SSC based on conservation equations or empirical data.

In LAR Section 4.5.2, "Performance-Based Approaches," the licensee stated that the FM approach (NFPA 805, Section 4.2.4.1) was not used to demonstrate compliance with NFPA 805 for DBNPS. The licensee used the FRE method (i.e., FPRA) with input from FM analyses. The NRC staff reviewed the technical adequacy of the licensee's FREs, including the supporting FM analyses (SE Section 3.4.2), to evaluate the licensee's compliance with the NSPC.

3.4 Fire Risk Assessments

As allowed by NFPA 805, Section 4.2.4, the licensee used FREs to meet the NSPC as a PB alternative to the deterministic requirements in NFPA 805, Section 4.2.3. NFPA 805, Section 4.2.4.2, states:

> 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 [Section] 2.4.4.1. The fire risk shall be calculated using the approach described in [Section] 2.4.3.
>
> The proposed alternative shall also ensure that the philosophy of defense in depth and sufficient safety margin are maintained.

Section 2.4.3, "Fire Risk Evaluations," of NFPA 805 states that the PRA methods, tools, and data used to provide risk information for the PB evaluation of fire protection features or the change analysis described in Section 2.4.4, "Plant Change Evaluation," of NFPA 805 shall conform with the following:

1. The PRA evaluation shall use CDF and LERF as measures for risk. (NFPA 805, Section 2.4.3.1)

2. The PRA evaluation shall address the risk contribution associated with all potentially risk-significant fire scenarios. (NFPA 805, Section 2.4.3.2)

3. The PRA approach, methods, and data shall be acceptable to the NRC. They shall be appropriate for the nature and scope of the change being evaluated, be based on the as-built and as-operated and maintained plant, and reflect the operating experience at the plant. (NFPA 805, Section 2.4.3.3)

NFPA 805, Section 2.4.4, provides risk acceptance criteria, DID requirements, and safety margins requirements for plant change evaluations. Section 2.4.4.1, "Risk Acceptance Criteria," of NFPA 805 states:

> The change in public health risk from any plant change shall be acceptable to the [NRC]. CDF and LERF shall be used to determine the acceptability of the change.
>
> When more than one change is proposed, additional requirements shall apply. If previous changes have increased risk but have met the acceptance criteria, the cumulative effect of those changes shall be evaluated. If more than one plant change is combined into a group for the purposes of evaluating acceptable risk, the evaluation of each individual change shall be performed along with the evaluation of combined changes.

SE Section 3.4.1 provides the NRC staff's evaluation of the LAR with respect to NFPA 805, Section 2.4.4.2, "Defense-in-Depth," which requires that the plant change evaluation ensure that the philosophy of DID is maintained relative to fire protection and nuclear safety.

SE Section 3.4.2 provides the NRC staff's evaluation of the LAR with respect to NFPA 805, Section 2.4.4.3, "Safety Margins," which requires that the plant change evaluation ensure that sufficient safety margins are maintained.

SE Section 3.4.3 provides the NRC staff's evaluation of PRA quality as it relates to NFPA 805, Section 2.4.3.3. This evaluation includes consideration of the licensee's use of the PRA for post-transition FREs to support the self-approval process.

SE Section 3.4.4 provides the NRC staff's evaluation of the method used by the licensee for calculating the change in risk associated with the proposed transition to NFPA 805 with respect to the requirements of NFPA 805, Section 4.2.4.2.

SE Section 3.4.5 provides the NRC staff's evaluation of the additional risk presented by RAs, which, in accordance with NFPA 805, Section 4.2.4, must be evaluated by the licensee when the use of RAs resulted in the use of a PB approach.

SE Section 3.4.6 provides the NRC staff's evaluation of the risk associated with the proposed transition to NFPA 805 with respect to the risk acceptance criteria in NFPA 805, Section 2.4.4.1.

SE Section 3.4.7 provides the NRC staff's evaluation of PRA uncertainty with respect NFPA 805, Section 2.7.3.5, "Uncertainty Analysis," which requires that an uncertainty analysis be performed to provide reasonable assurance that the performance criteria have been met.

SE Section 3.4.8 provides the overall conclusions for the fire risk assessment methods, tools, and assumptions used to support transition to NFPA 805.

3.4.1 Maintaining Defense-in-Depth

The NRC staff reviewed the LAR, as supplemented, to determine whether the principles of DID will be maintained for the proposed transition to NFPA 805 at DBNPS. NFPA 805, Section 2.4.4.2, requires that the plant change evaluation ensure that the philosophy of DID is maintained relative to fire protection and nuclear safety. In its review, the NRC staff considered the DID requirements in NFPA 805, Section 1.2 (see SE Section 2.0), including the three echelons of DID:

(1) Preventing fires from starting;

(2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting damage; and

(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 safety functions from being performed.

The licensee provided a high level overview of its FRE process, including DID considerations, in LAR Section 4.5.2.2, "Fire Risk Approach" (Reference 6). In response to PRA RAI 16 (Reference 10), the licensee provided more details regarding its considerations for DID. The

licensee stated that it reviewed the impact of the change on DID using the guidance in NEI 04-02. The licensee described the evaluation for each of the three DID echelons, including identification of fire protection features and issues considered in the evaluation that impact fire risk. The evaluation determined whether there was an overreliance on an echelon of DID and whether changes were needed to ensure that each echelon of DID was achieved. Many of the identified fire protection features are required to demonstrate compliance with the fundamental FPP and design elements of NFPA 805, Chapter 3 (e.g., the combustible control and hot work control programs).

In response to PRA RAI 16, the licensee explained that it used the fire compartment CDF, LERF, and conditional core damage probability (CCDP) values to identify whether additional automatic fire detection systems should be credited for risk reduction or for DID. The licensee stated that two fire compartments would require automatic fire detection systems for DID.

LAR Table 4-3, as revised (Reference 10), provides a summary of the DBNPS NFPA 805 compliance basis and required fire protection systems and features for each fire compartment. LAR Attachment C (Reference 6), as supplemented, provides the results of the fire area transition review, and summarizes DBNPS compliance with NFPA 805, Chapter 4, for each fire compartment. Collectively, LAR Table 4-3 and Attachment C, as supplemented: (1) document the fire protection systems and features required to either meet the deterministic criteria of NFPA 805, Section 4.2.3, or to support the FPRA; (2) identify changes or improvements for each fire protection system and feature, if needed to maintain a balance among the DID echelons; and (3) provide justification that the required fire protection systems and features are adequate for DID.

Based on the information in the LAR, as supplemented, the NRC staff finds that the licensee systematically and comprehensively evaluated the fire hazards, area configuration, fire detection and suppression features, and administrative controls in each fire area. Therefore, the NRC staff concludes that the licensee's FPE process adequately evaluates DID for fires, as required by NFPA 805, and the proposed RI/PB FPP will adequately maintain DID.

3.4.2 Safety Margins

NFPA 805, Section 2.4.4.3, requires that the plant change evaluation ensure that sufficient safety margins are maintained. Consistent with RG 1.174, the guidance in Section 5.3.5.3, "Safety Margins," of NEI 04-02 (Reference 4) states that the following guidelines are acceptable for assessing whether safety margins would be maintained:

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

- safety analysis acceptance criteria in the licensing basis are met or sufficient margin is provided to account for analysis and data uncertainty.

The licensee stated that the FRE methodology was based on the requirements in NFPA 805 and guidance in RG 1.205 and NEI 04-02. LAR Section 4.5.2.2 (Reference 6) describes how safety margins were addressed as part of the FRE process. An FRE was performed for each fire area containing a VFDR. The FREs contain the details of the licensee's review of safety margins for each applicable fire area. LAR Section 4.5.2.2 states, in part, that:

> In assessing safety margin, the expert panel reviewed other analyses performed for the transition to NFPA 805 to ensure the fire protection equipment was adequate to meet applicable NFPA codes or exceptions to the NFPA codes were properly documented. The expert panel also reviewed fire modeling and Fire PRA outputs to ensure that these were factored into the FRE determination for adequate safety margin. The expert panel verified in each instance that the performance-based approach safety margin was equal to or greater than the deterministic approach employed per 10 CFR 50, Appendix R. In most instances, the performance-based approach safety margin was greater than the deterministic safety margin since issues associated with the deterministic approach were corrected during the NFPA 805 transition process.

In LAR Attachment J and in response to PRA RAI 16 (Reference 10), the licensee described the methodology used to evaluate safety margins in the FREs. The results of the licensee's safety margin assessment by fire area are provided in LAR Attachment C, as supplemented. The licensee established safety margins criteria for FM, plant system performance, and the FPRA logic model.

FM for the FPRA was, in general, developed using acceptable industry and NRC guidance, including NUREG/CR-6850 (References 33–35), NEI 04-02, and associated FAQ resolutions, as described in LAR Section 3.4 and specifically identified throughout the LAR. V&V performed in support of the FM used accepted codes and standards. Plant system performance was evaluated for specific demands associated with the postulated fire event, and the licensee determined that the safety margin established with the plant design-basis events will be maintained. The FPRA logic model, including supporting FM, was developed in accordance with NUREG/CR-6850, and ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 26).

The NRC staff considers the safety margin criteria used by the licensee acceptable because it was based on the requirements in NFPA 805 and guidance in RG 1.205 and NEI 04-02, which are consistent with the safety margins guidance in RG 1.174. Based on the information provided in the LAR, the NRC staff concludes that the licensee's approach has adequately addressed the issue of safety margins in the implementation of the FRE process, which will ensure that NFPA 805, Section 2.4.4.3, is met.

3.4.3 Quality of the Fire Probabilistic Risk Assessment

The objective of the NRC staff's PRA quality review was to determine whether the plant-specific PRA used to support the LAR is of sufficient scope, level of detail, and technical adequacy. The NRC staff evaluated the PRA quality information in the 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," LAR Section 4.7, "Program Documentation, Configuration Control, and Quality Assurance," LAR Attachment C, LAR Attachment U, LAR Attachment V, and LAR Attachment W, and associated supplemental information.

The licensee developed its FPRA model for both level 1 (core damage) and partial level 2 (large early release) PRA during at-power conditions. For the development of the FPRA, the licensee modified its internal events PRA model to include the effects of fire.

In LAR Section 4.8.2, "Plant Modifications and Items to be Completed During the Implementation Phase" (Reference 6), the licensee stated that the FPRA model included credit for the planned modifications in LAR Attachment S (Reference 16), and there are no other outstanding plant changes that would require adjustment to the FPRA model.

3.4.3.1 Internal Events PRA Model

In LAR Attachment U (Reference 6), the licensee stated: "The technical acceptability of the DBNPS PRA model has been demonstrated by the peer review process." In 2008, the licensee had a gap assessment of its internal events PRA performed, which compared the internal events PRA to the supporting requirements in ASME/ANS RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum B to ASME/ANS RA-S-2002" (Reference 88), as qualified by RG 1.200, Revision 1 (Reference 89). The licensee had a focused-scope peer review performed on its LERF modeling in October 2011 and its flooding PRA in July 2012.

In response to PRA RAI 04 (Reference 10), the licensee provided a comparison between the supporting requirements in ASME/ANS RA-Sb-2005 and ASME/ANS RA-Sa-2009. For every supporting requirement that was revised, the licensee explained why the internal events PRA met the newer supporting requirement. In response to PRA RAI 05 (Reference 10), the licensee clarified that the focused-scope peer reviews for the LERF and internal flooding PRAs followed the NEI 05-04 (Reference 27) process using ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2.

The ASME/ANS RA-Sa-2009 standard defines three capability categories (CCs) used to describe the capability of a PRA to support a particular RI application, with CC-I being the minimum capability, CC-II considered widely acceptable, and CC-III indicating the maximum achievable capability. These CCs differ in scope and level of detail, plant specificity, and realism of the PRA. For each CC, the supporting requirements in the ASME/ANS standard define the minimum requirements to meet that CC. For many supporting requirements, the CCs are combined, or the requirement may be the same across all CCs so that the requirement is simply met or not met.

Table U-1 in LAR Attachment U provides the licensee's disposition of findings from the 2008 gap assessment. Table U-2 provides the licensee's disposition of the facts and observations (F&Os) from the 2011 focused-scope peer review of the LERF PRA, and Table U-3 provides the licensee's disposition of the F&Os from the 2012 focused-scope peer review of the internal flooding PRA. The gap assessment findings indicated whether the internal events PRA complied with CC-II for each supporting requirement, and identified, if applicable, shortcomings in the internal events PRA associated with the supporting requirement. For the focused-scope peer reviews, an F&O was written for any supporting requirement judged not to be met or not fully satisfying CC-II of the ASME/ANS RA-Sa-2009 standard and RG 1.200, Revision 2. F&Os from the focused-scope peer reviews consisted of both findings and suggestions, as defined by the NEI 05-04 peer review guidelines.

In LAR Attachment U, the licensee provided the disposition of each gap assessment finding and F&O by either describing the resolution or assessing the impact of the resolution on the FPRA and the results for the NFPA 805 application. The NRC staff evaluated each gap assessment finding, F&O, and the licensee's disposition to determine whether they had any significant impact for the LAR. The NRC staff determined that the resolution of each gap assessment finding and F&O supports the determination that the quantitative results are adequate or have

no significant impact on the FPRA. Additional details regarding the NRC staff's review of the gap assessment findings for supporting requirements HR-A1 and LE-E4-01 are discussed below.

In LAR Attachment W (Reference 15), Section W.3.9, "Peer Review PRA Upgrades," the licensee stated: "For the internal events [PRA] model, closure of findings related to common cause modeling were determined to be upgrades, and were reviewed in a focused scope peer review performed after the end of the independent assessment in October 2017." The peer review assessed two supporting requirements and determined they were met (one at CC-III). The NRC staff determined that the performance of the focused-scope peer review for this PRA upgrade is consistent with the ASME/ANS RA-Sa-2009 standard, as qualified by RG 1.200.

For supporting requirement HR-A1, LAR Table U-1 states that the gap assessment found the modeling of pre-initiator HFEs was limited to "potentially important" events and noted that the risk significance of a pre-initiator HFE can be sensitive to the configuration modeled in the PRA. In response to PRA RAI 01 (Reference 10), the licensee described the criteria used to screen pre-initiator HFEs. The licensee stated that screening based on risk-significance was not performed, and this option was removed from the HRA documentation. The NRC staff reviewed the criteria used to screen pre-initiator higher risk evolutions (HREs) and found them to be reasonable. Although some pre-initiator HREs were screened, many other pre-initiator HREs were included in the FPRA. The licensee's screening of pre-initiator HREs is acceptable because it does not exclude significant risk contributors.

For supporting requirement LE-E4-01, LAR Table U-2 states that the gap assessment identified that it was difficult to determine if the breaking of circular logic was addressed in an appropriate manner. In response to PRA RAI 01, the licensee explained that for dependent systems a special gate containing all the failures in the supporting system were duplicated, except for the gate with the dependency, such that "no risk contributors are lost in the process." The licensee also stated in its response to PRA RAI 04 (Reference 10) that circular logic was broken using guidance from NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide" (Reference 90), with care to ensure no unnecessary conservatisms or nonconservatisms were introduced into the model. The NRC staff found that the licensee's treatment of circular logic is consistent with NRC guidance.

As discussed above, the NRC staff reviewed all the gap assessment findings and F&Os provided by the peer reviewers. The NRC staff determined that the resolution of every gap assessment finding and F&O supports the determination that the quantitative results are adequate or have no significant impact on the FPRA. The licensee's internal events PRA was reviewed against the applicable supporting requirements in ASME/ANS RA-Sa-2009. The NRC staff found that the licensee's internal events PRA is consistent with the guidance in RG 1.200, Revision 2, and that it is technically adequate to support the FREs and other risk calculations associated with the transition to NFPA 805. Therefore, the NRC staff concludes that the internal events PRA is adequate and can be used to support the FPRA.

3.4.3.2 Fire PRA Model

In LAR Attachment V (Reference 6), the licensee stated that a full-scope peer review of the DBNPS FPRA was performed in 2013 using the NEI 07-12 process (Reference 28). The review was performed against Part 4, "Requirements for Fires At-Power PRA," of ASME/ANS RA-Sa-2009 (Reference 26), as qualified by RG 1.200, Revision 2 (Reference 25).

Table V-1, "Fire PRA Facts and Observations," in LAR Attachment V, provides the licensee's dispositions of the F&Os from the 2013 full-scope peer review. Table V-2, "Fire PRA – Summary of Capability Category I SRs [Supporting Requirements]," identifies three supporting requirements that the peer review determined did not to meet CC-II and provides justification that CC-I is adequate to support the FPRA. Supporting requirements identified in Table V-2 are encompassed by F&Os presented in Table V-1.

In LAR Attachment W (Reference 15), Section W.3.9, the licensee stated that a finding related to fire-induced MSO modeling constituted an upgrade to the FPRA model, so a focused-scope peer review was performed. The peer review determined that the supporting requirements applicable to the MSO modeling were all met at CC-II or higher. The NRC staff determined that the performance of the focused-scope peer review for this PRA upgrade is consistent with the ASME/ANS RA-Sa-2009 standard, as qualified by RG 1.200.

The licensee dispositioned each F&O by assessing the impact of the F&O on the FPRA and on the results for the LAR. The NRC staff evaluated the licensee's disposition of each F&O provided in LAR Attachment V, including any revisions and supplemental information provided by the licensee, and determined that the licensee's disposition of each F&O is acceptable. Additional details regarding the NRC staff's review of certain F&Os and supplemental information provided by the licensee are discussed below.

Based on the information in the LAR, the NRC staff concludes that the FPRA is of sufficient technical quality. In addition, the NRC staff concludes that the quantitative results of the FPRA, in conjunction with the results of the sensitivity studies provided by the licensee, can be used to evaluate the change in risk due to the transition to NFPA 805 against the acceptance guidelines in RG 1.174. These conclusions are subject to the completion of the modification in Table S-1 and the implementation items in Table S-2 of LAR Attachment S (Reference 16). Therefore, the NRC staff will require the completion of these actions as part of the new FPP license condition (see SE Section 4.0).

F&O for Supporting Requirement PP-C3-01

In LAR Table V-1, the F&O for supporting requirement PP-C3-01 identified a lack of documented justification for non-rated fire barriers. In response to PRA RAI 02.a(i) (Reference 11), the licensee explained that it applied outdoor and indoor spatial separation to the FPRA. The outdoor spatial separation considered that a damaging HGL will not form, and this was credited when the radial distance between structures ensured that fire risk from one location would not contribute to the fire risk at another location. The indoor spatial separation considered the impact of fire propagation across open doorways, and this was credited for separating the containment annulus into two halves and for four annulus doors with solid lower-halves and metal-mesh upper-halves.

In response to PRA RAI 02.a(i), the licensee stated that the fire compartments for the annulus and areas connected by the four annulus doors are not capable of generating a damaging HGL. For the containment annulus, the boundary area between the two annulus halves is free of fixed or transient combustibles that could spread fire between the two halves. For the annulus doors, the area around the doors contains no ignition sources or intervening combustibles. However, in the MCA, the licensee conservatively assumed that fire-generated conditions propagate across the five open boundaries and assigned a barrier failure probability of 1.0.

The NRC staff concludes that the licensee's treatment of spatial separation is acceptable because the licensee only credited spatial separation outdoors or in large-volume indoor spaces with minimal combustible fuel loads. These are areas where the potential for developing a damaging HGL can be dismissed consistent with guidance in NUREG/CR-6850.

In response to PRA RAI 02.a(ii) (Reference 11), the licensee explained that there are three fire compartment boundaries that are protected by automatic deluge water systems activated by heat detectors located on either side of the boundary. For two of the boundaries, the purpose of the water curtain is to prevent fire propagation through openings created when a blowout panel opens on overpressure. The licensee explained that the heat detector set points (190 °F or 225 °F) are significantly less than the damage threshold for the thermoset cables (626 °F) used for the heat detector circuits. In addition, the heat detector circuits are routed in conduit providing additional thermal delay. The heat detectors will activate the water curtain prior to heat detector cable damage. The NRC staff determined that the heat detector set points are lower than the damage threshold for thermoset cables. The NRC staff concludes that the licensee's treatment of active fire barriers is acceptable because the licensee demonstrated that their performance is not degraded by the fire impact.

F&O for Supporting Requirement CS-B1-02

In LAR Table V-1, the licensee's disposition of the F&O for supporting requirement CS-B1-02 indicates that the beaker/fuse coordination for some circuits was not confirmed. The NRC staff notes that breaker/fuse coordination inadequacies are typically resolved rather than modeled in the FPRA.

In response to PRA RAI 02.b (Reference 11), the licensee explained that it modeled inadequate breaker/fuse coordination in the FPRA in one of three ways. For low-risk cables that were not specifically located, the licensee conservatively assumed that a fire in any compartment or scenario could result in fire damage to the cable. For cables not specifically located but known to exist within the same fire area as the cabinet, the licensee conservatively assumed that a fire anywhere in the fire compartment where the cabinet was located could impact cables associated with that cabinet. For circuits with known cable locations, the licensee determined fire impact based on FM. The licensee stated that in all cases it assumed that any supply breakers connected to cables impacted by fire fail to open, and the fault cascades to the next coordinated breaker (the feeder breakers) resulting in a loss of power to all loads powered from those sources. In response to PRA RAI 02.b(iii), the licensee explained that the two circuits where cable length was previously used to demonstrate breaker/fuse coordination were subsequently determined to be fully coordinated. Thus, breaker/fuse coordination no longer relies on cable length. The NRC staff finds that the licensee's modeling of inadequate breaker/fuse coordination in the FPRA is acceptable because it is done in either a realistic or conservative manner.

F&O for Supporting Requirement PRM-B7-01

The F&O for supporting requirement PRM-B7-01 (LAR Table V-1) states that a thermal-hydraulic analysis may be needed to support the assumption made in in the FPRA regarding a failure of the PORV to reclose following closure of main steam isolation valves or loss of main feedwater. The licensee's disposition of this F&O states that a simulator evaluation demonstrated that the PORV will not open, so the failure of the PORV to reclose was not added to the model. In response to PRA RAI 02.c (Reference 11), the licensee stated that it reviewed the UFSAR and determined that the peak RCS pressure, with a loss of feedwater and no

pressurizer spray, would lift the PORV. The licensee further stated that the FPRA model will include the failure of the PORV to reclose following a loss of feedwater. The licensee included this change in its integrated safety analysis provided in response to PRA RAI 03 (Reference 14), as supplemented (Reference 15). The NRC staff concludes that the licensee's resolution of this F&O is acceptable because the licensee added the PORV failure to the FPRA.

F&O for Supporting Requirement PRM-B7-02

The licensee's disposition of the F&O for supporting requirement PRM-B7-02 (LAR Table V-1) states that the PRA model management procedure requires the FPRA to be updated when modifications are implemented. In response to PRA RAI 02.d (Reference 11), as supplemented by its response to PRA RAI 02.d.01 (Reference 15), the licensee stated that, as part of Implementation Item DB-1695, the FPRA model would be updated as necessary to properly reflect the as-built, as-operated plant. In addition, if the updated risk results exceed the RG 1.174, Revision 2, risk acceptance guidelines, then more refined analysis will be performed, or plant modifications or procedure changes will be implemented before using the PRA model. The NRC staff finds that the actions to be taken as part of Implementation Item DB-1695 will adequately address the F&O.

F&Os for Supporting Requirements PRM-B9-01 and PRM-B13-01

The F&Os for supporting requirements PRM-B9-01 and PRM-B13-01 (LAR Table V-1) discussed the licensee's modeling of diverse and flexible coping strategies (FLEX) equipment in the FPRA. In its responses to PRA RAI 02.e (Reference 11), PRA RAI 02.01 (Reference 12), and PRA RAI 03 (Reference 14), the licensee provided additional information regarding the FLEX equipment and how it is credited in the FPRA.

In response to PRA RAI 02.e(i), the licensee identified the FLEX equipment and associated operator actions credited in the FPRA and described the conditions under which the FLEX equipment would be used in a fire scenario. The FLEX equipment credited in the FPRA includes the diesel-driven emergency feedwater pump, the alternate low-pressure emergency feedwater pump, two 480-volt diesel generators, and two FLEX RCS charging pumps. In response to PRA RAI 03, the licensee explained that although there is both a primary and an alternate 480-volt diesel generator, the alternate diesel generator is not explicitly modeled in the FPRA because of uncertainties associated with the reliability of the operator actions required to relocate this generator from its storage location. The alternate diesel generator is credited indirectly as a backup when the primary diesel generator is unavailable due to maintenance. During maintenance of the primary diesel generator, the alternate diesel generator will be relocated to the EFWF; therefore, maintenance unavailability for the primary diesel generator is not included in the FPRA. In response to PRA RAI 02.e(vi), the licensee stated that a fire in the compartments where the FLEX equipment is located is assumed to result in the failure of the equipment.

In response to PRA RAI 02.e(ii), the licensee stated that the fire procedures to install and operate FLEX equipment are not yet developed. These fire procedures will work together with the emergency operating procedures. The development of the fire procedures is included as part of Implementation Item DB-0572, and the update to the HRA (Implementation Item DB-1943) will encompass the fire procedure changes.

In response to PRA RAI 02.e(iii), the licensee explained that it evaluated the feasibility of operator actions for FLEX consistent with the guidance in NUREG-1921 (Reference 46). In

response to PRA RAI 02.e(iv), the licensee explained that the times required for performing operator actions were established by the FLEX feasibility study as part of the FLEX V&V. The times required to perform the operator actions were based on operator input, and the time available to put the FLEX equipment into service for different scenarios was based on MAAP computer program runs. The licensee identified the cues for putting each type of FLEX equipment into service and showed that for worst case conditions the time available to perform the operator actions exceeded the time required by a large margin (i.e., by at least a factor of three).

In response to PRA RAI 02.e(v), the licensee described the basis for the FLEX equipment failure rates. In response to PRA RAI 02.01, the licensee explained that the NRC component unreliability dataset for permanently-installed equipment (Reference 91) will be used for FLEX equipment until sufficient industry data is developed for estimating FLEX failure rates. The licensee justified using this dataset for FLEX equipment based on similarities of the FLEX equipment to permanently-installed equipment, such as the fact that the FLEX equipment at DBNPS does not need to be relocated and installed. The main difference between the FLEX equipment at DBNPS and permanently-installed equipment is that the FLEX equipment lines (e.g., hoses and electrical cables) need to be connected before use. The licensee accounted for this difference in the HRA by modeling operator errors associated with making these connections. In response to PRA RAI 03, the licensee stated that its dataset was updated to the 2015 version of the NRC's component unreliability dataset.

In response to PRA RAI 03, in reference to PRA RAI 02.01, the licensee described the results of a sensitivity study that showed the impact of using the component unreliability dataset for permanently-installed equipment is small. The total CDF, LERF, change in CDF, and change in LERF related to fire events were calculated in the sensitivity study using failure rates associated with the 5th percentile, mean, and 95th percentile of the probability distribution for a feedwater pump, diesel generator, and an RCS charging pump using the dataset for permanently-installed equipment. The results showed that the total CDF and LERF for the 5th and 95th percentile varied by about 1 to 2 percent from the mean values, while the change in CDF and LERF varied by less than 1 percent from the mean values. The NRC staff determined that the diesel generator failure rates used in the integrated analysis were based on the failure rates for combustion turbine generators (the FLEX diesel generators are diesel-fueled combustion turbine generators), as opposed to the lower failure rates for safety-related EDGs.

The NRC staff concludes that the licensee's modeling of operator actions and equipment failures for FLEX is acceptable. The NRC staff finds the FLEX operator actions to be feasible and can be accomplished in the time required for various accident conditions. In addition, the fire-related operator actions that use FLEX equipment will be reassessed (Implementation Item DB-1943) as the fire procedures are completed. The impact of a fire to FLEX equipment is explicitly addressed. The use of the NRC component unreliability dataset for permanently-installed equipment for FLEX equipment is reasonable because the main differences between the FLEX equipment and permanently-installed equipment is reflected through the modeling of operator errors. The licensee's sensitivity study also showed that the impact of using this dataset for FLEX equipment is too small to impact the application.

F&O for Supporting Requirement PRM-B15-01

The F&O for supporting requirement PRM-B15-01 (LAR Table V-1) states that a "possible accident progression beyond the onset of core damage has been identified." In response to PRA RAI 02.g (Reference 11), the licensee revised its disposition of the F&O for supporting

requirement PRM-B15-01. The licensee stated that subatmospheric containment pressure could lead to containment bucking, and that spurious closure of the containment vacuum breakers and spurious start of the containment spray pumps or actuation of the safety features actuation system could result in a subatmospheric pressure. The licensee stated that this scenario was included in the integrated analysis provided in the response to PRA RAI 03, as supplemented (Reference 15). In response to PRA RAI 03, the licensee explained that this scenario was conservatively modeled by assuming that actuation of the safety features actuation system would lead to containment buckling when all containment vacuum breaker isolation valves close and containment spray is initiated. The NRC staff finds that the licensee's resolution of this F&O is acceptable because the containment buckling scenario was conservatively modeled in the FPRA model.

F&O for Supporting Requirement FSS-C8-01

The licensee's disposition of the F&O for supporting requirement FSS-C8-01 (LAR Table V-1), states that the FM calculations credit the use of Kaowool. In response to PRA RAI 02.h (Reference 11), the licensee stated:

> Cable trays provided with solid bottom covers and Kaowool ceramic fiber blankets have been credited in the DBNPS fire PRA to prevent ignition of the cables within the trays and fire propagation. Kaowool is not credited to maintain cable functionality. Kaowool was not credited to prevent cable tray ignition if the tray did not have a solid bottom cover, if gaps in the blankets were identified, or if the Kaowool was within the zone of influence of a high hazard event (high energy arc fault, hydrogen, or transformer explosion).

The licensee also explained that the testing of cable trays, as documented in NUREG/CR-0381, "A Preliminary Report on Fire Protection Research Program Fire Barriers and Fire Retardant Coating Tests" (Reference 92), provides reasonable assurance that the Kaowool ceramic blankets prevent fire propagation. In addition, the licensee stated that the NFPA 805 monitoring program will be updated to maintain configuration control of the Kaowool blankets (Implementation Item DB-1744).

The NRC staff finds that the credit for Kaowool in the FPRA is acceptable because the credit is consistent with NRC guidance and configuration control of the Kaowool blankets will be maintained by the NFPA 805 monitoring program.

F&O for Supporting Requirement FSS-G2-01

The F&O for supporting requirement FSS-G2-01 (LAR Table V-1) discussed the need to consider cumulative risk as part of the screening criteria for MCA scenarios. In response to PRA RAI 03 (Reference 14) (which superseded its response to PRA RAI 02.i (Reference 11)), the licensee stated that screening of MCA scenarios based on scenario frequency is no longer performed for the FPRA. The licensee stated that all valid MCA scenarios are included in the FPRA regardless of scenario frequency. The NRC staff finds that the licensee's disposition of the F&O is acceptable because the concern regarding the cumulative risk of screened MCA scenarios has been eliminated.

F&O for Supporting Requirement IGN-A8-01

The F&O for supporting requirement IGN-A8-01 (LAR Table V-1) concerns the exclusion of self-ignited cable fires in containment. In response to PRA RAI 02.j (Reference 11), as supplemented by the response PRA RAI 03 (Reference 14), the licensee explained that the contribution of self-ignited cable fires in containment was included in the integrated analysis. The licensee stated that the ignition frequency for self-ignited cable fires in containment was determined by multiplying the ignition frequency of self-ignited cable fires plant-wide by the ratio of the weight of cables in containment to the weight of cables plant-wide. The licensee further stated that it conservatively applied the frequency for self-ignited cable fires in containment to the highest raceway CCDP within containment. The NRC staff finds that the licensee has adequately addressed this F&O because the licensee's treatment of self-ignited cable fires in containment is consistent with NRC guidance.

F&O for Supporting Requirement CF-A1-01

The F&O for supporting requirement CF-A1-01 (LAR Table V-1) concerned the use of guidance in NUREG/CR-6850 for the circuit failure likelihood analysis. This guidance had been superseded by guidance in NUREG/CR-7150, Volume 2 (Reference 44). In response to PRA RAI 02.k (Reference 11), as supplemented by the response PRA RAI 03 (Reference 14), the licensee stated that the integrated analysis provided in response to PRA RAI 03 used the uncertainty values for circuit failure and hot short probabilities based on guidance in NUREG/CR-7150, Volume 2. In addition, the modeling assumptions for panel wiring, trunk cables, and instrument cables were consistent with this guidance. The licensee added one instance of dc hot short duration modeling to the FPRA, which was a PRA upgrade, and included it in the integrated analysis. A focused-scope peer review of this PRA upgrade was performed, which resulted in no findings. The NRC staff finds that the F&O was adequately addressed because the circuit failure likelihood analysis was performed using the most current available NRC guidance.

Joint Human Error Probability

In reference to PRA RAI 06, the integrated analysis provided in response to PRA RAI 03 (Reference 14) used a lower limit of 10^{-5} for the joint human error probability (HEP). The NRC staff finds that the licensee's treatment of the minimum joint HEPs is acceptable because it is consistent with NRC guidance in NUREG-1792 (Reference 39).

Assumed Cable Routing

The F&O for supporting requirement PRM-B10-01 indicates that the FPRA assumed that equipment with untraced cables would fail in a fire. The NRC staff determined that this conservativism in the compliant plant model (described in SE Section 3.4.4) can lead to underestimation of the calculated change in risk for the transition. In response to PRA RAIs 07 and 15 (Reference 10), the licensee explained that four systems (i.e., instrument air, main feedwater, turbine plant cooling water, and circulating water) were not modeled in the FPRA due, in part, to the lack of cable routing information. The systems with untraced cables were assumed to fail in their worst failure mode in both the compliant and post-transition plant models. The licensee acknowledged that this assumption can lead to underestimation of the change in risk because the compliant plant risk is overestimated. In LAR Attachment W, Table W-4 (Reference 15), the licensee provided the results of an updated sensitivity study,

along with the updated risk results, where the assumed failures were removed from the compliant plant model.

The NRC staff determined that the sensitivity study overestimated the change in risk because: (1) the compliant plant risk is somewhat underestimated, since the conservatisms (i.e., the assumed failure of equipment with untraced cables) from the compliant plant model are removed, and (2) the post-transition plant model is acknowledged to be conservative, which results in an overestimation of the post-transition plant risk. The NRC staff concludes that the impact of these modeling conservatisms on the change in risk is acceptable because they are less than the conservatisms reflected in LAR Attachment W, Table W-4 (also see SE Section 3.4.7).

State-of-Knowledge Correlation

Section W.3.1 of LAR Attachment W (Reference 15) states that the risk estimates provided in LAR Table W-3 are based on point estimates rather than mean values, but that an uncertainty analysis was performed as a sensitivity study to support use of point estimates. In the variant case of the sensitivity study, mean values from parametric probability distributions were used, and a state-of-knowledge correlation was performed. In response to PRA RAI 08 (Reference 10), the licensee stated that it would include the state-of-knowledge correlation evaluation as part of the integrated analysis. The state-of-knowledge correlation evaluation included correlation of distributions for equipment failure rates, initiating events, HEPs (including dependencies), and circuit failure mode likelihoods. The licensee used the alpha and beta factors in NUREG/CR-7150 to define the beta distributions for hot short probabilities. The licensee further stated that the state-of-knowledge correlation will be retained for self-approval of post-transition changes. The integrated analysis was provided in response to PRA RAI 03 (Reference 14). The NRC staff found the correlation of fire sequence parameter distributions used in the integrated analysis acceptable because it included the parameters that could impact mean risk values.

Fire Modeling of Inaccessible Floor Space

LAR Attachment J (Reference 6), Table J-2 (p. J-19), states: "Transient and hot work fires were not postulated in locations within fire compartments that were considered inaccessible." An inaccessible area was defined, in part, as: "An area where access is prohibited or extremely difficult due to the presence of a permanent fixture ... and there is no credible reason to expect transient material to accumulate (e.g., areas on top of half height rooms, confined areas behind a floor to ceiling stack of cable trays with no expected reason for access)."

Guidance in NUREG/CR-6850 pertaining to exclusion of locations for transient fire analysis states:

> It is assumed that transient fires may occur at all areas of a plant unless precluded by design and/or operation.... Administrative controls significantly impact the characteristics and likelihood of transient fires, but they do not preclude their occurrence, since there is industry evidence of failure to follow administrative control procedures.

In response to PRA RAI 09 (Reference 10), the licensee explained that it would incorporate the transient fire scenarios for areas not explicitly precluded by design or operation, such as the top of half-height rooms or behind floor-to-ceiling cable tray stacks, into its integrated analysis. The

integrated analysis was provided in response to PRA RAI 03 (Reference 14). The NRC staff finds that the licensee's treatment of inaccessible areas not precluded by design or operation is acceptable because it is consistent with the guidance in NUREG/CR-6850.

Reduced Transient Heat Release Rates

LAR Attachment J (p. J-20) identifies three fire compartments (D-01, BF-01, and BG-01) where the FM used an HRR for transient fires that was less than 98th percentile HRR (317 kilowatts) in NUREG/CR-6850. The licensee stated that use of the lower HRR was based on NRC guidance provided in a June 21, 2012 (Reference 93), letter to NEI. In response to PRA RAI 10 (Reference 10), the licensee stated that the FM would be updated to use the 98th percentile HRR (317 kilowatts) recommended in NUREG/CR-6850 for fire compartments BF-01 and BG-01. The revised FM was used for the integrated analysis provided in its response to PRA RAI 03 (Reference 14). The use of the 98th percentile HRR (317 kilowatts) recommended in NUREG/CR-6850 for fire compartments BF-01 and BG-01 is acceptable.

LAR Attachment J stated that fire compartment D-01 (containment) used a 69-kilowatt HRR for transient fires. The response to PRA RAI 10 also stated:

> The containment entry procedures provide the requirements for entering containment (fire compartment D-01) at power. Foreign material exclusion and fire watch requirements are included. These requirements ensure all materials taken into the fire compartment are accounted for when leaving, thereby reducing the probability of leaving these materials unattended while at-power.

The licensee further stated that administrative controls, including daily inspections of accessible areas, ensure that if any combustibles are allowed into containment then they are removed prior to closeout. The NRC staff finds that the use of a reduced HRR for containment is acceptable because it is consistent with the guidance in the June 21, 2012, NRC letter to NEI.

Main Control Room Abandonment

In the original LAR (Reference 6), Section W.3.7 indicated that FM was performed to determine when the MCR needs to be abandoned due to loss of habitability, and that the CCDPs of associated scenarios were evaluated using a simplified fault tree model. LAR Table W-2 presented a single MCR abandonment scenario in the list of dominant fire scenarios but stated that this single scenario represents "the summation of all scenarios leading to abandonment." In response to PRA RAI 11.a (Reference 11), the licensee stated that it modeled required operator actions as a single HFE. The licensee also provided additional information in response to PRA RAI 11.01 (Reference 12). However, in the response to PRA RAI 03 (Reference 14), the licensee explained that it updated its approach to modeling MCR abandonment to address, in part, the large overestimation in LERF that resulted from the simplified modeling.

In response to PRA RAI 03, in reference to PRA RAI 12, the licensee stated that MCR abandonment due to loss of control is only directed for MCR or cable spreading room fires that cause one of the following conditions: (1) loss of all auxiliary feedwater, the motor-driven feed pump, and the emergency feedwater pump; (2) loss of both C1 and D1 essential buses; or (3) loss of both E1 and F1 essential buses. The licensee assumed that a fire in MCR cabinet C5715 would meet these conditions. The licensee modeled the same three core damage sequences that were used for the loss of habitability of the MCR.

In response to PRA RAI 03, the licensee stated that it modeled three specific core damage MCR abandonment sequences: (1) a transient-induced loss-of-coolant accident after a loss of DHR and failure of makeup or high-pressure injection, (2) a transient with the loss of DHR via the steam generators and failure of makeup or high-pressure injection cooling, and (3) a transient-induced small loss-of-coolant accident with the failure of high-pressure injection. The licensee stated that failure of operators to perform abandonment actions (e.g., actions at the alternate shutdown panel) were incorporated into these sequences, as applicable, depending on the fire damage.

In response to PRA RAI 03.01 (Reference 15), the licensee explained that it made simplifying assumptions for MCR abandonment scenarios that limited the number of sequences required to model them. The licensee assumed that an RCS leak is not recoverable from outside the MCR. In addition, feed-and-bleed cooling cannot be successfully performed from outside the MCR to recover from the loss of all feedwater. Scenarios involving LOCAs, steam generator tube ruptures, and interfacing LOCAs are not postulated to occur concurrently with a fire. The licensee initially excluded some scenarios because their maximum estimated CDF and LERF contribution was several orders of magnitude less than the total fire CDF and LERF. However, the licensee stated in its response to PRA RAI 03.01.a that it revised the PRA model to include these previously excluded scenarios for completeness. In response to PRA RAI 03.01.b, the licensee stated that the failure to abandon the MCR was added to the FM for two MCR abandonment sequences. The licensee's September 11, 2018, letter (Reference 15), also provided updated risk estimates for LAR Attachment W, Tables W-3 and W-4.

The licensee stated, in response to PRA RAI 03.01.c, that a handheld instrument credited in the FPRA would be used to obtain steam generator indications following a loss of power at the auxiliary shutdown panel. The licensee stated that the credit taken for using this instrument was based on the HEP associated with failure of the operator to obtain the desired indication. The licensee explained that the HEP bounds any equipment failure rates or unavailability of the instrument.

In response to PRA RAI 03, the licensee provided updated estimates of the CCDPs and CLERPS for MCR abandonment scenarios (see table below).[6]

	Post-Transition Plant Model	**Compliant Plant Model**
CCDP for loss of control	$1.51\times10^{-6} - 1.73\times10^{-1}$	$3.98\times10^{-6} - 1.00$
CCDP for loss of habitability	$1.90\times10^{-4} - 1.65\times10^{-3}$	$1.90\times10^{-4} - 2.21\times10^{-3}$
CLERP for loss of control	$4.54\times10^{-8} - 2.42\times10^{-1}$	$1.23\times10^{-8} - 1.00$
CLERP for loss of habitability	$4.44\times10^{-6} - 4.82\times10^{-4}$	$4.44\times10^{-6} - 1.98\times10^{-5}$

The licensee stated (see PRA RAI 03 reference to PRA RAI 13.b) that MCR abandonment scenarios are treated identically to all other scenarios when calculating the change in risk for the transition. One main control board was not included as a fire source for these scenarios because operators would abandon the MCR due to loss of control before the habitability thresholds would be reached. The CCDP and CLERP values for loss of control in the compliant plant model ranged to 1.00 because (1) auxiliary feedwater is unrecoverable during MCR abandonment if a fire-induced spurious signal blocks initiation of the steam feed rupture control system, resulting in the failure of the main steam isolation valves to close and depressurization of the steam generators, and (2) the emergency feedwater system is not credited in the

[6] The plant models are described in SE Section 3.4.4.

compliant plant model. The NRC staff determined that these CCDP and CLERP ranges reflect the range of fire-induced damage that can occur from MCR abandonment scenarios.

The NRC staff concludes that the licensee's treatment of MCR abandonment scenarios in the FPRA is acceptable because (1) it adequately addressed the complexities associated with a fire that results in abandonment of the MCR due to the loss of habitability or control and (2) the required operator actions are explicitly modeled in the applicable scenarios.

RCP Seals

In LAR Attachment G (Reference 6), the licensee identified three RAs associated with VFDRs that involved the loss of RCP seal cooling via the seal injection flow path. The licensee stated that within 8 hours of the loss of RCP seal cooling, the RAs require either manually aligning seal injection flow to all the RCP seals, manually align component cooling water flow to the RCP thermal barrier, or cooling down the RCS to a temperature between 280 °F and 350 °F.

In response to SSA RAI 07.01 (Reference 12), the licensee described the RCP seal model used in the FPRA, which is a slightly modified version of the model used for the DBNPS individual plant examination (Reference 94). The model assumes an RCP seal loss-of-coolant accident will occur if the RCPs are not tripped within a specific timeframe, which depends on whether there is a loss of seal return or a loss of seal cooling and injection. The simplified approach does not include the failure of the RCP seals after a successful trip of RCPs. The licensee indicated that the probability of mechanical failure of the seals after successfully tripping the RCPs upon loss of seal cooling is less than the probability of failing to trip the RCPs in time to protect the seals.

In its letter dated October 9, 2017 (Reference 13), the licensee provided the results of a sensitivity study and information about how the compliant plant model was developed for RCP seal modeling. The licensee stated:

> In variances from deterministic requirements (VFDRs) relating to a loss of seal cooling and seal injection, or seal return, the compliant cases were performed by setting the HFE for tripping the RCPs to false, thereby ensuring no seal failures occur. With no seal failures allowed to occur in the compliant case, it most likely underestimates compliant plant risk, since random, non-fire related failures that could cause RCP seal failures are also prevented in the compliant plant evaluation. Thus, the calculated change in risk is likely overestimated in those cases.

The licensee showed that its RCP seal model is dominated by operator errors, and that its compliant plant model contributes to overestimating the change in risk associated with RCP seal failure. Based on the results of the sensitivity study, the NRC staff determined that including RCP seal failures in the model, after the RCPs are tripped, does not significantly alter the results of the FPRA. The overall change in risk for the application is a large decrease in CDF (-1.65×10^{-4}/year) and LERF (-5.54×10^{-5}/year). Therefore, the RG 1.174 guidelines for the change in risk are unlikely to be challenged by uncertainties associated with the RCP seal model.

Single Compartment FM

In response to PRA RAI 03.e (Reference 14), the licensee identified a number of single-compartment fire scenario modeling changes to the FPRA, including adjustments to target sets and detection timing. The FM changes included incorporation of guidance from NUREG-2178 (Reference 48) for fire compartments Q-01 and S-01.

In response to PRA RAI 03.02 (Reference 15), the licensee stated that the single-compartment FM updates included an accumulation of minor FM refinements, incorporation of routine plant changes, and minor corrections identified from open items. The licensee stated that there were no changes in methodology or significant changes to PRA modeling scope or capability. The licensee also stated that for fire compartments Q-01 and S-01 the FM only incorporated the updated HRRs and gamma distributions from NUREG-2178, and it did not use the obstructed plume correlation. The NRC staff determined that the licensee's use of the guidance in NUREG-2178 does not represent a PRA upgrade as no new methods are employed.

The NRC staff finds that a focused-scope peer review for these FM changes is not necessary because the changes were made using methods and approaches that were included in the full-scope peer review of the FPRA and they do not represent significant changes to the scope or capability of the PRA model.

3.4.3.3 Fire Modeling in Support of the Development of Fire Risk Evaluations

NFPA 805, Section 2.4.3.3, requires that the PRA approach, methods, and data to be acceptable to the NRC. The licensee provided information regarding its FM in LAR Attachment J (Reference 6), and in supplemental responses. The NRC staff reviewed the FM used to support the FREs to confirm that the methods and approaches used for the application to transition to NFPA 805 were technically adequate. The NRC staff's review of compliance with NFPA 805 quality assurance requirements for analyses, calculations, and evaluations is provided in SE Section 3.9.3.

Overview of Fire Models Used to Support the FREs

The licensee used FM to develop the ZOI around ignition sources to determine the thresholds at which a target would exceed the critical temperature or radiant heat flux. The following algebraic fire models and correlations were used for this purpose:

- flame height – Heskestad method (Reference 95);
- plume centerline temperature – Heskestad method;
- radiant heat flux – point source radiation model (Reference 96); and
- ceiling jet temperature – Alpert method (Reference 97).

The first three algebraic models are also described in NUREG-1805 (Reference 40). Alpert's ceiling jet temperature correlation is also described in the EPRI "Fire Induced Vulnerability Evaluation Methodology (FIVE)" (Reference 98). These models serve as the basis for FDT^s that are used to estimate sprinkler and smoke detector response times (NUREG-1805

(Reference 40), Chapters 10 and 11, respectively). V&V of these algebraic models is documented in NUREG-1824 (Reference 41), Volumes 3 and 4.

In addition, the licensee developed screening approaches for the evaluation of ignition sources to determine the potential for the generation of an HGL in the compartment or fire area being analyzed. The FREs used these HGL screening approaches for 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 licensee used the McCaffrey, Quintiere, and Harkleroad correlation method for naturally ventilated compartments and the Beyler correlation method for closed compartments to determine the potential for the development of an HGL (Reference 99). These HGL correlation methods are also described in NUREG-1805, Chapter 2, and their V&V is documented in NUREG-1824, Volume 3.

In LAR Attachment J, the licensee also identified the use of the following empirical correlations that are not addressed in NUREG-1824, Volumes 3 and 4:

- plume radius - Heskestad method (Reference 95);
- smoke detection actuation correlation - Heskestad and Delichatsios method (Reference 40);
- sprinkler activation correlation (Reference 40);
- correlation for HRRs of cables (Reference 34);
- corner and wall HRRs described in Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Program," February 2005 (Reference 100); and
- correlation for flame spread over horizontal cable trays described in NUREG/CR-7010, Volume 1 (Reference 42).

The licensee used the plume radius (based on temperature) to calculate the horizontal radius of the plume at a given height. The licensee implemented all algebraic fire models and empirical correlations in a database and fire modeling workbook.

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 plant walkdowns to identify ignition sources and surrounding targets or SSC in compartments, and they applied the empirical correlation screening tool to assess whether the SSC 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 specific ignition sources that did not adversely affect the operation of credited SSC or targets following a fire. The licensee's screening was based on the 98th percentile fire HRR from the methodology in NUREG/CR-6850, Volume 2 (Reference 34), and, as discussed in its response to PRA RAI 03.02 (Reference 15), NUREG-2178 (Reference 48).

The Consolidated Model of Fire and Smoke Transport, Version 6 (Reference 101), was used for HGL temperature calculations, MCR abandonment calculations, and a temperature-sensitive equipment HGL study. The Fire Dynamics Simulator, Version 5 (Reference 102), was used for

a temperature-sensitive equipment ZOI study and a plume/HGL integration study. The licensee used the Pyrosim software package to create the Fire Dynamics Simulator input files. V&V of Consolidated Model of Fire and Smoke Transport and Fire Dynamics Simulator are documented in NUREG-1824, Volumes 5 and 7, respectively.

The V&V of the correlations and fire models that were used to support the FPRA is discussed in detail in SE Section 3.9.3.2.

NRC Evaluation of FM in Support of the DBNPS FPRA

LAR Attachment J provides the technical bases for the FM methods and approaches in Table J-2 and the damage thresholds used in the FM in Table J-3.

In LAR Attachment J (p. J-17), the licensee stated that to calculate the burning area the entire width of the cable tray was assumed to ignite, and the length of the tray assumed to initially ignite was determined by the length of the tray exposed to the fire. In response to FM RAI 01 (Reference 10), the licensee explained that the diameter of the ignition source was used as the initially ignited length of a cable tray located above the source. The licensee further explained that a cable tray heated by radiation was assumed to ignite over a length equal to the width of the tray. The NRC staff determined that these assumptions are either consistent with the guidance in Appendix R of NUREG/CR-6850, Volume 2, or they will result in conservative estimates of the HRR of a vertical stack of horizontal cable trays.

In LAR Attachment J (p. J-17), the licensee stated that for most areas the FM used the spread rates and the most conservative HRRs per unit area in NUREG/CR-6850, Table R-1, for each cable type in the fire growth analysis for cable trays. For some risk-significant fire scenarios, the FM was refined using the HRRs per unit area in NUREG/CR-7010. In response to FM RAI 01, the licensee provided the HRRs per unit area used for thermoset and Kerite-FR cable in both the initial and refined analysis. The NRC staff confirmed that these values are consistent with the guidance in Appendix R of NUREG/CR-6850 and Chapter 9 of NUREG/CR-7010, as applicable.

In LAR Attachment J (p. J-18), the licensee stated that if fire propagation to non-cable secondary combustibles was possible, it was included in the ignition source fire growth analysis. In response to FM RAI 01 (Reference 10), the licensee stated:

> All non-cable secondary combustibles were ignited at one minute, which corresponds with the shortest failure time identified in Appendix H of NUREG/CR-6850, Volume 2. The heat release rate (HRR) of the non-cable secondary combustibles was estimated based on the type and quantity. The [HRR per unit area] values were taken from Table 8-1 of NUREG-1805, "Fire Dynamics Tools," and the subsequent burning duration was determined utilizing NUREG-1805 Fire Dynamics Tool (FDT) 08. The total exposed area of the combustibles was determined by walkdowns. The non-cable combustibles were assumed to reach the peak HRR at one minute and remain steady for the entire burning duration.

The NRC staff finds that the licensee's approach to calculating fire propagation in non-cable secondary combustibles is acceptable because it leads to conservative estimates of the HRR.

In LAR Attachment J (p. J-19), the licensee described its process for placing transient fires in each compartment in the FPRA. In response to FM RAI 01, the licensee explained that transient fires were modeled as a fire with an area of 4 square feet and an elevation of 2 feet. The fire area was selected because it was a realistic representation of a transient fuel package and corresponds to a Froude number that is within the validated range of NUREG-1824. The licensee also stated that an elevation of 2 feet affects more targets than if the fire elevation is assumed to be at the floor. The NRC staff finds that the assumptions regarding the transient fire area and elevation are acceptable because they are consistent with the recommendations in NRC Inspection Manual Chapter 0609, Appendix F.

In its response to FM RAI 01, the licensee stated that the MCR has three different sizes of electrical cabinets with Plexiglas doors. These doors are not explicitly modeled in the MCR abandonment scenarios. The licensee demonstrated that the modeled HRR for the cabinets bounds the HRR contribution of the doors. Therefore, the NRC staff finds it acceptable to not explicitly model these doors in the MCR abandonment scenarios.

In LAR Section 4.5.1 (Reference 6), the licensee stated that the FPRA model for DBNPS was developed in compliance with Part 4 of ASME/ANS RA-Sa-2009, which requires damage thresholds to be established to support the FPRA. Thermal impacts must be considered in determining the potential for thermal damage of SSC, and appropriate temperature and critical heat flux criteria must be used in the analysis. In response to FM RAI 02 (Reference 10), the licensee identified two compartments that were not modeled as full-room burnout where an HGL could create a heat flux of 3 kilowatts per square meter in compartments that contain sensitive electronics. For one compartment, the HGL would be dispersed at the ceiling of the turbine deck, and it could not create a heat flux of 3 kilowatts per square meter where sensitive electronics are located. For the second compartment, the sensitive electronics are located in an enclosure, and, based on FAQ 13-0004 (Reference 68), the damage threshold for these sensitive electronics is equal to 11 kilowatts per square meter. The licensee determined that there are no relevant fire scenarios capable of generating an HGL heat flux that would meet or exceed this damage threshold. The NRC staff finds that the licensee's treatment of sensitive electronics in the lower layer of a compartment bounds the combined effect of the elevated temperature of the lower layer and thermal radiation from the HGL

The NRC staff reviewed the information in the LAR to confirm that the FM methods and approaches used to support the FREs for the transition to NFPA 805 were technically adequate. In SE Section 3.9.3.2, the NRC staff concludes that the licensee's V&V basis for the fire models and model correlations used in its FPRA provides reasonable assurance that the licensee's FM is appropriate and acceptable for use in the licensee's transition to NFPA 805. Therefore, the NRC staff concludes, in accordance with NFPA 805, Section 2.4.3.3, that the licensee's FM used to support the FREs is acceptable.

3.4.3.4 Fire PRA Model Quality During Self-Approval

As discussed in SE Section 2.6, 10 CFR 50.48(c) is intended to permit self-approval of certain changes to the FPP following the transition to an RI/PB FPP. The licensee proposed a license condition which includes the self-approval process (see SE Section 2.4.2), which is further discussed in SE Sections 2.6 and 4.0. The FPRA will support the self-approval of changes to the FPP. In accordance with NFPA 805, Section 2.4.3.3, the PRA approach, methods, and data must be acceptable to the NRC. As discussed in SE Section 3.4.3.2, the NRC staff found that the FPRA model is of sufficient technical adequacy for use to support transition to NFPA 805, subject to the completion of the modification in Table S-1 and the implementation items in

Table S-2 of LAR Attachment S (Reference 16). In LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," the licensee identified administrative controls it will use, both during and after the transition to NFPA 805, to maintain the FPRA model current with plant changes. Specifically, the FPRA will be integrated into the same process used to ensure configuration control of the internal events PRA model. In addition, the licensee has a program for ensuring that developers and users of these models are appropriately trained and qualified. Therefore, the NRC staff concludes that the licensee's PRA can be adequately maintained after the licensee fully implements the new FPP, such that it can be used for post-transition FREs to support the self-approval process. The NRC staff's review of the new FPP against the NFPA 805, Section 2.7.3, quality assurance requirements is provided in SE Section 3.9.3.

3.4.3.5 Conclusions Regarding FPRA Quality

Based on the information provided in the LAR, the NRC staff concludes that the licensee's PRA is consistent with the guidance in RG 1.174 (Reference 24), Section 2.3, and RG 1.205 (Reference 5), Section 4.3, regarding the technical adequacy of the PRA used to support risk assessment for transition to NFPA 805.

The NRC staff finds that the PRA model will adequately represents the as-built, as-operated configuration, as it will be configured after full implementation of NFPA 805; therefore, the PRA model is capable of being adapted to both the post-transition and compliant plant, as needed. The NRC staff also finds that the PRA model conforms to the applicable industry standards for internal events and fire PRAs at an appropriate CC. The NRC staff confirmed that the FM used to support the development of the FPRA is appropriate and acceptable. Therefore, the NRC staff concludes that the licensee's PRA approach, methods, and data are acceptable, as required by NFPA 805, Section 2.4.3.3.

Based on the licensee's administrative controls to maintain the PRA models current and assure continued quality, using qualified personnel, the NRC staff concludes that the PRA maintenance process is adequate to maintain the quality of the PRA to support self-approval of future RI changes to the FPP in accordance with the new FPP license condition.

3.4.4 Fire Risk Evaluations

As allowed by NFPA 805, Section 4.2.4, the licensee used FREs to meet the NSPC as a PB alternative to the deterministic requirements in NFPA 805, Section 4.2.3. In LAR Section 4.5.2.2 (Reference 6), the licensee stated that the FRE methodology was based on the requirements in NFPA 805 and guidance in Section C.2.2.4 of RG 1.205 and NEI 04-02. The NRC staff evaluation of the licensee's FREs was primarily based on information provided in LAR Section 4.5.2, LAR Attachment C, LAR Attachment W, and in related supplements.

Plant configurations that did not meet the deterministic requirements of NFPA 805, Section 4.2.3.1, were considered VFDRs. In LAR Attachment C (Reference 6), the licensee identified the VFDRs that will be retained and become part of the licensing basis. The licensee used the RI approach, in accordance with NFPA 805, Section 4.2.4.2, to demonstrate that the increased risk from the retained VFDRs is acceptable. VFDRs that will be brought into deterministic compliance through plant modifications do not require a risk evaluation.

In LAR Section 4.2.4 (Reference 6), "Fire Area Transition," the licensee stated that each VFDR was categorized as either a separation issue or a degraded fire protection system or feature.

Separation issues are generally associated with inadequate separation that could result in (1) fire-induced damage to process equipment or associated cables required for the identified success path; (2) spurious operation of equipment that may defeat the identified success path; (3) failure of process monitoring instrumentation or associated cables required for the identified success path; or (4) a combination of these issues. In response to PRA RAI 13.c (Reference 10), the licensee explained that it excluded VFDRs from the change-in-risk calculations if they were qualitatively determined not to impact CDF or LERF. The licensee provided a summary of VFDRs excluded from the change-in-risk calculations and the rationale for the exclusions. The NRC staff found that the excluded VFDRs involve failures that cannot impact the PRA success criteria, are negligible contributors to risk, or involve equipment not credited in the FPRA.

In LAR Attachment W (Reference 15), Section W.2.1, and in response to PRA RAI 13.a (Reference 10), the licensee described how the change in risk associated with VFDRs was determined. This involved the development of three FPRA models: a current plant model, a compliant plant model, and a post-transition plant model. The change in CDF and LERF associated with each fire area was obtained by calculating the difference between the compliant plant model and the post-transition plant model. The total change in risk was obtained by summing the change in risk for each fire area.

The current plant model represents the as-built, as-operated plant as it existed at the beginning of the transition to NFPA 805, so it does not include the plant modifications identified in LAR Attachment S (Reference 16), Tables S-1 and S-3. The compliant plant model was created by removing the impact on equipment due to fires identified for VFDRs from the current plant model. The licensee stated that the primary approach to removing VFDRs from the model involved "modeling the cables as being removed from the ZOI to represent a re-route of cables such that they are no longer located within the fire compartment of interest." For scenarios where RAs were credited, the model was revised to assume that the RA was always successfully executed. No additional logic or data was developed to create the compliant plant model.

For the post-transition plant model, the licensee started with the current plant model and incorporated the modifications identified in LAR Tables S-1 and S-3. If the post-transition CDF was less than 5×10^{-7}/year for a fire compartment, then the change in risk was conservatively set equal to the post-transition plant CDF. The licensee also calculated the risk offset, which is the risk reduction associated with modifications that reduce risk but do not bring the fire compartment into deterministic compliance with NFPA 805. The risk offset was calculated by subtracting the risk estimate results for the current plant model from the risk estimate results for the post-transition plant model.

In response to PRA RAI 03 (Reference 14), in reference to PRA RAI 13.b, the licensee stated that it updated the compliant plant model such that MCR abandonment scenarios were no longer treated differently from other fire scenarios in the FPRA (see SE Section 3.4.3.2 for more details).

The NRC staff finds that the licensee's methods for calculating the change in risk associated with VFDRs are acceptable because they are consistent with Section C.2.2.4.1 of RG 1.205 and FAQ 08-0054 (Reference 65). Based on LAR Attachment W (Reference 15), Table W-3, and in accordance with NFPA 805, Section 2.4.4.1, the NRC staff concludes that the difference between the risk associated with implementation of the deterministic requirements and the risk associated with retaining the VFDRs is acceptable.

3.4.5 Additional Risk Presented by Recovery Actions

The NRC staff reviewed LAR Attachments C, G, and W for its evaluation of the additional risk presented by the NFPA 805 RAs. As discussed in SE Section 3.2.5, the licensee used the guidance in NEI 04-02, RG 1.205, and FAQ 07-0030 to establish the population of RAs being carried forward in the RI/PB FPP. Based on this guidance, the licensee defined the primary control station as the auxiliary shutdown panel. Actions that take place in the MCR or at the auxiliary shutdown panel are not considered RAs.

LAR Attachment G (Reference 15), Table G-1, identifies a large number of RAs required to meet risk and DID criteria, and indicated which RAs were required to resolve VFDRs and which were credited for DID. LAR Table G-1 also identified actions taken at the auxiliary shutdown panel.

The additional risk of RAs for each fire area is presented in LAR Attachment W (Reference 15), Table W-3. In response to PRA RAI 14 (Reference 10), the licensee explained that if the fire compartment CDF was less than 5×10^{-7}/year, then the additional risk of RAs was conservatively set equal to the post-transition plant CDF. For fire compartments with a CDF greater than or equal to 5×10^{-7}/year, the additional risk of RAs was determined by calculating the contribution to the change in risk from scenarios that credited the RAs. This calculation provided the residual risk associated with using RAs versus completely resolving the VFDR, which is consistent with the guidance in FAQ 07-0030.

In LAR Table W-3, the licensee indicated that the total additional risk of RAs equates to a change in CDF of 2.51×10^{-4}/year and a change in LERF of 5.30×10^{-5}/year. Accordingly, the total additional risk of RAs exceeds the acceptance guidelines in RG 1.174, Revision 2, of 10^{-5}/year for a change in CDF and 10^{-6}/year for a change in LERF. Section C.2.2.4.2 in RG 1.205 states: "If the additional risk associated with previously approved recovery actions is greater than the acceptance guidelines in RG 1.174, then the net change in total plant risk incurred by any proposed alternatives to the deterministic criteria in NFPA 805, Chapter 4 (other than the previously approved recovery actions), should be risk-neutral or represent a risk decrease." The NRC staff found that application of this guidance to RAs in general (i.e., not solely to previously approved RAs) indicates that the proposed additional risk of RAs is acceptable, because the licensee reported a decrease in the total risk.

LAR Attachment S (Reference 16), Table S-2, also includes an action (Implementation Item DB-1943) for the licensee to update the HRA after procedures credited for transition to NFPA 805 are completed.

The NRC staff found that the licensee's methods for determining the additional risk of RAs are consistent with Section C.2.2.4.1 of RG 1.205 and FAQ 07-0030. Furthermore, the NRC staff found that there is a net decrease in fire risk resulting from non-VFDR risk reduction modifications. Therefore, the NRC staff concludes that the additional risk of RAs will meet the requirements of NFPA-805, Sections 4.2.4 and 2.4.4.1.

3.4.6 Cumulative Risk and Combined Changes

The NRC staff considered the guidance in RG 1.174 for combined change requests in its review because the LAR included several individual changes that were evaluated and will be implemented in an integrated fashion. The total CDF and LERF for a plant are generally

estimated by adding the results of the risk assessments for internal events, internal flooding, fire, seismic events, and other external hazard events. RG 1.174, Section 2.4, provides acceptance guidelines for RI applications. If an application clearly shows a decrease in CDF and LERF, then the change is acceptable. If the change results in an increase in CDF of less than 10^{-6}/year and an increase in LERF of less than 10^{-7}/year, then the change will be considered regardless of whether there is a calculation of the total CDF and LERF. If the increase in CDF is in the range of 10^{-6}/year to 10^{-5}/year and the increase in LERF is in the range of 10^{-7}/year to 10^{-6}/year, the application will be considered only if it can be reasonably shown that the total CDF is less than 10^{-4}/year and the total LERF is less than 10^{-5}/year. Applications that result in an increase in CDF greater than 10^{-5}/year or an increase in LERF greater than 10^{-6}/year would not normally be considered.

In LAR Attachment S (Reference 16), Tables S-1 and S-3, the licensee identified the modifications to DBNPS that have been or will be made to implement NFPA 805. These include modifications that remove VFDRs and modifications that reduce fire risk but do not bring the facility into compliance with the deterministic requirements of NFPA 805 (i.e., non-VFDR modifications). The licensee credited non-VFDR modifications by including them in the post-transition risk assessment but not in the compliant plant risk assessment.

Consistent with RG 1.174, the licensee did not report the total CDF and LERF contribution from internal and external events given the large decrease in CDF and LERF for the application. In LAR Attachment W (Reference 15), Table W-3, the licensee indicated that the FPRA estimated a CDF due to fire of 4.83×10^{-5}/year and a LERF due to fire of 3.97×10^{-6}/year for the post-transition plant model. Table W-3 also provided the change in CDF and LERF estimates for each fire area that was evaluated using an FRE. The risk estimates for these fire areas addressed the modifications and administrative controls that will be implemented as part of the transition to NFPA 805, including the RAs to reduce risk associated with VFDRs. The total change in CDF was -1.65×10^{-4}/year and the total change in LERF was -5.54×10^{-5}/year, which is considered acceptable per the RG 1.174 guidelines. The change in risk for each individual fire area was either negative or represented an increase of less than 10^{-5}/year for CDF and less than 10^{-6}/year for LERF. The NRC staff compared the licensee's risk estimates to the guidelines in RG 1.174, and finds the risk associated with the proposed transition to be acceptable.

RG 1.205, Section 3.2.5, states that risk decreases may be combined with risk increases for the purposes of evaluating combined changes with regulatory positions presented in Sections 1.1 and 1.2 of RG 1.174. LAR Table W-3 stated that the total CDF and LERF increases associated with unresolved VFDRs were 3.02×10^{-4}/year and 4.36×10^{-5}/year, respectively. In addition, the total CDF and LERF decreases associated with non-VFDR modifications were 4.67×10^{-4}/year and 9.90×10^{-5}/year, respectively. The NRC staff concludes that the change in risk is acceptable because combining the risk values for unresolved VFDRs and non-VFDR modifications results in a decrease in the CDF and LERF.

In response PRA RAI 15 (Reference 10), the licensee identified three key sources of modeling conservatisms in the compliant plant model. First, the FPRA assumed that instrument air, main feedwater, turbine plant cooling water, and circulating water systems failed in the worst failure mode, because no cable tracing was performed. Second, the circuit failure mode analysis was only performed for circuits contributing to the top cutsets, while the other circuits were assigned conservative failure probabilities in the FPRA. Third, the level of detail applied to FM was prioritized according to risk: (1) low-risk components were assumed to fail for every fire that propagated within the same fire compartment; (2) high-risk components were only failed if,

based on measurements, they were within the ZOI of the postulated fire; and (3) medium-risk components were failed if they were within a modified ZOI (ZOI radius multiplied by 1.5) of the postulated fire based on plant drawings.

In Table B-3 of its response to PRA RAI 03 (Reference 14), the licensee summarized the risk-significant scenarios for fire areas in the compliant plant model that are most impacted by risk reduction modifications and included descriptions, percent contribution, and dominant failures. In LAR Attachment W (Reference 15), Table W-4, the licensee provided the results of a sensitivity study in which the conservatisms identified above were removed from the compliant model but were retained in the post-transition plant model. The sensitivity study showed that with the modeling conservatisms removed there is still a large reduction in CDF and LERF (CDF and LERF are reduced by 8.13×10^{-5}/year and 4.21×10^{-5}/year, respectively). In response to PRA RAI 03, the licensee also identified additional modeling conservatisms affecting the compliant plant model that were not included in the sensitivity study results presented in LAR Table W-4. The impact of these additional modeling conservatisms on the change in risk is less than the conservatisms reflected in LAR Table W-4 (see SE Section 3.4.7).

The NRC staff compared the licensee's risk estimates to the guidelines in RG 1.174, and finds the risk associated with the proposed transition to NFPA 805 to be acceptable. In addition, the NRC staff finds that the licensee will meet the requirements of NFPA 805, Section 2.4.4.1, and concludes that the risk associated with the proposed alternatives to the deterministic criteria of NFPA 805 is acceptable.

3.4.7 Uncertainty and Sensitivity Analyses

As discussed in SE Section 3.4.6, the licensee's response to PRA RAI 15 (Reference 10) identified three assumptions that resulted in the underestimation of the change in risk and performed a sensitivity study in which it removed these conservatisms. LAR Attachment W (Reference 15), Table W-4, provided the results of a sensitivity study in which certain conservatisms were removed from the compliant model but were retained in the post-transition plant model. The NRC staff determined that the results of this sensitivity study provided a conservative estimate of the change in risk. The compliant plant risk is somewhat underestimated because all cited failures are removed from the compliant plant model, even though some failures may possibly occur. The post-transition plant model is conservative, and, therefore, provides an overestimation of the post-transition plant risk.

In its response to PRA RAI 03 (Reference 14), the licensee identified three additional modeling conservatisms affecting the compliant plant model that were not included in the sensitivity study. First, the licensee stated that additional analyses showed that a damaging HGL cannot form in the cable spreading room as assumed in the integrated analysis. Therefore, the assumption that equipment damage from such a fire is not recoverable by control room abandonment procedures is conservative. If these scenarios were removed from the FPRA models, the change to the transition and compliant plant CDF and LERF values would be negligible. Second, the licensee stated that fire impact to a particular power cable for the PORV would not lead to spurious opening of the PORV as assumed in the integrated analysis. The licensee performed a sensitivity study with this modeling conservatism removed and showed that the change in CDF and LERF would still be negative. Third, the licensee stated that the success logic for the makeup pump suction three-way valve is incorrect for a particular mode of operation during makeup/high-pressure injection cooling. The licensee provided the results of a sensitivity study, which showed that with this modeling conservatism removed there would be a larger reduction in CDF and LERF. The licensee stated that these modeling conservatisms will

be removed from the FPRA model, and included this action as Implementation Item DB-2120 in LAR Attachment S (Reference 16), Table S-2.

As discussed in SE Section 3.4.3.2, the licensee described the results of a sensitivity study for FLEX equipment in its response to PRA RAI 03. The licensee stated that it varied the assigned failure rates for FLEX equipment using the 5^{th} and 95^{th} percentile. The results showed that the total CDF and LERF values for the 5^{th} and 95^{th} percentile varied by about 1 to 2 percent from the mean values, while the change in CDF and LERF values varied less than 1 percent from the mean values.

Conservatism in the compliant plant model could lead to an underestimation of the change in risk. Based on the results of the licensee's sensitivity studies, the NRC staff finds that the risk decrease associated with planned modifications is much larger than the increase associated with underestimation of the change in risk for the transition due to modeling conservatisms. The NRC staff concludes that the modeling treatments discussed in this section are acceptable and do not impact the conclusions for this application.

3.4.8 Conclusion for Section 3.4

Based on the information in the LAR, as supplemented, regarding the fire risk assessment methods, tools, and assumptions used to support transition to NFPA 805, the NRC staff concludes that:

- The transition process included a detailed review of fire protection DID and safety margin. The licensee's evaluation of DID and safety margin are acceptable because the licensee followed the guidance in NEI 04-02, Revision 2, and RG 1.205, Revision 1, which provide an acceptable approach for meeting the requirements of 10 CFR 50.48(c).

- The licensee's PRA used to perform the risk assessments in accordance with NFPA 805, Sections 2.4.4 and 4.2.4.2, is of sufficient quality to support the application to transition to NFPA 805. The PRA approach, methods, tools and data are acceptable and are in accordance with NFPA 805, Section 2.4.3.3.

- The FPRA model may be used post-transition to support self-approval of FPP changes, subject to completion of all implementation items identified in LAR Attachment S, Table S-2, because the licensee will use methods acceptable to the NRC and the licensee's PRA maintenance process is adequate.

- The total updated additional risk from RAs is 2.51×10^{-4}/year for CDF and 5.30×10^{-5}/year for LERF, which is greater than the risk acceptance guideline in RG 1.174, Revision 2, of 10^{-5}/year for the change in CDF and 10^{-6}/year for the change in LERF. Based on the guidance in RG 1.205, Section C.2.2.4.2, the additional risk from RAs is acceptable because the licensee has reported a decrease in the total risk for this application.

- The LAR to transition to NFPA 805 is a combined change that includes risk increases from retained VFDRs and risk decreases resulting from non-VFDR related modifications. Based on the guidance in RG 1.174, the change in risk associated with the proposed alternatives to compliance with the deterministic criteria of NFPA 805 FREs is acceptable because the combined risk values result in a decrease in CDF and LERF for the application.

- The uncertainties associated with modeling treatments in the FRE do not impact the conclusions for this application.

3.5 Nuclear Safety Capability Assessment Results

In SE Section 3.2.1, the NRC staff provided its review of the LAR as it relates to the first three steps of the NSCA required by NFPA 805, Section 2.4.2. In SE Section 3.5.1, the NRC staff provides its review of the last step (i.e., step 4), which requires the licensee to assess the ability to achieve the NSPC (see SE Section 2.0) given a fire in each fire area. In SE Section 3.5.2, the NRC staff provides its review of the NSCA results for NPO at DBNPS.

3.5.1 Nuclear Safety Capability Assessment Results by Fire Area

NFPA 805, Section 2.2.4, "Performance Criteria," requires the NSPC to be examined on a fire area basis. In addition, NFPA 805, Section 2.2.3, "Evaluating Performance Criteria," states:

> 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 2.4.2.4, "Fire Area Assessment," requires an engineering analysis to be performed, in accordance with NFPA 805, Section 2.3, for each fire area to determine the effects of fire or fire suppression activities on the ability to achieve the NSPC. NFPA 805, Chapter 4, provides the methodology for both the deterministic and PB approach to meet the NSPC. NFPA 805, Section 4.2.2, states that the PB approach shall be permitted to use deterministic methods for simplifying assumptions within the fire area.

DBNPS is a single unit facility with 47 individual fire areas, including yard areas, and each fire area is composed of one or more fire compartments. The licensee performed the NSCA on a fire compartment basis. LAR Table 4-3 (Reference 6), as revised, provides a summary of the DBNPS NFPA 805 compliance basis and required fire protection systems and features for each fire compartment. LAR Attachment C (Reference 6), as supplemented, provides the results of the fire area transition review for at-power modes of operation, and summarizes DBNPS compliance with NFPA 805, Chapter 4, for each fire compartment. LAR Table 4-3 and Attachment C indicate that the PB approach allowed by NFPA 805, Section 4.2.4, was used for each fire compartment. For each fire compartment, LAR Attachment C included the following information:

- The regulatory bases for the new FPP, including fire detection and suppression systems required to me the NSPC.
- A performance goal summary of the method to accomplish each of the performance criteria in NFPA 805, Section 1.5, including identification of the SSC credited for achieving and maintaining fuel in a safe and stable condition.
- A summary of the evaluation of fires suppression effects on the NSPC.
- Licensing actions for exemptions that will transition to the revised licensing basis.
- EEEEs that will transition to the revised licensing basis.
- Disposition of VFDRs.

3.5.1.1 Fire Detection and Suppression Systems Required to Meet the NSPC

NFPA 805, Chapter 4, requires the licensee to determine, by analysis, what fire protection features and systems need to be credited to meet the NSPC. The NFPA 805 requirements for fire detection systems (Section 3.8.2), automatic water-based fire suppression systems (Section 3.9.1), gaseous fire suppression systems (Section 3.10.1), and passive fire protection features (Section 3.11) are dependent upon the results of the engineering analyses performed in accordance with NFPA 805, Chapter 4. These features and systems are only required when the analyses indicate that the features and systems are required to meet the NSPC.

The licensee performed a detailed analysis of fire protection features and identified the fire suppression and detection systems required to meet the NSPC for each fire compartment. For each fire compartment, LAR Table 4-3 (as revised) identifies if the fire suppression and detection systems installed in the compartment are required to meet criteria for separation, DID, risk, licensing actions, or EEEEs. NFPA 805, Section 4.1, requires that once a determination has been made that a fire protection system or feature is required to achieve the performance criteria of NFPA 805, Section 1.5, its design and qualification shall meet the applicable requirement of NFPA 805, Chapter 3.

In LAR Table 4-3 and LAR Attachment C, the licensee indicated that "other" fire protection features were credited in Licensing Action 8 to meet the NSPC. In response to SSA RAI 11 (Reference 10), the licensee stated that no fire protection systems are required under Licensing Action 8, and the licensee revised LAR Table 4-3 to clarify the licensing requirement for the associated fire compartment.

The NRC staff reviewed LAR Attachment C, as revised, for each fire area to ensure fire detection and suppression systems met the principles of DID for the planned transition to NFPA 805. Based on this review, the NRC staff concludes that the licensee adequately identified the fire detection and suppression systems required to meet the NFPA 805 NSPC on a fire compartment basis.

3.5.1.2 Evaluation of Fire Suppression Effects on the NSPC

NFPA 805, Sections 2.4.2.4 and 4.2.1, require that the effects of fire suppression activities on the ability to achieve the NSPC be evaluated for each fire area. For each fire compartment, LAR Attachment C included a discussion of how the licensee met the requirement to evaluate the fire suppression effects on the ability to meet the NSPC. For most fire compartments, the licensee stated that it evaluated the drainage capabilities and determined that there is adequate capability to remove the anticipated water from fire suppression activities to prevent immediate damage to equipment and avoid adverse consequences. For all fire compartments, the licensee concluded that fire suppression activities would not be expected to adversely affect achievement of the NSPC, would not affect the NSPC, or would negligibly affect achievement of the NSPC.

For fire compartment OS (outside areas), LAR Attachment C (p. 308) stated that impacts to equipment due to activation of the automatic fire suppression located in fire compartment OS-02 (rooms 001 and 002) are beyond the scope of the detailed FM analysis and are to be treated as an uncertainty. In response to SSA RAI 01 (Reference 10), the licensee stated that fire compartment OS-02 consists of the SBO diesel generator building, and the equipment in these rooms is not required to achieve the NSPC. In addition, the licensee revised LAR Attachment C to delete this statement.

Based on the information provided in LAR Attachment C, as supplemented, the NRC staff concludes that the licensee will meet NFPA 805, Sections 2.4.2.4 and 4.2.1, because the licensee evaluated the effects of fire suppression for each fire compartment and determined that fire suppression activities will not adversely affect achievement of the NSPC.

3.5.1.3 Licensing Actions

The LAR identified exemptions from specific requirements in 10 CFR Part 50, Appendix R, that the NRC previously granted for certain fire areas (see SE Section 2.5). The bases for the exemptions that will be transitioned to the new FPP are identified in the LAR as Licensing Actions 3, 8, 11, and 12. These exemptions are summarized in LAR Attachment C on a fire area basis, and described in further detail in LAR Attachment K (Reference 13). The licensee does not have any elements of the current FPP for which NRC clarification is needed.

Section C.2.3.2 of RG 1.205 states that existing exemptions can be transitioned as an alternative to meeting the deterministic requirements of NFPA 805, Section 4.2.3, if the technical basis for approval of the exemption or deviation is still valid. RG 1.205 further states that the NRC staff must determine that the licensee has acceptably addressed the continued validity of the exemption or deviation, and that the exemption or deviation does not involve an RA that is used to demonstrate the availability of a success path for the NSPC. As discussed in detail below, the NRC staff reviewed the current exemptions from 10 CFR Part 50, Appendix R, that will be transitioned with the NFPA 805 FPP. For each exemption, the LAR included a description, basis, and a statement regarding the continued validity of the exemption.

In the original LAR Attachment K (Reference 6), Licensing Action 2 identified an exemption related to component cooling water pump separation that was to transition to the new FPP. However, as discussed in SE Section 3.1.1.4, the licensee stated in response to SSA RAI 12 (Reference 10) that it will not transition this exemption to the new FPP.

Licensing Action 3

On August 20, 1984 (Reference 73), the NRC granted an exemption to Section III.G.2 of 10 CFR Part 50, Appendix R, to the extent that it requires separation of redundant SSD components by a 3-hour-rated fire barrier. The exemption was granted for door 215 which separates two auxiliary feedwater pumps located in adjacent rooms (fire compartments E-01 and F-01).

The NRC staff's evaluation for the exemption stated that door 215 is a pressure-rated door to protect against the consequences of high-energy line breaks. The NRC staff's evaluation stated, in part, that:

> ... an engineering evaluation has been performed [by the licensees] to determine the fire resistance of Door 215, simulating the fire test requirements of NFPA 251. The evaluation demonstrates that the door would permit a temperature rise on the unexposed face of 250°F when subjected to a 1300°F fire exposure for 25 minutes. The licensees have determined that the combustible material in either pump room would have a fire duration of less than 10 minutes. The licensees conclude that the door, if tested, would have a fire resistance significantly longer than the maximum postulated fire duration. We [the NRC staff] have reviewed the analysis and agree with the licensees. Due to low fuel load in the area and installed smoke detection system, there is

> reasonable assurance that an incipient fire would be detected promptly, and the response of the fire brigade to the [auxiliary feedwater] pump room would be expected in less than 25 minutes. It is our [the NRC staff] opinion that this combination of features provides reasonable assurance that one train of [auxiliary feedwater] pumps will remain free of fire damage.

In LAR Attachment K, the licensee stated that it validated the basis for the exemption by a review of station records, and confirmed that the basis for the exemption, as described in the NRC staff's evaluation, remains valid. The licensee stated that although the combustible loading values, as currently calculated, exceed the 10-minute duration, the hazards in the room have not changed. The NRC staff determined that the revised values for combustible loading remain low, and, therefore, do not invalidate the previous basis for approval.

Based on the previous NRC approval of this exemption and the licensee's confirmation that the basis for the exemption remains valid, the NRC staff concludes that Licensing Action 3 is acceptable for transitioning to the new FPP.

Licensing Action 8

By letter dated April 18, 1990 (Reference 74), the NRC granted an exemption to Section III.G.2 of 10 CFR Part 50, Appendix R, to addresses the lack of separation of redundant SSD train cables in manhole 3001 (fire compartment MA-01). Redundant train cables in this area are separated by less than 20 feet. In addition, the area is not provided with an automatic fire suppression system, and fire detection coverage is not provided throughout the area.

The NRC staff evaluation for the exemption relied on the following information provided by the licensee:

- There are no credible external sources of fire since the manhole is constructed with a concrete raised sill whose top opening is covered with a steel cap bolted into place.
- The insulation on the cables in this manhole satisfies the criteria of IEEE Standard 383-1974 (Reference 85), or its equivalent.
- The redundant cables are separated in accordance with the guidance in RG 1.75, "Physical Independence of Electric Systems" (Reference 103), and the cables are protected against protracted fault conditions by overcurrent devices.

In LAR Attachment K, the licensee stated that it validated the basis for the exemption by a review of station procedures, drawings, and design control documents, and confirmed that the basis for the exemption, as described in the NRC staff's evaluation, remains valid. Based on the previous NRC approval of this exemption and the licensee's confirmation that the basis for the exemption remains valid, the NRC staff concludes that Licensing Action 8 is acceptable for transitioning to the new FPP.

Licensing Action 11

By letter dated April 18, 1990 (Reference 74), the NRC granted an exemption to Section III.G.2.a of 10 CFR Part 50, Appendix R, to the extent that it requires separation of redundant SSD components by a 3-hour-rated fire barrier. LAR Attachment K states that this

exemption is for cables in conduits embedded in concrete in fire compartments A-08, DF-01, DH-01, EE-01, II-01, Q-01, R-01, and X-01.

The NRC staff evaluation for the exemption relied on the results of the licensee's engineering analysis, which demonstrated that the temperature of the embedded cables would not exceed 310 °F in a 30-minute period if a fire were to occur in any of the subject areas. In addition, the analysis showed that the cable temperature would reach a maximum of 370 °F and then diminish with commencement of fire suppression actions by the fire brigade. The combustible loading conditions and availability of automatic fire suppression system in some areas were also considered in the evaluation.

In LAR Attachment K, the licensee stated that it validated the basis for the exemption by performing limited walkdowns combined with a review of station procedures, drawings, and design control documents, and confirmed that the basis for the exemption, as described in the NRC staff's evaluation, remains valid. Based on the previous NRC approval of this exemption and the licensee's confirmation that the basis for the exemption remains valid, the NRC staff concludes that Licensing Action 11 is acceptable for transitioning to the new FPP.

Licensing Action 12

By letter dated August 20, 1984 (Reference 73), the NRC granted an exemption to Section III.O of 10 CFR Part 50, Appendix R, to the extent that it requires the RCP OCS to be designed to collect lube oil leakage in a closed, vented container that can hold the entire lube oil system inventory. In each RCS loop, the oil collection tank is sized to contain the oil from only one of the two RCPs. By letter dated January 30, 1998 (Reference 75), the NRC granted an exemption to Section III.O of 10 CFR Part 50, Appendix R, to permit the licensee to use the remote oil fill system without a collection system. These exemptions are associated with fire compartment D-01.

The NRC staff evaluation for the August 20, 1984, exemption states, in part, that:

> Since any lube oil overflow will drain to the containment building sump where there is no other flammable material or hot surfaces which may ignite the oil, the overflow oil will not present an exposure fire hazard to or otherwise endanger safety-related equipment, and since the RCP motor lube oil collection system is capable of withstanding the safe shutdown earthquake, we [the NRC] find the oil collection system acceptable.

The NRC staff evaluation for the January 30, 1998, exemption relied on compensatory actions that the licensee would take each time oil is added to the system and visual inspections that would be performed following refueling outages to confirm the integrity of the remote oil fill system.

In LAR Attachment K, the licensee stated that it validated the bases for the exemptions by reviewing station procedures, drawings, and design control documents, and confirmed that the bases for the exemptions, as described in the NRC staff's evaluation, remains valid. Based on the previous NRC approval of these exemptions and the licensee's confirmation that the bases for the exemptions remain valid, the NRC staff concludes that Licensing Action 12 is acceptable for transitioning to the new FPP.

3.5.1.4 Existing Engineering Equivalency Evaluations

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

> 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.

LAR Section 4.2.2, "Existing Engineering Equivalency Evaluation Transition" (Reference 6), states that the EEEEs that support compliance with Chapters 3 or 4 of NFPA 805 were reviewed by the licensee using the methodology contained in NEI 04-02. The licensee also listed the determinations made as part of the EEEE review.

Based on the guidance in RG 1.205 (Section C.2.3.2), NEI 04-02, and FAQ 08-0054, the licensee summarized the EEEEs that demonstrate that a fire protection system or feature is adequate for the hazard in the LAR. The licensee identified and summarized the EEEEs for each fire compartment, as applicable, in LAR Attachment C. The licensee did not request the NRC staff to review or approve any of these EEEEs.

Based on its 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 licensee's use of EEEEs meets the requirements of NFPA 805.

3.5.1.5 Variances from Deterministic Requirements

LAR Section 4.5.2.2 (Reference 6) states that plant configurations that did not meet the deterministic requirements of NFPA 805, Section 4.2.3.1, were considered VFDRs. The licensee used FREs, in accordance with NFPA 805, Section 4.2.4.2, to demonstrate that the increased risk from the retained VFDRs would be acceptable. VFDRs that will be brought into deterministic compliance through plant modifications did not require a risk evaluation. In LAR Attachment C, the licensee identified, characterized, and summarized the resolution for the VFDRs that will be retained and become part of the licensing basis.

The licensee stated that the acceptability of the VFDRs was based on the change in CDF and LERF and the maintenance of DID and safety margins. The licensee further stated that the FREs determined that the applicable risk, DID, and safety margin criteria were satisfied with the implementation of the specified RAs, completion of proposed modifications, or the existence of installed fire detection and suppression systems.

The NRC staff reviewed the VFDRs identified in LAR Attachment C, as supplemented, for all fire compartments. Based on this review, the NRC staff concludes that the licensee's identification and resolution of the VFDRs is acceptable because the licensee's analysis was performed consistent with the criteria in NEI 04-02, as endorsed by RG 1.205. Additional review of some VFDRs is discussed below.

VFDR DB-2033

In LAR Attachment C, VFDR DB-2033 states that instruments required in fire compartment A-08 may fail due to spurious containment spray, and the licensee would modify the transmitters to ensure that the instruments can provide accurate indications to the MCR following a spurious containment spray event. However, in response to FPE RAI 01.01.a (Reference 12), the licensee stated that the instrumentation listed in VFDR DB-2033 will not need to be modified to meet the NSPC. The licensee stated that an evaluation concluded that the components in VFDR DB-2033 are protected from an inadvertent containment spray, but some of the temperature elements were not qualified for submergence. Thus, in the event of inadvertent containment spray, operator actions to stop containment spray are required to prevent equipment submergence. Finally, in its response to PRA RAI 03 (Reference 14), the licensee stated that it determined that at least one success path of equipment for each VFDR would survive the containment spray actuation, and, therefore, no additional RAs were necessary.

VFDR DB-1184

In LAR Attachment G (Reference 15), the licensee identified an RA required for risk reduction in fire compartment G-02 that is associated with VFDR DB-1184. However, this VFDR was not identified in the NSCA summary for fire compartment G-02 in LAR Attachment C. In response to SSA RAI 08 (Reference 10), the licensee stated that VFDR DB-1184 is required and evaluated for fire compartment G-02. The additional risk associated with the RA required by this VFDR is included in the values reported in LAR Attachment W (Reference 15) for fire compartment G-02.

VFDR DB-2012

In LAR Attachment C, the licensee identified VFDR DB-2012 as applicable to all fire compartments. VFDR DB-2012 states that fire damage to installed makeup pumps could result in the loss of ability to maintain RCS inventory and pressure control, which could challenge the NSPC. The licensee stated that this VFDR will be corrected by plant modifications to install additional RCS charging pumps, connections, and associated auxiliaries. However, LAR Attachment S did not list DB-2012 in the list of plant modifications. In response to SSA RAI 10 (Reference 10), the licensee stated that in every deterministic case, makeup or high-pressure injection remains available, and the NSPC are met without VFDR DB-2012. Therefore, the licensee closed DB-2012 and no RAs are credited to resolve the concern.

The licensee's response to SSA RAI 10 also clarified that the installation of the FLEX RCS makeup pumps (Implementation Item DB-1983) is credited by the FPRA as an overall plant design modification, but it is not credited to meet any NSPC. In response to SSA RAI 10.01 (Reference 12), the licensee stated that, where non-deterministic methods evaluated separation for makeup or high-pressure injection, VFDRs were identified and dispositioned in the applicable FREs. The FPRA model and the FREs credit the FLEX modifications for risk offset, and any actions necessary to use the FLEX equipment would be considered other actions that do not involve the success path. In addition, the licensee stated:

> As a modification credited in the fire PRA for risk reduction, operation of the FLEX equipment is incorporated into the human reliability analysis (HRA) for the fire PRA.

A feasibility study was completed based on NUREG-1921 guidance on all FLEX-related operator actions and was found to be satisfactory. Modeling was done for FLEX-related human failure events (HFEs) that are performed within one hour, using NUREG-1921 guidance, and it was determined to be an acceptable modeling approach. For HFEs performed after one hour, the internal events human error probability was used without alteration. The feasibility also established timelines for operator actions that implement the FLEX actions, and indicated an adequate margin for inclusion into the fire PRA model.

FLEX RCS makeup pumps are credited for risk offset. Actions expected to place the pumps in service are not recovery actions because they are not credited for the one success path that is deterministically evaluated on a fire area by fire area basis. These other operator actions that do not involve the success path have been evaluated for feasibility, as explained above, and therefore meet the risk, defense-in-depth and safety margin criteria for NFPA 805.

The NRC staff finds that for the fire compartments where non-deterministic methods were used to evaluate separation for makeup or high-pressure injection, the FLEX RCS makeup pumps meet the risk, DID, and safety margin criteria for NFPA 805.

3.5.1.6 Recovery Actions

The NRC staff reviewed LAR Section 4.2.1.3, "Establishing Recovery Actions" (Reference 6), and LAR Attachment G (Reference 15) to evaluate whether the licensee will meet the NFPA 805 requirements for the use of RAs. The details of the NRC staff's review of RAs are described in SE Section 3.2.5. The NRC staff's evaluation of the additional risk of RAs credited to meet the risk acceptance guidelines is provided in SE Section 3.4.5. LAR Attachment G lists the RAs identified in the resolution of the VFDRs described in LAR Attachment C. The RAs identified include both actions considered necessary to meet risk acceptance criteria and actions relied upon for DID.

In LAR Attachment G, the licensee stated: "The defense in depth recovery actions have been conservatively retained to provide plant operations with written guidance where such actions will enhance Echelon #3 of defense in depth, to provide additional assurance that one success path of safe shutdown capability can be restored in the event that Echelon #1 and Echelon #2 of defense in depth become degraded or rendered ineffective." Therefore, the NRC staff determined that these DID RAs will be part of the RI/PB FPP, and subject to a plant change evaluation if subsequently modified or removed.

In LAR Attachment C, the licensee identified an RA in the disposition of VFDR DB-1923 for fire compartment II-01, which was not included in LAR Attachment G. VFDR DB-1923 states that fire damage could result in the loss of ability to trip the main turbine from the MCR, which could challenge the NSPC. In response to SSA RAI 09.a (Reference 10), the licensee stated that the usual RA for DB-1923 is to trip the turbine locally. However, the local manual trip on the turbine front standard is located within fire compartment II-01. The licensee stated that the analyzed action for this VFDR is to close the main steam isolation and bypass valves from the MCR. Therefore, this action is not an RA, but it is a DID action in the FRE. In response to SSA RAI 09.01 (Reference 12), the licensee confirmed that the operator actions required can be fulfilled in the MCR and revised LAR Attachment C to indicate that no further action is required for the disposition for VFDR DB-1923. In addition, the licensee revised LAR Attachment G to

remove VFDR DB-1923 from fire compartments CC-01, DF-01, II-04, and U-01, since the same operator action from the MCR can be accomplished for all five fire compartments.

In LAR Attachment C, the licensee also identified RAs in the disposition of VFDRs DB-1227 and DB-1268 for fire compartment V-01, which were not included in LAR Attachment G. In response to SSA RAI 09.a, the licensee stated that these VFDRs were duplicates and removed them from LAR Attachment C for fire compartment V-01.

In response to SSA RAI 09.b (Reference 10), the licensee confirmed that there were no other discrepancies between the RAs credited to meet the risk, safety margin, and DID acceptance criteria in the VFDRs identified in LAR Attachment C and those identified in LAR Attachment G.

3.5.1.7 Plant Fire Barriers and Separations

Except for the electrical raceway fire barrier systems (ERFBS), passive fire protection features include the fire barriers used to form fire compartment boundaries (and barriers separating SSD trains) that were established in accordance with the current FPP. For the transition to NFPA 805, the licensee retains previously established fire compartment boundaries as part of the RI/PB FPP.

Fire compartment boundaries are established for those areas described in LAR Attachment C, as modified by applicable EEEEs that determined 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 was evaluated as part of the NRC staff's review of LAR Attachment A (SE Section 3.1).

3.5.1.8 Electrical Raceway Fire Barrier Systems

In LAR Attachment A, the licensee indicated that the ERFBS meet the deterministic requirements of NFPA 805, Chapter 3. Each fire compartment using ERFBS is identified in LAR Table 4-3 and in LAR Attachments A and C (Reference 6). The licensee analyzed the ERFBS using the PB approach in accordance with NFPA 805, Section 4.2.4. There were no VFDRs associated with ERFBS.

In LAR Table 4-3, the licensee identified that 1-hour fire-rated ERFBS and cable tray systems protected on three sides by metal covers and protected on top by ceramic fiber (e.g., Kaowool) are credited to protect selected cables for risk reduction. These systems are included in the table as "other" types of fire protection features. In LAR Table 4-3, the licensee also identified that 1-hour fire-rated ERFBS in fire compartment A-08 are credited to protect cables that are required to achieve and maintain the NSPC.

In response to SSA RAI 05 (Reference 10), the licensee stated:

> In Fire Compartment A-08, along with other fire compartments, electrical cable raceways are protected from fire by the following three configurations:
>
> - Selected electrical raceways are configured and credited for NFPA 805 transition with ERFBSs;
>
> - Other electrical raceways in Fire Compartment A-08 (along with other compartments) are protected by a cable tray configuration

consisting of three solid metal sides (two solid sides and solid bottom) with a ceramic fiber blanket over the top; except

- Fire compartment II-01 has electrical raceways consisting of three solid metal sides (two solid sides and solid bottom) and no ceramic fiber over the top of the electrical cables.

The fire barriers identified in the "ERFBS" column of Table 4-3 are tested configurations, and the insulation and fire properties provide protection from fire damage to the electrical cables for a time duration greater than or equal to its configuration acceptance test duration.

The electrical raceway configuration of three-sided metal (sides and bottom) and topped with ceramic fiber material is not a tested configuration, but provides reasonable assurance to limit the electrical cable jacketing from becoming exposed to and ignited during a fire. The three metal sides provide reasonable assurance to limit the electrical cable jacketing from becoming exposed to fires initiating at lower elevations by acting as radiant heat shielding to delay the propagation of cable damage in the electrical raceways above the fire. Therefore, it is used within the analysis to delay subsequent electrical tray ignition. This configuration is not identified in the "ERFBS" column of Table 4-3, but is identified in the "Other" column, along with an appropriate note.

Based on a review of the information in the LAR, the NRC staff finds that the licensee has adequately identified the ERFBS configurations that are credited to meet the deterministic requirements of NFPA 805, Chapter 4, and the other configurations credited in the FM, PRA, and FREs to meet the PB requirements of NFPA 805, Chapter 4.

3.5.1.9 Conclusion for Section 3.5.1

As discussed in LAR Table 4-3 and Attachment C, the licensee used the PB approach in accordance with NFPA 805, Section 4.2.4, for all fire compartments. The NRC staff concludes that each fire compartment has been properly analyzed and meets the applicable requirements of NFPA 805, Section 4.2, "Nuclear Safety." This conclusion is based on the following:

- The licensee adequately identified the fire detection and suppression systems required to meet the NFPA 805 NSPC on a fire compartment basis.
- The licensee adequately evaluated the effects of fire suppression on the NSPC and determined that the NSPC will not be adversely affected.
- The current exemptions from 10 CFR Part 50, Appendix R, that are being transitioned to the NFPA 805 licensing basis (Licensing Actions 3, 8, 11, and 12) are acceptable for transition.
- The licensee's use of EEEEs meets the requirements of NFPA 805.
- The VFDRs were adequately evaluated and found to be acceptable based on either (1) an integrated assessment of risk, DID, and safety margins, or (2) plant

modifications or RAs were identified that will be implemented to address the issue.

- The licensee adequately identified and evaluated RAs used to demonstrate the availability of a success path to achieve the NSPC (SE Section 3.2.5). The additional risk of RAs is acceptable (SE Section 3.4.5).

- Fire area boundaries (ceilings, walls, and floors), fire barriers, fire barrier penetrations, and through penetration fire stops were found to be acceptable.

- The licensee adequately identified the ERFBS configurations that are credited to meet the deterministic requirements of NFPA 805, Chapter 4, and the other configurations credited in the FM, PRA, and FREs to meet the PB requirements of NFPA 805, Chapter 4

3.5.2 Fire Protection During Non-Power Operational Modes

NFPA 805 is applicable to all plant operating modes including NPO modes. Thus, the nuclear safety goals, objectives, and performance criteria of NFPA 805 (SE Section 2.0) must be met during NPO modes. The NRC staff reviewed LAR Section 4.3, "Non-Power Operational Modes," and LAR Attachment D (Reference 6) to evaluate the licensee's treatment of potential fire impacts during NPO modes. In LAR Section 4.3, the licensee stated that it used the process outlined in NEI 04-02 and FAQ 07-0040 (Reference 63) to demonstrate that the NSPC are met during NPO modes.

3.5.2.1 Non-Power Operation Strategy and Plant Operating States

In LAR Section 4.3, the licensee stated that it used the process outlined in NEI 04-02 and FAQ 07-0040 (Reference 63) to demonstrate that the NSPC are met during NPO modes. LAR Attachment D indicates that the licensee's procedures use the DID concept to minimize shutdown risk. The licensee stated: "Identification of high-risk evolution is achieved by the use of administrative and operational procedures that have requirements and restrictions that ensure acceptable levels of risk and defense in depth are maintained based on reactor coolant system and fuel pool inventory, decay heat removal capability, and time to core boil." HREs during plant outages are activities or plant conditions which would make the plant more susceptible to an event causing the loss of a key shutdown DID function.

The licensee stated (LAR Attachment D) that it selected two plant operational states for review to identify the NPO systems and components required for shutdown HREs. A third plant operational state was selected for review to identify any electrical components that could potentially spuriously operate and result in an RCS inventory loss.

3.5.2.2 NPO Analysis Process

In LAR Section 4.3.1, the licensee stated:

> The goal ... is to ensure that contingency plans are established when the plant is in an NPO mode where the risk is intrinsically high. During low-risk periods, normal risk management controls and fire prevention/protection processes and procedures will be used.

In LAR Section 4.3.2, the licensee stated:

> Based on FAQ 07-0040 (Revision 4), the Plant Operating States (POS) considered for equipment and cable selection are defined in ARS-DB-11-003, "Non-Power Operational Modes Transition Report." Components were identified to support the KSFs [key safety functions] of Inventory Control, Decay Heat Removal Capability, Reactivity Control, Containment Closure, and associated support functions (process cooling and electrical power). A model was developed in the NFPA 805 Analysis Database (Genesis Solution Suite, SAFE Module). Equipment was logically tied to the supported KSF. Power supplies, interlocks, and supporting equipment were logically tied to their parent component.
>
> For those components which had not been previously analyzed in support of the at-power analysis or whose functional requirements may have been different for the non-power analysis, cable selection was performed in accordance with approved project procedures. Cables necessary to support the selected function of a component were selected and analyzed for fire impact.
>
> ARS-DB-11-003 also contains the fire area assessment, the identified pinch points, and general recommendations for administrative controls to reduce that fire risk, as well as a proposed strategy for recovering the KSF should a fire occur. In accordance with FAQ 07-0040, any compartment experiencing fire damage which eliminates the success paths for a KSF (without recovery actions outside the MCR) is considered a pinch point. Fire modeling was not used to eliminate any fire compartment from being a pinch point.

3.5.2.3 NPO Key Safety Functions and SSC Used to Achieve Performance

In LAR Attachment D, the licensee stated that it evaluated the following KSFs against the selected plant operating states for inclusion in the NPO transition review: DHR, reactor coolant inventory control, electrical power availability, reactivity control, containment closure, and spent fuel pool cooling. The licensee stated that the evaluation resulted in the exclusion of the containment closure and spent fuel pool cooling from further consideration. The licensee explicitly modeled all other KSFs in the NPO analysis database. The licensee reviewed the various modes of operation for each system used to satisfy each KSF and developed a comprehensive list of equipment. The equipment was selected based on the systems identified for meeting each applicable KSF. Where applicable, the licensee reviewed the NPO functional requirement for the selected equipment against the functions previously analyzed for the at-power analysis. Cable selection was performed as necessary per applicable project procedures.

In response to SSA RAI 02.a (Reference 10), the licensee stated that "NPO components that were not previously analyzed as SSD components, or for which their NPO position is different from SSD positions, were selected for cable identification and cable analysis." These components include, but are not limited to, the following: system interface valves between the DHR system and the RCS; cooling system, supporting components, and associated instrumentation for the DHR systems; borated water sources and flow paths; and nuclear instrumentation.

In LAR Attachment D, the licensee stated:

> 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 in the analysis database. The selected components were flagged as NPO within the database to allow "pinch point" analysis by fire area.

A pinch point is 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.

Based on the review of the information provided in the LAR, the NRC staff finds that the licensee used acceptable methods, consistent with the guidance provided in RG 1.205 and FAQ 07-0040, to identify the equipment required to achieve and maintain the fuel in a safe and stable condition during NPO modes. The NRC staff concludes that the licensee has an acceptable process in place to ensure that fire protection DID measures will be implemented to achieve the KSFs during plant outages.

3.5.2.4 NPO Pinch Point Resolutions and Program Implementation

In LAR Attachment D, the licensee stated that it performed a deterministic fire separation analysis for NPO to identify pinch points. The licensee determined that 32 fire compartments have an adequate number of KSF success paths to survive the loss of the entire contents of the compartment, and 46 compartments have pinch points resulting in the potential loss of one or more KSF success paths.

In LAR Attachment D, the licensee stated that it evaluated fire compartments with identified pinch points and considered plant controls, consistent with FAQ 07-0040, to minimize fire risk. Enhancements will be developed to preclude or mitigate the KSF failures in certain fire compartments, and include planned revisions to the procedures, as necessary (Implementation Item DB-1908). These revisions will incorporate the insights and strategies documented in its analyses for the plant to reduce fire risk during HREs. In response to SSA RAI 02 (Reference 10), the licensee stated Implementation Item DB-1908 will include "the necessary steps to ensure that the KSFs are achieved and maintained as appropriate administrative procedures." Potential actions for DID strategies include pre-positioning or removing power from valves. There are no RAs credited to achieve a KSF. In LAR Attachment D, the licensee also stated that the strategies will include, but not be limited to, consideration of the generic actions in FAQ 07-0040.

NFPA 805 requires that the NSPC be met during any operational mode or condition, including NPO. As described above, the licensee has performed engineering analyses to demonstrate that it meets the NSPC for NPO. Based on the information provided in the LAR, as supplemented, the NRC staff concludes that the licensee provided reasonable assurance that the NSPC of NFPA 805 will be met during NPO modes and HREs.

3.5.3 Conclusion for Section 3.5

The NRC staff reviewed the LAR to evaluate the NSCA results. The licensee used the PB approach in accordance with NFPA 805, Section 4.2.4, for each fire compartment and the NRC staff confirmed that:

- The bases for the current exemptions to 10 CFR Part 50, Appendix R, that will transition to the new FPP continue to be valid and will meet the 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 compartment.
- All VFDRs were evaluated using FREs to address risk impact, DID, and safety margin, and found to be acceptable in accordance with NFPA 805, Section 4.2.4.2.
- 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.
- All DID RAs were properly documented for each fire compartment.
- The required automatic fire suppression and automatic fire detection systems were appropriately documented for each fire compartment.

Accordingly, the NRC staff concludes that, for each applicable fire compartment, the PB approach used by the licensee will meet the requirements of NFPA 805, Section 4.2.4. In addition, the licensee's analysis and outage management process for NPO provides reasonable assurance that the NSPC will be met during NPO modes including HREs. The overall approach for fire protection during NPO modes is acceptable because the NFPA 805 requirements for risk, DID, and safety margin will be met.

3.6 Radioactive Release Performance Criteria

As discussed in SE Section 2.0, NFPA 805, Chapter 1, specifies the radioactive release goals, objectives, and performance criteria that must be met by the FPP in the event of a fire at a nuclear power plant. The NRC staff has endorsed, with exceptions, NEI 04-02 as it provides methods acceptable to the staff for establishing an RI/PB FPP consistent with NFPA 805. Additional guidance is provided in FAQ 09-0056 (Reference 66). The licensee's review against the NFPA 805 radioactive release requirements is discussed in LAR Section 4.4, "Radioactive Release Performance Criteria," and LAR Attachment E (Reference 6), as supplemented (Reference 10).

In LAR Section 4.4, the licensee stated that it reviewed pre-fire plans, fire brigade training materials, and engineering controls being credited during all plant operating modes. The licensee screened out pre-fire plans for areas where there is no possibility of radioactive materials being present.

In LAR Attachment E, the licensee identified the engineering controls credited for containment of gaseous effluents (e.g., forced air ventilation and filtered ventilation exhaust) and liquid effluents (e.g., floor drains and sumps routed to the radioactive waste system). For each area where radioactive materials are present, the licensee determined that the engineered controls for containment of liquid and gaseous effluent were adequate. Operator actions were not credited for mitigating a potential radioactive release. The NRC staff determined that the existing engineering controls in these areas are acceptable because, prior to discharge, the gaseous effluent is contained, monitored, and filtered to remove radioactive materials and the liquid effluent is collected, processed, and monitored.

There are radioactive materials stored in other plant areas (e.g., outside areas) where there are limited or no engineered controls to contain radioactive effluents. As discussed in the response to radiological release RAI 01 (Reference 10), the licensee performed a quantitative analysis of the potential public dose from the release of contaminated gaseous and liquid effluents resulting from a fire. The analysis results bound the dose consequences for all types of low specific activity containers stored at DBNPS.

The bounding analysis was for a fire in a sea-land storage container fully loaded with radioactive waste. The licensee performed a quantitative dose assessment based on the type and maximum quantity of radionuclides that are stored in a sea-land storage container and assumed that the contents were released during a fire. The NRC staff determined that the bounding analysis was based on conservative assumptions regarding the source term, dispersion and dilution factors, and the effluent concentration limits. The NRC staff concludes that the results demonstrate that the maximum offsite dose from the gaseous and liquid effluents at the exclusion area boundary are less than the 10 CFR Part 20 dose limits for members of the public.

In LAR Attachment E, the licensee stated:

> The applicable pre-fire plans and applicable training lesson plans will be updated to include guidance for containment and monitoring of potentially contaminated fire suppression water and products of combustion. This guidance will be incorporated into plant specific area pre-fire plans and available for fire brigade use by the scheduled fire protection program implementation date.

The actions to revise the pre-fire plans and training to address the radioactive release requirements of NFPA 805 are identified as Implementation Items DB-0341, DB-0538, DB-1074, DB-1093, DB-1095, and DB-1915 in LAR Attachment S (Reference 16), Table S-2.

The NRC staff concludes that, with the transition to the new FPP, radiation releases to any unrestricted area resulting from the direct effects of fire suppression activities at DBNPS will be as low as reasonably achievable and are not expected to exceed the public dose limits in 10 CFR Part 20. This conclusion is based on (1) the licensee's use of engineered controls to contain and monitor potential releases, (2) the licensee's planned development and implementation of revised pre-fire plans and fire brigade training procedures, and (3) the results of the licensee's quantitative analysis. Therefore, the NRC staff has reasonable assurance that the licensee will comply with the requirements of NFPA 805, Sections 1.3.2, 1.4.2, and 1.5.2. The NRC staff will ensure that the actions to revise the pre-fire plans and training are completed as part of the transition to the new FPP by making this a requirement in the new FPP license condition (see SE Section 4.0).

3.7 NFPA 805 Monitoring Program

LAR Section 4.6, "Monitoring Program" (Reference 6), describes the program the licensee will implement to monitor the availability, reliability, and performance of FPP systems and features at DBNPS after transitioning to the new FPP. NFPA 805, Section 2.6, states that:

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

The focus of the NRC staff review was on critical elements related to the monitoring program, including the selection of FPP systems and features to be included in the program, the attributes of those systems and features that will be monitored, and the methods for monitoring those attributes.

In LAR Section 4.6, the licensee stated that the process to establish the monitoring program will be comprised of scoping, screening using risk criteria, risk target value determination, and monitoring implementation. The results of these steps will be documented as part of the licensee's maintenance rule program. In addition, the monitoring program failure criteria and action level targets will be documented in the licensee's periodic assessment and evaluation forms.

LAR Section 4.6 states that the scope of the program will include fire protection systems and features, NSCA equipment, SSC relied upon to meet radioactive release criteria, and fire protection programmatic elements. The SSC identified during scoping will be part of an inspection and test program and system/program health reporting, and, if not in the current program, the SSC will be added to ensure that the criteria can be met reliably. The results of monitoring activities will be analyzed in a timely manner to ensure that appropriate action is taken. The corrective action process will be used to address performance of fire protection and nuclear safety SSC that do not meet performance criteria.

Based on the information provided in the LAR, the NRC staff concludes that the licensee's NFPA 805 monitoring program development and implementation process provides reasonable assurance that DBNPS will implement an effective program for monitoring risk significant fire SSC because the process will ensure that the monitoring program:

- establishes the appropriate scope of SSC to be monitored;
- uses an acceptable screening process for determining the SSC to be included in the program;
- establishes availability, reliability, and performance criteria for the SSC being monitored; and,
- requires corrective actions when SSC availability, reliability, or performance criteria targets are exceeded to bring performance back within the required range.

The actions to establish the monitoring program required by NFPA 805, Section 2.6, are identified as Implementation Items DB-1147, DB-1744, and DB-1949 in LAR Attachment S (Reference 16), Table S-2.

The NRC staff reviewed the licensee's proposed process for establishing a monitoring program. Based on this review, the NRC staff concludes that the licensee will have a monitoring program that meets the requirements of NFPA 805, Section 2.6, and that implementation of the monitoring program on the same schedule as the overall implementation of the new FPP is acceptable. The NRC staff will ensure that the actions to establish the monitoring program are completed as part of the transition to the new FPP by making this a requirement in the new FPP license condition.

3.8 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" (Reference 6), for compliance with the NFPA 805 plant change evaluation process requirements to address potential changes to the new FPP after implementation is completed. The licensee indicated that it developed its change process based on the guidance provided in Section 5.3, "Plant Change Process," and Appendices B, I, and J of NEI 04-02, as modified by Sections C.2.2.4, C.3.1, C.3.2, and C.4.3 of RG 1.205.

NFPA 805, Section 2.4.4, requires that plant change evaluations consist of an integrated assessment of risk, DID, and safety margins. NFPA 805, Section 2.4.3.1, requires that the PRA use CDF and LERF as measures for risk. NFPA 805, Section 2.4.3.3, requires the risk assessment approach, methods, and data to be acceptable to the NRC. NFPA 805, 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.

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

1. defining the change;
2. performing the preliminary risk screening;
3. performing the risk evaluation; and
4. evaluating the acceptance criteria.

The licensee stated that the plant change evaluation process begins (step 1) by defining the change or altered condition to be examined and the baseline configuration, as defined by the design and licensing basis. A screening is then performed (step 2) to identify and resolve minor changes to the FPP. The screening process is modeled after NEI 02-03, "Guidance for Performing a Regulatory Review of Proposed Changes to the Approved Fire Protection Program," June 2003 (Reference 104).

For step 3, the licensee stated:

> The screening is followed by engineering evaluations that may include fire modeling and risk assessment techniques. The results of these evaluations are

> then compared to the acceptance criteria. Changes that satisfy the acceptance criteria of NFPA 805, Section 2.4.4 and the license condition can be implemented within the framework provided by NFPA 805. Changes that do not satisfy the acceptance criteria cannot be implemented within this framework. The acceptance criteria require that the resultant change in CDF and LERF be consistent with the license condition. The acceptance criteria also include consideration of DID and safety margin, which would typically be qualitative in nature.
>
> The risk evaluation involves the application of fire modeling analyses and risk assessment techniques to obtain a measure of the changes in risk associated with the proposed change. In certain circumstances, an initial evaluation in the development of the risk assessment could be a simplified analysis using bounding assumptions, provided the use of such assumptions does not unnecessarily challenge the acceptance criteria....

For step 4, the licensee stated that the plant change evaluations will be assessed for acceptability using the license condition criteria for the change in CDF and LERF. Specifically, the proposed license condition states that prior NRC review and approval is not required for (1) changes that clearly result in a decrease in risk and (2) individual changes that result in a risk increase less than 10^{-7}/year for CDF and less than 10^{-8}/year for LERF. These risk criteria are acceptable to the NRC staff because they are consistent with the criteria in the standard fire protection license condition in Section C.3.1 of RG 1.205, Revision 1. The proposed changes will also be "assessed to ensure they are consistent with the DID philosophy and that sufficient safety margins were maintained." The licensee further stated:

> The DBNPS Fire Protection Program configuration is defined by the program documentation. To the greatest extent possible, the existing configuration control processes for modifications, calculations, and analyses, and Fire Protection Program Licensing Basis Reviews will be utilized to maintain configuration control of the Fire Protection program documents. The configuration control procedures which govern the various DBNPS documents and databases that currently exist will be revised to reflect the new NFPA 805 licensing bases requirements.
>
> Several NFPA 805 document types such as NSCA Supporting Information, Non-Power Mode NSCA Treatment, etc., generally require new control procedures and processes to be developed since they are new documents and databases created as a result of the transition to NFPA 805. The new procedures will be modeled after the existing processes for similar types of documents and databases. System level design basis documents will be revised to reflect the NFPA 805 role that the system components now play.
>
> The process for capturing the impact of proposed changes to the plant on the Fire Protection Program will continue to be a multiple step review. The first step of the review is an initial screening for process users to determine if there is a potential to impact the Fire Protection program as defined under NFPA 805 through a series of screening questions/checklists contained in one or more procedures depending upon the configuration control process being used. Reviews that identify potential Fire Protection program impacts will be sent to qualified individuals ... to ascertain the program impacts, if any. If Fire Protection

program impacts are determined to exist as a result of the proposed change, the issue would be resolved by one of the following:

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

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

This process follows the requirements in NFPA 805 and the guidance outlined in RG 1.174, which requires the use of qualified individuals, procedures that require calculations to 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....

In LAR Attachment S (Reference 16), Table S-2, the licensee identified two implementation items related to the plant change evaluation process. Specifically, the licensee stated it will develop the DBNPS NFPA 805 design basis document and the new NFPA 805 control procedures and processes (Implementation Items DB-2049 and DB-2050, respectively).

NFPA 805 requires the use of a plant change evaluation, regardless of what element requires the change. Therefore, if FPP impacts are determined to exist as a result of the proposed change, the licensee would resolve the issue using its plant change process. This process will be used to determine if prior NRC approval of the proposed change is required.

The NRC staff finds that the licensee's plant change evaluation process will be consistent with the guidance in NEI 04-02, as modified by RG 1.205, and will meet the requirements of NFPA 805. Specifically, the licensee's plant change evaluation process will include: (1) an integrated assessment of risk, DID, and safety margins, using appropriate risk acceptance criteria; (2) the calculation of the change in risk using CDF and LERF; and (3) the use of an FPRA that is acceptable to the NRC (see SE Section 3.4). Therefore, the NRC staff concludes that the licensee's plant change evaluation process is acceptable. The NRC staff will ensure that the actions related to implementation of the plant change evaluation process are completed as part of the transition to the new FPP by making this a requirement in the new FPP license condition.

3.9 Program Documentation, Configuration Control, and Quality Assurance

NFPA 805, Section 2.7, "Program Documentation, Configuration Control, and Quality," specifies the content, configuration control, and quality requirements for the documentation used to support an FPP based on NFPA 805. In LAR Section 4.7 (Reference 6) the licensee described how it will meet these requirements.

3.9.1 Documentation

NFPA 805, Section 2.7.1, "Content," specifies the documentation requirements for an FPP based on NFPA 805. LAR Section 4.7.1, "Compliance with Documentation Requirements in Section 2.7.1 of NFPA 805," states:

> In accordance with the requirements and guidance in NFPA 805, Section 2.7.1 and NEI 04-02, DBNPS has documented analyses to support compliance with 10 CFR 50.48(c). The analyses are being performed in accordance with FENOC's processes for ensuring assumptions are clearly defined, that results are easily understood, that results are clearly and consistently described, and that sufficient detail is provided to allow future review of the entire analyses.
>
> Analyses, as defined by NFPA 805, Section 2.4, performed to demonstrate compliance with 10 CFR 50.48(c) will be maintained for the life of the plant and organized to facilitate review for accuracy and adequacy....
>
> The Fire Protection Design Basis Document described in Section 2.7.1.2 of NFPA 805 and necessary supporting documentation described in Section 2.7.1.3 of NFPA 805 will be created as part of the transition to 10 CFR 50.48(c) to ensure program implementation following receipt of the safety evaluation....

Based on the statements in LAR Section 4.7.1, the NRC staff concludes that the licensee will comply with the documentation requirements of NFPA 805, Section 2.7.1.

3.9.2 Configuration Control

NFPA 805, Section 2.7.2, "Configuration Control," specifies the requirements to maintain the FPP documentation up-to-date. LAR Section 4.7.2 states, in part, that:

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

LAR Section 4.7.2 further states:

> The DBNPS Fire Protection Program configuration is defined by the program documentation. To the greatest extent possible, the existing configuration control processes for modifications, calculations, and analyses, and Fire Protection Program Licensing Basis Reviews will be utilized to maintain configuration control of the Fire Protection program documents. The configuration control procedures which govern the various DBNPS documents and databases that currently exist will be revised to reflect the new NFPA 805 licensing bases requirements.

LAR Attachment S (Reference 16), Table S-2, includes an action to develop the new NFPA 805 configuration control procedures and processes (Implementation Item DB-2050).

LAR Section 4.7.3 states, in part, that:

> Configuration control of the Fire PRA model will be maintained by integrating the Fire PRA model into the existing processes used to ensure configuration control of the internal events PRA model. This process complies with Section 1-5 of the ASME PRA Standard and ensures that FENOC maintains an as-built, as-operated PRA model of the plant.

SE Sections 3.4.3.4 and 3.9.3 describe the NRC staff's review of the licensee's process for updating and maintaining the DBNPS FPRA to reflect plant changes made after completion of the transition to NFPA 805.

Based on the statements in LAR Sections 4.7.2 and 4.7.3, the NRC staff concludes that the licensee will comply with the configuration control requirements of NFPA 805, Section 2.7.2. The NRC staff will ensure that the actions related to implementation of the configuration control process are completed as part of the transition to the new FPP by making this a requirement in the new FPP license condition.

3.9.3 Quality

NFPA 805, Section 2.7.3, "Quality," specifies the quality assurance requirements for analyses, calculations, and evaluations used for an FPP based on NFPA 805. These requirements 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. In LAR Section 4.7.3, the licensee stated that it will maintain the existing fire protection quality assurance program at DBNPS. The licensee stated that it performed work in support of the transition to the new FPP in accordance with NFPA 805, Section 2.7.3, and that it will continue to comply with NFPA 805, Section 2.7.3, post-transition. The NRC staff's review of the LAR against each of the NFPA 805, Section 2.7.3, requirements is discussed below.

3.9.3.1 Independent Review

NFPA 805, Section 2.7.3.1, requires that each analysis, calculation, or evaluation performed be independently reviewed. In LAR Section 4.7.3, the licensee stated: "Analyses, calculations, and evaluations performed in support of compliance with 10 CFR 50.48(c) are performed in accordance with FENOC procedures that require independent review." Based on this statement, the NRC staff concludes that the licensee will comply with NFPA 805, Section 2.7.3.1.

3.9.3.2 Verification and Validation

NFPA 805, Section 2.7.3.2, requires each calculational model or numerical method to be verified and validated through comparison to test results or other acceptable models. In LAR Section 4.7.3, the licensee stated that it verified and validated the calculational models and numerical methods used in support of the transition to NFPA 805, and that it will continue to comply with NFPA 805, Section 2.7.3.2, post-transition.

NUREG-1824 (Reference 41) documents the V&V of five selected fire models commonly used to support applications of RI/PB FPPs at nuclear power plants. This NUREG provides technical documentation concerning the predictive capabilities of a specific set of fire dynamics

calculation tools and fire phenomenological models that may be used for the analysis of fire hazards in postulated nuclear power plant scenarios. When used within the limitations of the fire models, and considering the identified uncertainties, these models may be employed to demonstrate compliance with the requirements of 10 CFR 50.48(c).

In LAR Attachment J (Reference 6), Table J-1, the licensee identified the V&V basis for the fire models and model correlations used in its FPRA (also see SE Section 3.4.3.3). For each correlation used in the FM at DBNPS, Table J-1 describes how the correlation was applied and provides the V&V basis and references for the correlation. The licensee also provided justification when a correlation was used outside the validated range. LAR Attachment J, Table J-4, "Fire Modeling Application," indicates that the licensee followed the guidance in NUREG-1934 (Reference 47) to demonstrate that the correlations and fire models were applied within their validated input parameter range, or that their use outside the limits of the validated range was justified.

The NRC staff reviewed the V&V basis for each of the correlations in LAR Table J-1. For some FM elements, the licensee used the fire models in NUREG-1824 consistent with the limitations, which the NRC staff finds acceptable. However, the licensee also identified that it used several empirical correlations that are not addressed in NUREG-1824 and some that were applied outside their validated range. The NRC staff determined 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, peer-reviewed journal articles, peer-reviewed conference papers, and national research laboratory reports. For correlations that were applied outside their validated range, the NRC found that the licensee provided acceptable justification. The FM employed by the licensee in the development of the DBNPS FREs used empirical correlations that provide bounding solutions for the ZOI and conservative input parameters, which produced conservative results for the FM analysis.

Based on the statements in the LAR, the NRC staff concludes that the licensee's V&V basis for the fire models and model correlations used in its FPRA provides reasonable assurance that the licensee's FM is appropriate and acceptable for use in the licensee's transition to NFPA 805. Therefore, the NRC staff concludes that the licensee's approach for meeting the requirements of NFPA 805, Section 2.7.3.2, is acceptable.

3.9.3.3 Limitations of Use

NFPA 805, Section 2.7.3.3, states that:

> Acceptable engineering methods and numerical models shall only be used for applications to the extent these methods have been subject to verification and validation. These engineering methods shall only be applied within the scope, limitations, and assumptions prescribed for that method.

In LAR Section 4.7.3, the licensee stated that it appropriately applied the engineering methods and numerical models used in support of the transition to the new FPP, and that it will continue to comply with NFPA 805, Section 2.7.3.3, post-transition. The limitations of the fire models used at DBNPS are discussed in LAR Attachment J, Table J-7, "Fire Modeling Limitations."

The NRC staff assessed the acceptability of the empirical correlations and fire models used by the licensee in terms of their limitations of use. As discussed in SE Section 3.9.3.2, the NRC staff found that the licensee provided acceptable justification for correlations that were applied

outside their validated range. Based on the licensee's statements that it used the fire models to support development of the FREs within their limitations, and the description of the DBNPS 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.

3.9.3.4 Qualification of Users

NFPA 805, Section 2.7.3.4, requires that personnel performing engineering analyses and applying numerical methods be competent in that field and experienced in the application of these methods as they relate to nuclear power plants, nuclear power plant fire protection, and power plant operations.

In LAR Section 4.7.3, the licensee stated that cognizant personnel who use and apply engineering analysis and numerical methods in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by NFPA 805, Section 2.7.3.4, and that post-transition, will perform work in accordance with the requirements of NFPA 805, Section 2.7.3. The qualifications of the personnel performing the FM at DBNPS are discussed in LAR Attachment J, Table J-5, "Qualifications of Users in Fire Modeling." The licensee stated that it will develop and maintain qualification requirements for personnel performing FM post-transition. The development of FM qualification guides and procedures is included in LAR Attachment S (Reference 16), Table S-2, as Implementation Item DB-2037.

Based on the statements in LAR Section 4.7.3 and LAR Attachment J, the NRC staff has reasonable assurance that the licensee will comply with the requirements of NFPA 805, Section 2.7.3.4. The NRC staff will ensure that the actions related to user qualification for FM are completed as part of the transition to the new FPP by making this a requirement in the new FPP license condition (see SE Section 4.0).

3.9.3.5 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. This is modified by 10 CFR 50.48(c)(2)(iv), which states that this uncertainty analysis is not required to support calculations used in conjunction with a deterministic approach.

According to NUREG-1855, Volume 1 (Reference 45), 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 (Reference 47).

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.

In LAR Section 4.7.3, the licensee stated that uncertainty analyses were performed as required by NFPA 805, Section 2.7.3.5, and the results were considered in the context of the application. The licensee also stated that post transition it will perform work in accordance with the requirements of NFPA 805, Section 2.7.3. The uncertainties in the FM performed at DBNPS are discussed in LAR Attachment J, Table J-6, "Uncertainty in Fire Modeling." The NRC staff reviewed LAR Table J-6 and found that the licensee adequately addressed the three types of uncertainty described in NUREG-1855, Volume 1.

The NRC staff's evaluation of the licensee's treatment of uncertainties in the FPRA is discussed in SE Section 3.4.7. Based on the information in the LAR, the NRC staff finds that the licensee performed an appropriate uncertainty analysis for the application and has reasonable assurance that the licensee will comply with NFPA 805, Section 2.7.3.5.

4.0 FIRE PROTECTION LICENSE CONDITION

As discussed in SE Section 2.4.2, the licensee proposed a new FPP license condition (LAR Attachment M) to replace its current license condition 2.C(4) for fire protection. The proposed license condition is consistent with the standard fire protection license condition in Section C.3.1 of RG 1.205, Revision 1, with some plant-specific changes.

Overall, the proposed license condition provides structure and detailed criteria to allow the licensee to make changes to the DBNPS FPP, without prior NRC approval, if the requirements of NFPA 805 regarding engineering analyses, FREs, and plant change evaluations are met. The proposed license condition also defines limitations on self-approval 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 necessary because NFPA 805 requires that the risk analyses be based on the as-built, as-operated and maintained plant, and reflect the operating experience at the plant. Until the proposed implementation items and plant modification are completed, the risk analysis will not be consistent with the as-built, as-operated and maintained plant. The NRC staff's evaluation of the self-approval process for FPP changes (post-transition) is contained in SE Section 2.6.

The proposed license condition also references the plant-specific modifications that must be completed for DBNPS to complete the transition to NFPA 805 and comply with 10 CFR 50.48(c). In addition, the proposed license condition includes a requirement that appropriate compensatory measures remain in place until implementation of the specified plant modifications is completed. However, as discussed in SE Section 2.7, the licensee also credited the completion of the implementation items in LAR Attachment S (Reference 16), Table S-2, in its analysis supporting the LAR. These implementation items must also be completed for DBNPS to fully transition to NFPA 805. Therefore, the NRC staff will require the completion of these implementation items, in addition to the remaining plant modification, as part of the NFPA 805 license condition.

In addition, the NRC staff has decided to make some editorial changes to improve readability and consistency with the DBNPS license. These changes include spelling out some undefined acronyms, using scientific notation instead of calculator notation, and other editorial changes. In addition, the amendment approving the LAR is referenced instead of the SE.

The following license condition will replace DBNPS license condition 2.C(4) in its entirety. By letter dated May 13, 2019 (Reference 17), the licensee confirmed that this license condition is acceptable.

Fire Protection

FENOC 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 December 16, 2015, as supplemented by letters dated February 2, March 7, July 28, and December 16, 2016; January 17, June 16, and October 9, 2017; April 2, September 11, and November 20, 2018; and May 13, 2019, and as approved by Amendment No. 298. 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 probabilistic risk assessment model, methods that have been approved by the 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 10^{-7}/year for core damage frequency and less than 10^{-8}/year for large early release frequency. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

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

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

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

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 by Amendment No. 298, 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) and (3) 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 modification described in Attachment S, Table S-1, "Plant Modifications Committed," to the FENOC letter dated November 20, 2018, within 2 years following issuance of the license amendment. The licensee shall maintain appropriate compensatory measures in place until completion of this modification.

(3) The licensee shall implement the items listed in Attachment S, Table S-2, "Implementation Items," to the FENOC letter dated November 20, 2018, within 2 years following issuance of the license amendment.

5.0 SUMMARY

The NRC staff reviewed the LAR to transition DBNPS to an RI/PB FPP against the requirements in 10 CFR 50.48(c). The licensee identified that no orders need to be revised. As discussed in SE Section 2.4.3, the licensee identified the need to revise TS 5.4.1 to support the transition, and the NRC staff found the proposed change to TS 5.4.1 acceptable. As discussed in SE Section 2.4.2, the licensee identified the need to revise the FPP license condition 2.C(4) and provided a proposed new license condition. As discussed in SE Section 4.0, the NRC staff will impose a new FPP license condition to replace license condition 2.C(4) in its entirety. Therefore, in accordance with 10 CFR 50.48(c)(3)(i), the NRC staff concludes that the licensee has identified the orders, license conditions, and TSs that must be revised or superseded, and that the proposed revisions to the TSs and license conditions, as modified by the NRC staff, are adequate.

The NRC staff reviewed the LAR against the NFPA 805 requirements, as modified by 10 CFR 50.48(c). The NRC staff concludes that the licensee's approach, methods, and data are acceptable to establish, implement, and maintain the proposed RI/PB FPP in accordance with 10 CFR 50.48(c), subject to completion of the modification in Table S-1 and implementation items in Table S-2 of LAR Attachment S (Reference 16). As discussed in SE Section 4.0, the NRC staff will impose a license condition with the issuance of the amendment to ensure that the modification in LAR Table S-1 and implementation items in LAR Table S-2 are completed in accordance with a schedule that is acceptable to the NRC staff.

In LAR Attachment L (Reference 13), as supplemented, the licensee requested NRC staff review and approval of PB methods to demonstrate an equivalent level of fire protection for specific NFPA 805, Chapter 3, elements. As discussed in SE Section 3.1.4, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concluded that the proposed PB methods in Approval Requests 1–7 and 9–14 are acceptable alternatives to the corresponding NFPA 805, Chapter 3, requirements.

The license condition discussed in SE Section 4.0, in conjunction with NFPA 805, will permit the licensee to make certain changes to the DBNPS FPP without prior NRC approval. The license condition provides a structure and detailed criteria to allow the licensee to self-approve changes

to the DBNPS FPP if the requirements of NFPA 805 regarding engineering analyses, FREs, and plant change evaluations are met. The NRC staff concluded that the licensee's FPRA is adequate and will be appropriately maintained to support the self-approval of RI changes to the DBNPS FPP. The license condition, in conjunction with NFPA 805, will ensure that the licensee will continue to comply with 10 CFR 50.48(c), while permitting the licensee to make certain changes to the DBNPS FPP without prior NRC approval.

Overall, the NRC staff concludes that the proposed RI/PB FPP is acceptable and that the licensee has demonstrated that the new FPP at DBNPS will meet the requirements of GDC 3, 10 CFR 50.48(a), and 10 CFR 50.48(c).

As discussed in SE Section 2.5, the NRC staff determined that the exemptions to 10 CFR Part 50, Appendix R, for DBNPS will no longer be applicable if the proposed amendment to transition to NFPA 805 is approved. Therefore, the NRC staff finds it acceptable to rescind the current exemptions to 10 CFR Part 50, Appendix R, identified by the licensee with the approval of the proposed license amendment.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified on March 19, 2019, 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 April 12, 2016 (81 FR 21599). 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 U.S. Nuclear Regulatory Commission, Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection For Nuclear Power Plants," May 1, 1976 (ADAMS Accession No. ML070660461).

2 U.S. Nuclear Regulatory Commission, Appendix A to BTP APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976," February 24, 1977 (ADAMS Accession No. ML070660458).

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

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

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

6 Boles, Brian D., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Application for License Amendment to Adopt NFPA Standard 805, 'Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)'," December 16, 2015 (ADAMS Accession No. ML15350A314).

7 Boles, Brian D., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Distribution Notice for License Amendment Request to Adopt NFPA Standard 805, 'Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)'," February 2, 2016 (ADAMS Accession No. ML16033A085).

8 Byrd, Kendall W., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Supplemental Information for License Amendment Request to Adopt National Fire Protection Association (NFPA) Standard 805 (CAC No. MF7190)," March 7, 2016 (ADAMS Accession No. ML16067A195).

9 Boles, Brian D., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Correction of Fire Probabilistic Risk Analysis (PRA) Results (CAC No. MF7190)," July 28, 2016 (ADAMS Accession No. ML16210A422).

10 Boles, Brian D., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Association (NFPA) Standard 805 (CAC No. MF7190)," December 16, 2016 (ADAMS Accession No. ML16351A330).

11 Boles, Brian D., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Association (NFPA) Standard 805 (CAC No. MF7190)," January 17, 2017 (ADAMS Accession No. ML17017A504).

12 Boles, Brian D., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Association (NFPA)

Standard 805 (CAC No. MF7190)," June 16, 2017 (ADAMS Accession No. ML17170A000).

13 Bezilla, Mark B., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Supplemental Information Regarding License Amendment Request to Adopt National Fire Protection Association (NFPA) Standard 805 (CAC No. MF7190)," October 9, 2017 (ADAMS Accession No. ML17284A190).

14 Bezilla, Mark B., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information and Supplemental Information Regarding License Amendment Request to Adopt National Fire Protection Association (NFPA) Standard 805 (CAC No. MF7190)," April 2, 2018 (ADAMS Accession No. ML18094A798).

15 Bezilla, Mark B., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information and Supplemental Information Regarding License Amendment Request to Adopt National Fire Protection Association (NFPA) Standard 805 (CAC No. MF7190, EPID L-2015-LLF-0001)," September 11, 2018 (ADAMS Accession No. ML18254A073).

16 Huey, Douglas B., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Supplemental Information Regarding License Amendment Request to Adopt National Fire Protection Association (NFPA) Standard 805 (CAC No. MF7190, EPID L-2015-LLF-0001)," November 20, 2018 (ADAMS Accession No. ML18324A677).

17 Bezilla, Mark B., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Supplemental Information Regarding License Amendment Request to Adopt National Fire Protection Association (NFPA) Standard 805 (CAC No. MF7190, EPID L-2015-LLF-0001)," May 13, 2019 (ADAMS Accession No. ML19134A032.

18 Purnell, Blake A., U.S. Nuclear Regulatory Commission, letter to Boles, Brian D., FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station, Unit No. 1 - Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Associated Standard 805 (CAC No. MF7190)," October 18, 2016 (ADAMS Accession No. ML16256A066).

19 Purnell, Blake A., U.S. Nuclear Regulatory Commission, letter to Boles, Brian D., FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station, Unit No. 1 - Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Associated Standard 805 (CAC No. MF7190)," April 19, 2017 (ADAMS Accession No. ML17100A173).

20 Purnell, Blake A., U.S. Nuclear Regulatory Commission, E-Mail to Lashley, Phil H., FirstEnergy Nuclear Operating Company, "Davis-Besse - Request for Additional Information Regarding License Amendment Request to Adopt NFPA 805 (CAC No. MF7190, EPID L-2015-LLF-0001)," July 19, 2018 (ADAMS Accession No. ML18201A412).

21 Reid, Robert W., U.S. Nuclear Regulatory Commission, letter to Roe, Lowell, E., Toledo Edison Company, "Issuance of Amendment No. 18 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1," July 26, 1979 (ADAMS Accession No. ML021160382).

22 Hannon, John N., U.S. Nuclear Regulatory Commission, letter to Shelton, Donald C., Toledo Edison Company, "Safety Evaluation of Fire Protection Measures at the Davis-Besse Nuclear Power Station, Unit No. 1, Per Appendix R to 10 CFR Part 50 (TAC

Nos. M60994, M60995, M61745 and M61923)," May 30, 1991 (ADAMS Accession No. ML033490026).

23 Nuclear Energy Institute, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," NEI 00-01, Revision 2, May 2009 (ADAMS Accession No. ML091770265).

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

25 U.S. Nuclear Regulatory Commission, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," RG 1.200, Revision 2, March 2009 (ADAMS Accession No. ML090410014).

26 American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS), "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME/ANS RA-Sa-2009, Februrary 2, 2009.

27 Nuclear Energy Institute, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," NEI 05-04, Revision 2, November 2008 (ADAMS Accession No. ML083430462).

28 Nuclear Energy Institute, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," NEI 07-12, Revision 1, June 2010 (ADAMS Accession No. ML102230070).

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

30 U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection Program," Revision 0, December 2009 (ADAMS Accession No. ML092590527).

31 U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load," Revision 3, September 2012 (ADAMS Accession No. ML12193A107).

32 U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (ADAMS Accession No. ML071700658).

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

34 U.S. Nuclear Regulatory Commission, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology," NUREG/CR-6850, September 2005 (ADAMS Accession No. ML052580118).

35 U.S. Nuclear Regulatory Commission, "Fire Probabilistic Risk Assessment Methods Enhancements," NUREG/CR-6850, Supplement 1, September 2010 (ADAMS Accession No. ML103090242).

36 Correia, Richard P., memorandum to Giitter, Joseph G., U.S. Nuclear Regulatory Commission, "Interim Technical Guidance on Fire-Induced Circuit Failure Mode Likelihood Analysis," June 14, 2013 (ADAMS Package Accession No. ML13165A194).

37 U.S. Nuclear Regulatory Commission, "Cable Response to Live Fire (CAROLFIRE)," NUREG/CR-6931, Volumes 1, 2, and 3, April 2008 (ADAMS Accession Nos. ML081190230, ML081190248, and ML081190261).

38 U.S. Nuclear Regulatory Commission, "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire): Test Results," NUREG/CR-7100, April 2012 (ADAMS Package Accession No. ML121600316).

39 U.S. Nuclear Regulatory Commission, "Good Practices for Implementing Human Reliability Analysis (HRA)," NUREG-1792, April 2005 (ADAMS Accession No. ML051160213).

40 U.S. Nuclear Regulatory Commission, "Fire Dynamics Tools (FDTs): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," NUREG-1805, December 2004 (ADAMS Accession No. ML043290075).

41 U.S. Nuclear Regulatory Commission, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," NUREG-1824, May 2007. Volume 1: Main Report, Volume 2: Experimental Uncertainty, Volume 3: Fire Dynamics Tools (FDTs), Volume 4: Fire-Induced Vulnerability Evaluation (FIVE-Rev1), Volume 5: Consolidated Fire Growth and Smoke Transport Model (CFAST), Volume 6: MAGIC, and Volume 7: Fire Dynamics Simulator (ADAMS Accession Nos. ML071650546, ML071730305, ML071730493, ML071730499, ML071730527, ML071730504, ML071730543, respectively).

42 U.S. Nuclear Regulatory Commission, "Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE), Phase 1: Horizontal Trays," NUREG/CR-7010, Volume 1, July 2012 (ADAMS Accession No. ML12213A056).

43 U.S. Nuclear Regulatory Commission, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 1: Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," NUREG/CR-7150 (also designated as EPRI 1026424 and BNL-NUREG-98204-2012), October 2012 (ADAMS Accession No. ML12313A105).

44 U.S. Nuclear Regulatory Commission, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," NUREG/CR-7150 (also designated as EPRI 3002001989 and BNL-NUREG-98204-2012), May 2014 (ADAMS Accession No. ML14141A129).

45 U.S. Nuclear Regulatory Commission, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," NUREG-1855, Volume 1, March 2009 (ADAMS Accession No. ML090970525).

46 U.S. Nuclear Regulatory Commission, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," NUREG-1921, July 2012 (ADAMS Accession No. ML12216A104).

47 U.S. Nuclear Regulatory Commission, "Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG)," NUREG-1934, November 2012 (ADAMS Accession No. ML12314A165).

48 U.S. Nuclear Regulatory Commission, "Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE -FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume," NUREG-2178, Volume 1, April 2016, (ADAMS Accession No. ML16110A140).

49 U.S. Nuclear Regulatory Commission, Generic Letter 2006-03, "Potentially Nonconforming HEMYC and MT Fire Barrier Configurations," April 10, 2006 (ADAMS Accession No. ML053620142).

50 National Fire Protection Association, "Standard for the Installation of Sprinkler Systems," Standard 13 (NFPA 13), Quincy, Massachusetts.

51 National Fire Protection Association, "Standard for the Installation of Standpipe, Private Hydrant, and Hose Systems," Standard 14 (NFPA 14), Quincy, Massachusetts.

52 National Fire Protection Association, "Flammable and Combustible Liquids Code," Standard 30 (NFPA 30), Quincy, Massachusetts.

53 National Fire Protection Association, "Standard for Gaseous Hydrogen Systems at Consumer Sites," Standard 50A (NFPA 50A), Quincy, Massachusetts.

54 National Fire Protection Association, "Compressed Gases and Cryogenic Fluids Code," Standard 55 (NFPA 55), Quincy, Massachusetts.

55 National Fire Protection Association, "Liquid Petroleum Gas Code," Standard 58-2004 (NFPA 58-2004), Quincy, Massachusetts.

56 National Fire Protection Association, "National Fire Alarm Code," Standard 72 (NFPA 72), Quincy, Massachusetts.

57 National Fire Protection Association, "Life Safety Code," Standard 101 (NFPA 101), Quincy, Massachusetts.

58 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of National Fire Protection Association (NFPA) Standard 805 Frequently Asked Question 06-0008, Fire Protection Engineering Analyses," March 12, 2009 (ADAMS Accession No. ML073380976).

59 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of National Fire Protection Association Frequently Asked [Question] 07-0030 on Establishing Recovery Actions," February 4, 2011 (ADAMS Accession No. ML110070485).

60 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of National Fire Protection Association Frequently Asked [Question] 07-0038 on Lessons Learned on Multiple Spurious Operations," February 3, 2011 (ADAMS Accession No. ML110140242).

61 Nuclear Energy Institute, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," NEI 00-01, Revision 1, January 2005 (ADAMS Accession No. ML050310295).

62 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of National Fire Protection Association Standard 805 Frequently Asked Question 07-0039 Incorporation of Pilot Plant Lessons Learned - Table B-2," January 15, 2010 (ADAMS Accession No. ML091320068).

63 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of National Fire Protection Association 805 Frequently Asked Question 07-0040 on Non-Power Operations Clarifications," August 11, 2008 (ADAMS Accession No. ML082200528).

64 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of National Fire Protection Association Standard 805 Frequently Asked Question 08-0053, 'Kerite-FR Cable Failure Thresholds,' Revision 1," June 6, 2012 (ADAMS Accession No. ML121440155).

65 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of National Fire Protection Association Frequently Asked Question 08-0054 on Demonstrating Compliance with Chapter 4 of National Fire Protection Association 805, Revision 1," March 10, 2015 (ADAMS Accession No. ML15016A280).

66 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of National Fire Protection Association 805 Frequently Asked Question 09-0056 on

Radioactive Release Transition," January 14, 2011 (ADAMS Accession No. ML102920405).

67 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of National Fire Protection Association Standard 805 Frequently Asked Question 10-0059: National Fire Protection 805 Monitoring Program," March 19, 2012 (ADAMS Accession No. ML120750108).

68 Hamzehee, Hossein G., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of Fire Probabilistic Risk Assessment Frequently Asked Question 13-0004 on Clarifications Regarding Treatment of Sensitive Electronics," December 3, 2013 (ADAMS Accession No. ML13322A085).

69 Hamzehee, Hossein G., U.S. Nuclear Regulatory Commission, Memorandum to file, "Close-Out of Fire Probabilistic Risk Assessment Frequently Asked Question 13-0005 on Cable Fires Special Cases: Self-Ignited and Caused by Welding and Cutting," December 3, 2013 (ADAMS Accession No. ML13319B181).

70 Hamzehee, Hossein G., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of Fire Probabalistic Risk Assessment Frequently Asked Question 13-0006 on Modeling Junction Box Scenarios in a Fire PRA," December 12, 2013 (ADAMS Accession No. ML13331B213).

71 Hamzehee, Hossein G., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-Out of Fire Probabalistic Risk Assessment Frequently Asked Question 14-0009 on Treatment of Well-Sealed MCC Electrical Panels Greater Than 440V," April 29, 2015 (ADAMS Accession No. ML15114A441).

72 Eisenhut, Darrell G., U.S. Nuclear Regulatory Commission, letter to Crouse, Richard P., Toledo Edison Company, "Appendix R to 10 CFR 50 - Exemption from Certain Technical Requirements," November 23, 1982 (ADAMS Accession No. ML021160466).

73 DeAgazio, Albert, U.S. Nuclear Regulatory Commission, letter to Crouse, Richard P., Toledo Edison Company, "Exemption From Certain Requirements of Appendix R to 10 CFR 50," August 20, 1984 (ADAMS Accession No. ML021190037).

74 Wambach, Thomas V., U.S. Nuclear Regulatory Commission, letter to Shelton, Donald C., Toledo Edison Company, "Exemption to 10 CFR Part 50, Appendix R, Sections III.G & III.J (TAC No. 60995)," April 18, 1990 (ADAMS Accession No. ML021190569).

75 Hansen, Allen G., U.S. Nuclear Regulatory Commission, letter to Wood, John K., Toledo Edison Company, "Issuance of Exemption from the Requirements of 10 CFR Part 50, Appendix R, Section III.O, Regarding Oil Collection Systems for Reactor Coolant Pumps - Davis-Besse Nuclear Power Station, Unit 1 (TAC No. MA0161)," January 30, 1998 (ADAMS Accession No. ML021260237).

76 Hopkins, Jon B., U.S. Nuclear Regulatory Commission, letter to Myers, Lew W., FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station, Unit 1, Exemption from the Requirements of 10 CFR Part 50, Section III.G of Appendix R (TAC No. MB1078)," December 26, 2002 (ADAMS Accession No. ML020100366).

77 Macon, William A., Jr., U.S. Nuclear Regulatory Commission, letter to Bezilla, Mark B., FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear power Station, Unit 1 - Exemption from the Requirements of 10 CFR Part 50, Appendix R, Section III.G.3 (TAC No. MC1833)," July 21, 2005 (ADAMS Accession No. ML050970136).

78 Hopkins, Jon B., U.S. Nuclear Regulatory Commission, letter to Bezilla, Mark B., FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station, Unit 1, Withdrawal of Exemption from the Requirements of 10 CFR Part 50, Appendix R,

Subsection III.L.1 (TAC No. MC1632)," June 24, 2004 (ADAMS Accession No. ML041350327).

79 Electric Power Research Institute Technical Report TR 1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide," Palo Alto, CA, Final Report, July 2003.

80 Reid, R. W., U.S. Nuclear Regulatory Commission, letter to Crouse, R. P., Toledo Edison Company, July 17, 1980 (ADAMS Legacy Library Accession No. 8007280109).

81 Weerakkody, Sunil D., U.S. Nuclear Regulatory Commission, Memorandum to AFPB File, "Closure of National Fire Protection Association 805 Frequently Asked Question Number 06-0002," January 4, 2007 (ADAMS Accession No. ML070030276).

82 Shelton, D. C., Toledo Edison Company, letter to U.S. Nuclear Regulatory Commission, "Fire Protection - National Fire Protection Association (NFPA) Code Review (TAC Numbers 60994, 60995, and 61745)," July 31, 1989 (ADAMS Legacy Library Accession No. 8908070399).

83 Shelton, D.C., Toledo Edison Company, letter to U.S. Nuclear Regulatory Commission, "Fire Protection; Additional Information on the NFPA Standard Compliance Review," October 11, 1989 (ADAMS Legacy Library Accession No. 8910190169).

84 Pace, Danny L., FirstEnergy Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Generic Letter 2006-03," June 8, 2006 (ADAMS Accession No. ML061710429).

85 Institute of Electrical and Electronics Engineers, "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations," IEEE 383, New York, New York.

86 National Fire Protection Association, "Standard for the Installation of Centrifugal Fire Pumps," Quincy, Massachusetts, Standard 20 (NFPA 20),.

87 National Fire Protection Association, "Standard for Outside Protection," Standard 24 (NFPA 24 - 1970), Quincy, Massachusetts.

88 American Society of Mechanical Engineers and American Nuclear Society (ASME/ANS), "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum B to ASME/ANS RA-S-2002," New York, New York, ASME RA-Sb-2005, December 30, 2005.

89 U.S. Nuclear Regulatory Commission, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Revision 1, January 2007 (ADAMS Accession No. ML070240001).

90 U.S. Nuclear Regulatory Commission, "Interim Reliability Evaluation Program Procedures Guide," NUREG/CR-2728, January 1983.

91 U.S. Nuclear Regulatory Commission, "2015 Industry Average Parameter Estimates Update," NUREG/CR-6928, https://nrcoe.inl.gov/resultsdb/AvgPerf/.

92 U.S. Nuclear Regulatory Commission, "A Preliminary Report on Fire Protection Research Program Fire Barriers and Fire Retardant Coating Tests," NUREG/CR-0381, September 1978 (ADAMS Accession No. ML071690019).

93 Giitter, Joseph, U.S. Nuclear Regulatory Commission, letter to Bradley, Biff, Nuclear Energy Institute, "Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, 'Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires'," June 21, 2012 (ADAMS Package Accession No. ML12172A406).

94 Shelton, D. C., Centerior Energy for the Toledo Edison Company, "Individual Plant Examination (IPE) for Severe Accident Vulnerabilities for the Davis-Besse Nuclear Power Station, Unit 1 (Response to NRC Generic Letter 88-20)," February 26, 1993 (ADAMS Accession No. ML073600839).

95 Heskestad, G., "Fire Plumes, Flame Height, and Air Entrainment," in *The SFPE Handbook of Fire Protection Engineering*, 4th ed. Quincy, Massachusetts: National Fire Protection Association, 2008, ch. 2-1.

96 Beyler, C., "Fire Hazard Calculations for Large, Open Hydrocarbon Fires," in *SFPE Handbook of Fire Protection Engineering*, 4th ed. Quincy, Massachusetts, 2008, ch. 3-10.

97 Alpert, R. L., "Ceiling Jet Flows," in *The SFPE Handbook of Fire Protection Engineering*. Quincy, Massachusetts: National Fire Protection Association, 2008, ch. 2-2.

98 Electric Power Research Institute, "Fire Induced Vulnerability Evaluation (FIVE)," Revision 1, May 1992.

99 Walton W., and Thomas, P., "Estimating Temperatures in Compartment Fires," in *The SFPE Handbook of Fire Protection Engineering*, 4th ed. Quincy, Massachusetts: National Fire Protection Association, 2008, ch. 3-6.

100 U.S. Nuclear Regulatory Commission, Inspection Manual Chapter (IMC) 0609, Appendix F, "Fire Protection Significance Determination Process," Washington, DC, September 20, 2013.

101 Peacock, R., Reneke, P., Forney, G., "CFAST - Consolidated Model of Fire Growth and Smoke Transport (Version 6) Users Guide," National Institute of Standards and Technology, Gaithersburg, MD, Special Publication 1041r1, 2012.

102 McGrattan, K., McDermott, R., Hostikka, S., Floyd, J., "Fire Dynamics Simulator Users Guide, Version 5," National Institute of Standards and Technology, Gaithersgurg, MD, Special Publication 1019-5, 2010.

103 U.S. Nuclear Regulatory Commission, "Physical Independence of Electric Systems," RG 1.75, Revision 1, January 1975 (ADAMS Accession No. ML13350A340).

104 Marion, Alexander, Nuclear Energy Institute, letter to Hannon, John, U.S. Nuclear Regulatory Commission, "Guidance for Performing a Regulatory Review of Proposed Changes to the Approved Fire Protection Program," NEI 02-03, Revision 0, June 17, 2003 (ADAMS Accession No. ML031780500).

Principal Contributors:

NRC Office of Nuclear Reactor Regulation –
Jay Robinson, JS Hyslop, Brian Metzger, Naeem Iqbal, Charles Moulton, Todd Hillsmeier

Pacific Northwest National Laboratories –
Fleurdeliza De Peralta, Garill Coles

Center for Nuclear Waste Regulatory Analyses –
Marc Janssens

Attachment: Abbreviations and Acronyms

Date of issuance: June 21, 2019

ATTACHMENT

Abbreviations and Acronyms

ADAMS	Agencywide Documents Access and Management System
ANS	American Nuclear Society
APCSB	Auxiliary and Power Conversion Systems Branch
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BTP	Branch Technical Position
BWST	borated water storage tank
CC	capability category
CCDP	conditional core damage probability
CDF	core damage frequency
CFR	*Code of Federal Regulations*
CLERP	conditional large early release probability
CST	condensate storage tank
DBNPS	Davis-Besse Nuclear Power Station, Unit No. 1
dc	direct current
DHR	decay heat removal
DID	defense-in-depth
EDG	emergency diesel generator
EEEE	existing engineering equivalency evaluation
EFWF	emergency feedwater facility
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
ERFBS	electrical raceway fire barrier systems
ERO	Emergency Response Organization
°F	degrees Fahrenheit
F&O	facts and observations
FAQ	frequently asked question
FDT	fire dynamics tool
FLEX	diverse and flexible coping strategy
FM	fire modeling
FPE	fire protection engineering
FPP	fire protection program
FPRA	fire probabilistic risk assessment
FR	*Federal Register*
FRE	fire risk evaluation
GDC	general design criterion
HEP	human error probability
HFE	human failure event
HGL	hot gas layer
HRA	human reliability analysis
HRE	higher risk evolution
HRR	heat release rate
IEEE	Institute of Electrical and Electronics Engineers
JACQUE-FIRE	Joint Assessment of Cable Damage and Quantification of Effects from Fire
KSF	key safety function
LAR	license amendment request

LERF	large early release frequency
MCA	multi-compartment analysis
MCR	main control room
MSO	multiple spurious operation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NPO	non-power operation
NRC	U.S. Nuclear Regulatory Commission
NSCA	nuclear safety capability assessment
NSPC	nuclear safety performance criteria
OCS	oil collection system
PB	performance-based
PORV	pilot-operated relief valve
PRA	probabilistic risk assessment
RA	recovery action
RAI	requests for additional information
RCP	reactor coolant pump
RCS	reactor coolant system
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RI	risk-informed
RI/PB	risk-informed, performance-based
SBO	station blackout
SCBA	self-contained breathing apparatus
SE	safety evaluation
SFPE	Society of Fire Protection Engineers
SSA	safe shutdown analysis
SSC	structures, systems, and components
SSD	safe shutdown
TR	technical report
TS	technical specification
UFSAR	updated final safety analysis report
V&V	verification and validation
VFDR	variance from deterministic requirements
ZOI	zone of influence

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 298 TO ADOPT NATIONAL FIRE PROTECTION ASSOCIATION STANDARD 805 (CAC NO. MF7190, EPID L-2015-LLF-0001) DATED JUNE 21, 2019

DISTRIBUTION:
PUBLIC
RidsNrrDorlLpl3 Resource
RidsRgn3MailCenter Resource
RidsNrrLASRohrer Resource
RidsAcrs_MailCTR Resource
RidsNrrPMDavisBesse Resource
RidsNrrDraApla Resource
RidsNrrDraAplb Resource
RidsNrrDraArcb Resource
RidsNrrDssStsb Resource
JRobinson, NRR
JSHyslop, NRR
BMetzger, NRR
NIqbal, NRR
CMoulton, NRR
THillsmeier, NRR
CTilton, NRR

ADAMS Accession No: ML19100A306

*by email

OFFICE	NRR/LPL3/PM	NRR/LPL3/LA	NRR/APLB/BC	NRR/STSB/BC(A)
NAME	BPurnell	SRohrer	GCasto*	PSnyder*
DATE	5/30/19	4/24/19	4/8/19	4/26/19
OFFICE	OGC (NLO)*	NRR/LPL3/BC(A)	NRR/LPL3/PM	
NAME	ANaber	LRegner	BPurnell	
DATE	5/29/19	6/20/19	6/21/19	

OFFICIAL RECORD COPY